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Revision 05B

MAGNASTOR

(Modular Advanced Generation Nuclear All-purpose STORage)

SAFETY ANALYSIS REPORT

Docket No. 72-1031

NAC INTERNATIONAL

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Chapter 1

1.3 General Description of MAGNASTOR

MAGNASTOR provides for the long-term storage of PWR and BWR fuel assemblies as listed in Chapter 2. During long-term storage, the system provides an inert environment, passive structural shielding, cooling and criticality control, and a welded confinement boundary. The structural integrity of the system precludes the release of contents in any of the design basis normal conditions and off-normal or accident events, thereby assuring public health and safety during use of the system.

1.3.1 MAGNASTOR Components

The design and operation of the principal components of MAGNASTOR and the associated auxiliary equipment are described in this section. The design characteristics of the principal components of the system are presented in Table 1.3-1.

This list shows the auxiliary equipment generally needed to use MAGNASTOR.

- automated, remote, and /or manual welding equipment to perform TSC field closure welding operations
- an engine-driven or towed frame or a heavy-haul trailer to move the concrete cask to and from the storage pad and to position the concrete cask on the storage pad
- draining, drying, helium backfill, and water cooling systems for preparing the TSC and contents for storage
- hydrogen monitoring equipment to confirm the absence of explosive or combustible gases during TSC closure welding
- an adapter plate and a hydraulic supply system
- a lifting yoke for lifting and handling the transfer cask and rigging equipment for lifting and handling system components

In addition to these items, the system requires utility services (electric, helium, air, clean borated water, nitrogen gas supply, etc.), standard torque wrenches, tools and fittings, and miscellaneous hardware.

1.3.1.1 Transportable Storage Canister (TSC)

Two lengths of TSCs accommodate all evaluated PWR and BWR fuel assemblies. The TSC is designed for transport per 10 CFR 71 [3]. The load conditions in transport produce higher stresses in the TSC than are produced during storage conditions, except for TSC lifting.

Consequently, transport load conditions establish the design basis for the TSC and, therefore, the TSC design is conservative with respect to storage conditions.

The stainless steel TSC assembly holds the fuel basket structure and confines the contents (see Figure 1.3-2). The TSC is defined as the confinement boundary during storage. The welded closure lid, closure ring, and redundant port covers prevent the release of contents under normal conditions and off-normal or accident events. The fuel basket assembly provides the structural support and a heat transfer path for the fuel assemblies, while maintaining a subcritical configuration for all of the evaluated normal conditions and off-normal or accident events.

The major components of the TSC assembly are the shell, base plate, closure lid, closure ring, and redundant port covers for the vent and drain ports, which provide the confinement boundary during storage. The TSC component dimensions and materials of fabrication are provided in Table 1.3-1. The TSC overall dimensions and design parameters for the two lengths of TSCs are provided in Table 1.3-2.

The TSC consists of a cylindrical stainless steel shell with a welded stainless steel bottom plate at its closed end and a 9-in thick stainless steel closure lid at its open end. The stainless steel shell and bottom plate are dual-certified Type 304/304L. The closure lid, closure ring and port covers are Type 304 stainless steel. A fuel basket assembly is placed inside the TSC. The closure lid is positioned inside the TSC on the lifting lugs above the basket assembly following fuel loading. After the closure lid is placed on the TSC, the TSC is moved to a workstation, and the closure lid is welded to the TSC. After nondestructive examination and pressure testing of the closure lid weld, the closure ring is welded to the closure lid and TSC shell. The vent and drain ports are penetrations through the lid, which provide access for auxiliary systems to drain, dry, and backfill the TSC. The drain port has a threaded fitting for installing the drain tube. The drain tube extends the full length of the TSC and ends in a sump in the bottom plate. The vent port also provides access to the TSC cavity for draining, drying, and backfilling operations. Following completion of backfilling, the redundant port covers at each of the ports are installed and welded in place. Each of the port cover welds in nondestructively examined.

The TSC is designed, fabricated, and inspected to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division 1, Subsection NB [8], except as noted in the Alternatives to the ASME Code as provided in Table 2.1-2.

Refer to Table 1.3-3 for a summary of the TSC fabrication requirements.

1.8 License Drawings

This section presents the list of License Drawings for MAGNASTOR.

Drawing Number	Title	Revision No.
71160-551	Fuel Tube Assembly, MAGNASTOR – 37 PWR	3
71160-560	Assembly, Standard Transfer Cask, MAGNASTOR	1
71160-561	Structure, Weldment, Concrete Cask, MAGNASTOR	3
71160-562	Reinforcing Bar and Concrete Placement, Concrete Cask, MAGNASTOR	2
71160-571	Details, Neutron Absorber, Retainer, MAGNASTOR – 37 PWR	2
71160-572	Details, Neutron Absorber, Retainer, MAGNASTOR – 87 BWR	2
71160-574	Basket Support Weldments, MAGNASTOR – 37 PWR	2
71160-575	Basket Assembly, MAGNASTOR – 37 PWR	4
71160-581	Shell Weldment, Canister, MAGNASTOR	2
71160-584	Details, Canister, MAGNASTOR	2
71160-585	TSC Assembly, MAGNASTOR	3
71160-590	Loaded Concrete Cask, MAGNASTOR	3
71160-591	Fuel Tube Assembly, MAGNASTOR – 87 BWR	3
71160-598	Basket Support Weldments, MAGNASTOR – 87 BWR	3
71160-599	Basket Assembly, MAGNASTOR – 87 BWR	3
71160-600	Basket Assembly, MAGNASTOR – 82 BWR	1

Figure Withheld Under 10 CFR 2.390

571-3	
571-4	

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY,
MAGNASTOR - 37 PWR

A

PROJECT	DRAWING	REV
71160	551	3
	SH 1 OF 2	

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY,
MAGNASTOR – 37 PWR

A

PROJECT 71160	DRAWING 551	REV 3
	SH 2 OF 2	

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 37 PWR

A

71160	575	4
	1 of 3	

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 37 PWR

PROJECT 71160	DRAWING 575	REV 4
	SH 2 OF 3	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 37 PWR

A

71160 | 575 | 4

SH 3 OF 3

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TSC ASSEMBLY, MAGNASTOR

A

71160

585

3

1 of 2

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TSC ASSEMBLY,
MAGNASTOR

A

PROJECT 71160	DRAWING 585	REV 3
	SH 2 OF 2	

1

Figure Withheld Under 10 CFR 2.390

	PLATE
	PLATE
	DESCRIPTION

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY,
MAGNASTOR – 87 BWR

A

71160	DRAWING 591	REV 3	
	SH 1 OF 2		

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY, MAGNASTOR – 87 BWR

A

71160	DRAWING 591	REV 3	
	SH 2 OF 2		

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 87 BWR

PROJECT 71160	DRAWING 599	REV 3	
	SH 1 OF 3		

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL			
BASKET ASSEMBLY, MAGNASTOR – 87 BWR			
71160		DRAWING 599	REV 3
		SH 2 OF 3	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY, MAGNASTOR - 87 BWR

71160	599	3
	3 of 3	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 82 BWR

71160

600

1

A

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 82 BWR

A

71160	DRAWING 600	REV 1
	SH 2 OF 5	

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL		
BASKET ASSEMBLY, MAGNASTOR – 82 BWR		
71160	600	1
	SH 3 OF 5	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 82 BWR

A

PROJECT 71160	DRAWING 600	REV 1
	SH 4 OF 5	

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET ASSEMBLY,
MAGNASTOR – 82 BWR

A

71160	DRAWING 600	REV 1
	SH 5 OF 5	

1

Chapter 2

2.1 MAGNASTOR System Design Criteria

The design of MAGNASTOR ensures that the stored spent fuel is maintained subcritical in an inert environment, within allowable temperature limits, and is retrievable. The acceptance testing and maintenance program specified in Chapter 10 ensures that the system is, and remains, suitable for the intended purpose. The MAGNASTOR design criteria appear in Table 2.1-1.

Approved alternatives to the ASME Code for the design procurement, fabrication, inspection, and testing of MAGNASTOR TSCs and spent fuel baskets are listed in Table 2.1-2.

Proposed alternatives to ASME Code, Section III, 2001 Edition with Addenda through 2003, including alternatives listed in Table 2.1-2, may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate the following.

- The proposed alternatives would provide an acceptable level of quality and safety, or Compliance with the specified requirements of ASME Code, Section III, Subsections NB and NG, 2001 Edition with Addenda through 2003, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.
- Requests for alternatives shall be submitted in accordance with 10 CFR 72.

Table 2.1-1 MAGNASTOR System Design Criteria

Parameter	Criteria
Design Life	50 years
Design Code – Confinement	
TSC	ASME Code, Section III, Subsection NB [1] for confinement boundary
TSC Cavity Atmosphere	Helium
Gas Pressure	7.0 atmospheres gauge (103 psig)
Design Code - Nonconfinement	
Fuel Basket	ASME Code, Section III, Subsection NG [2] and NUREG/CR-6322 [3]
Concrete Cask	ACI-349 [4], ACI-318 [5]
Transfer Cask	ANSI N14.6 [6], NUREG-0612 [15]
Thermal	
Maximum Fuel Cladding Temperature	752°F (400°C) for Normal and Transfer [7] 1058°F (570°C) for Off-Normal and Accident [8]
Ambient Temperature	
Normal (average annual ambient)	100°F
Off-Normal (extreme cold; extreme hot)	-40°F; 106°F
Accident	133°F
Concrete Temperature	
Normal Conditions	≤150°F (bulk) [4]; ≤ 200°F (local) [9]
Off-Normal/Accident Conditions	≤ 350°F local/ surface [4]
Radiation Protection/Shielding	
Owner-Controlled Area Boundary Dose [10]	
Normal/Off-Normal Conditions	25 mrem (Annual Whole Body) [10]
Accident Whole Body Dose	5 rem (Whole Body) [10]

2.2 Spent Fuel To Be Stored

MAGNASTOR is designed to safely store up to 37 PWR or up to 87 BWR spent fuel assemblies, contained within a TSC. The fuel assemblies are assigned to two groups of PWR and two groups of BWR fuel assemblies on the basis of fuel assembly length. Refer to Chapter 1 for the fuel assembly length groupings. For TSC spent fuel content loads less than a full basket, empty fuel positions shall include an empty fuel cell insert.

Intact PWR and BWR fuel assemblies having parameters as shown in Table 2.2-1 and Table 2.2-2, respectively, may be stored in MAGNASTOR.

The minimum initial enrichment limits are shown in Table 2.2-1 and Table 2.2-2 for PWR and BWR fuel, respectively, and exclude the loading of fuel assemblies enriched to less than 1.3 wt% ^{235}U, including unenriched fuel assemblies. Fuel assemblies with unenriched axial end-blankets may be loaded into MAGNASTOR.

2.2.1 PWR Fuel Evaluation

MAGNASTOR evaluations are based on bounding PWR fuel assembly parameters that maximize the source terms for the shielding evaluations, the reactivity for criticality evaluations, the decay heat load for the thermal evaluations, and the fuel weight for the structural evaluations. These bounding parameters are selected from the various spent fuel assemblies that are candidates for storage in MAGNASTOR. The bounding fuel assembly values are established based primarily on how the principal parameters are combined, and on the loading conditions (or restrictions) established for a group of fuel assemblies based on its parameters. Each TSC may contain up to 37 intact PWR fuel assemblies.

The limiting parameters of the PWR fuel assemblies authorized for loading in MAGNASTOR are shown in Table 2.2-1. The maximum initial enrichments listed are based on a minimum soluble boron concentration of 2,500 ppm in the spent fuel pool water. Lower soluble boron concentrations are allowed in the spent fuel pool water for fuel assemblies with lower maximum enrichments. The maximum initial enrichment authorized represents the peak fuel rod enrichment for variably enriched PWR fuel assemblies. The PWR fuel assembly allowable loading characteristics are summarized by fuel assembly type in Table 6.4-1. Table 2.2-1 assembly physical information is limited to the critical analysis input of fuel mass, array configuration, and number of fuel rods. These analysis values are key inputs to the criticality and shielding evaluations in Chapters 5 and 6. Lattice parameters dictating system reactivity are detailed in Chapter 6. Enrichment limits are set for each fuel type to produce reactivities at the

upper safety limit (USL). The maximum TSC decay heat load for the storage of PWR fuel assemblies is 37 kW. Uniform and preferential loading patterns are allowed in the PWR basket. The uniform loading pattern permits assemblies with a maximum heat load of 1 kW/assembly. The preferential loading pattern permits peak heat loads of 1.30 kW, as indicated in the zone description in Figure 2.2-1. The bounding thermal evaluations are based on the Westinghouse 17×17 fuel assembly. The minimum cool times are determined based on the maximum decay heat load of the contents. The fuel assemblies and source terms that produce the maximum storage and transfer cask dose rates are summarized in Table 5.1-3. A bounding weight of 1,680 pounds, as shown in Table 2.2-1, based on a B&W 15×15 fuel assembly with control components inserted, has been structurally evaluated in each location of the PWR fuel basket.

As noted in Table 2.2-1, the evaluation of PWR fuel assemblies includes thimble plugs (flow mixers), burnable poison rod assemblies (BPRAs), control element assemblies (CEAs), and/or solid filler rods. Empty fuel rod positions are filled with a solid filler rod or a solid neutron absorber rod that displaces a volume not less than that of the original fuel rod.

2.2.2 BWR Fuel Evaluation

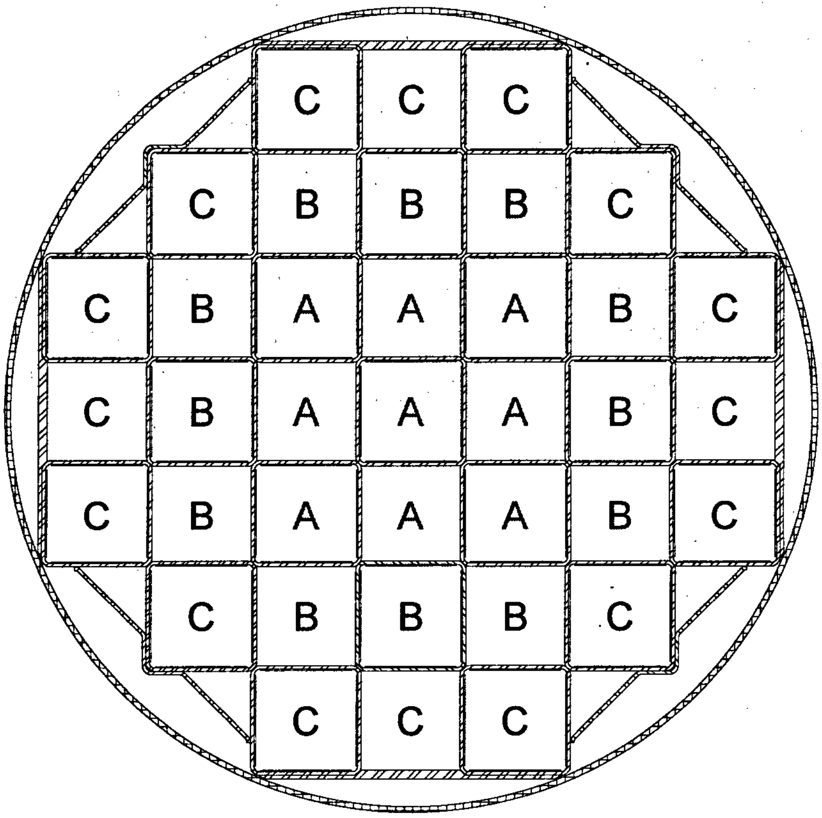
MAGNASTOR evaluations are based on bounding BWR fuel assembly parameters that maximize the source terms for the shielding evaluations, the reactivity for the criticality evaluations, the decay heat load for the thermal evaluations, and the fuel weight for the structural evaluations. These bounding parameters are selected from the various spent fuel assemblies that are candidates for storage in MAGNASTOR. The bounding fuel assembly values are established based primarily on how the principal parameters are combined, and on the loading conditions or restrictions established for a group of fuel assemblies based on its parameters. Each TSC may contain up to 87 intact BWR fuel assemblies. To increase allowed assembly enrichments over those determined for the 87-assembly basket configuration, an optional 82-assembly loading pattern may be used. The required fuel assembly locations in the 82-assembly pattern are shown in Figure 2.2-2.

The limiting parameters of the BWR fuel assemblies authorized for loading in MAGNASTOR are shown in Table 2.2-2. The minimum initial enrichment represents the peak planar-average enrichment. The BWR fuel assembly allowable loading characteristics are summarized by fuel type in Table 6.4-2. Table 2.2-1 assembly physical information is limited to the critical analysis input of fuel mass, array configuration, and number of fuel rods. These analysis values are key inputs to the criticality and shielding evaluations in Chapters 5 and 6. Lattice parameters dictating system reactivity are detailed in Chapter 6. Enrichment limits are set for each fuel type to produce reactivities at the upper safety limit (USL). The maximum decay heat load per TSC

for the storage of BWR fuel assemblies is 35.0 kW (average of 0.402 kW/assembly). Only uniform loading is permitted for BWR fuel assemblies. The bounding thermal evaluations are based on the GE 10×10 fuel assembly. The minimum cooling times are determined based on the maximum decay heat load of the contents. The fuel assemblies and source terms that produce the maximum storage and transfer cask dose rates are summarized in Table 5.1-3. A bounding weight of 704 pounds, as shown in Table 2.2-2, is based on the maximum weight of GE 7×7 and 8×8 assemblies with channels; this weight has been structurally evaluated in each storage location of the BWR basket.

As noted in Table 2.2-2, the evaluation of BWR fuel envelops unchanneled assemblies and assemblies with channels up to 120 mils thick. Empty fuel rod positions are filled with a solid filler rod or a solid neutron absorber rod that displaces a volume not less than that of the original fuel rod.

Figure 2.2-1 PWR Fuel Preferential Loading Zones



Zone Description	Designator	Heat Load (W/assy)	# Assemblies
Inner Ring	A	960	9
Middle Ring	B	1,300	12
Outer Ring	C	800	16

Figure 2.2-2 82-Assembly-BWR Basket Pattern

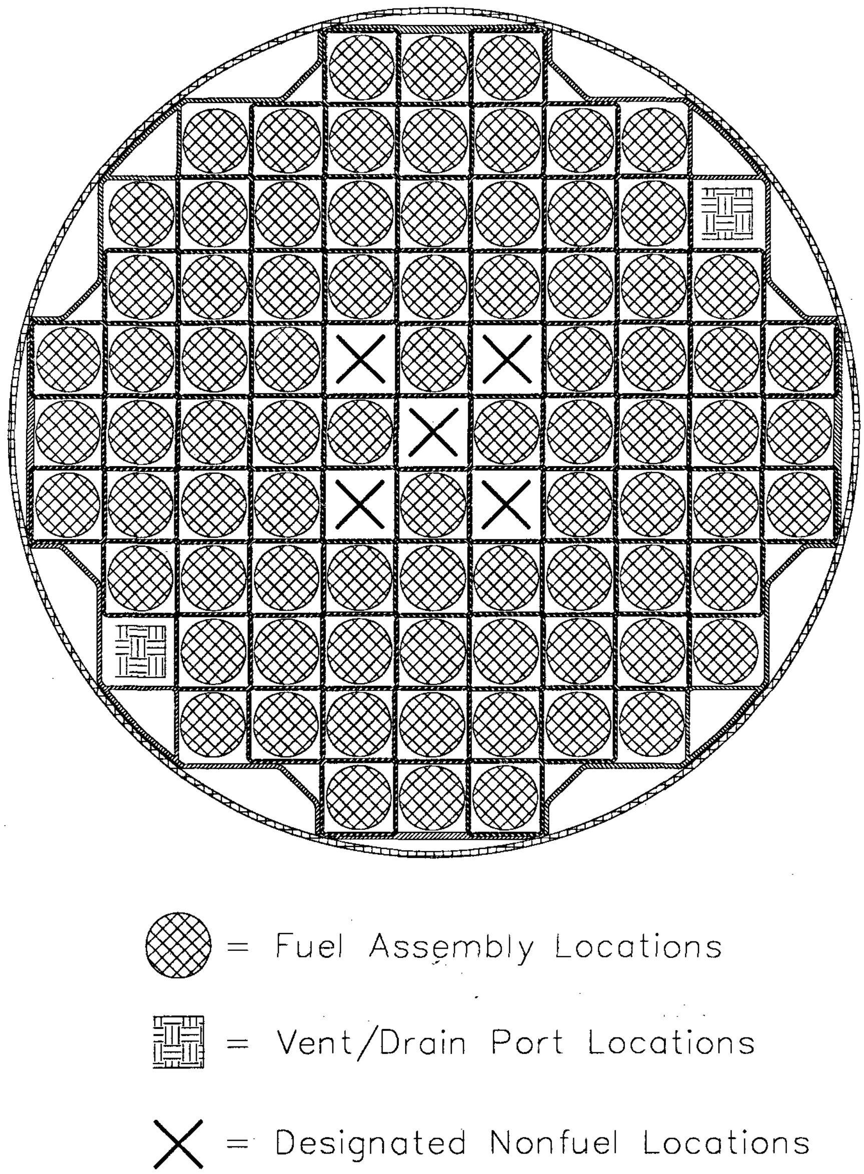


Table 2.2-1 PWR Fuel Assembly Characteristics

Characteristic	Fuel Class					
	14×14	14×14	15×15	15×15	16×16	17×17
Base Fuel Type[a]	CE, SPC	W, SPC	W, SPC	BW, FCF	CE	BW, SPC, W, FCF
Max Initial Enrichment (wt% ^{235}U)	5.0	5.0	5.0	5.0	5.0	5.0
Min Initial Enrichment (wt% ^{235}U)	1.3	1.3	1.3	1.3	1.3	1.3
Number of Fuel Rods	176	179	204	208	236	264
Max Assembly Average Burnup (MWd/MTU)	60,000	60,000	60,000	60,000	60,000	60,000
Peak Average Rod Burnup (MWd/MTU)	62,500	62,500	62,500	62,500	62,500	62,500
Min Cool Time (years)	4	4	4	4	4	4
Max Weight (lb) per Storage Location	1,680	1,680	1,680	1,680	1,680	1,680
Max Decay Heat (Watts) per Storage Location	1,300	1,300	1,300	1,300	1,300	1,300

- Fuel cladding is a zirconium-based alloy.
- All reported enrichment values are nominal preirradiation fabrication values.
- Weight includes the weight of nonfuel-bearing components.
- Assemblies may contain a flow mixer (thimble plug), a burnable poison rod assembly, a control element assembly, and/or solid stainless steel or zirconium-based alloy filler rods.
- Maximum initial enrichment is based on a minimum soluble boron concentration in the spent fuel pool water. Required soluble boron content is fuel type and enrichment specific. Minimum soluble boron content varies between 1,500 and 2,500 ppm. Maximum initial enrichment represents the peak fuel rod enrichment for variably-enriched fuel assemblies.
- Spacers may be used to axially position fuel assemblies to facilitate handling.

[a] Indicates assembly and/or nuclear steam supply system (NSSS) vendor/type referenced for fuel input data. Fuel acceptability for loading is not restricted to the indicated vendor provided that the fuel assembly meets the limits listed in Table 6.4-1. Table 6.2-1 contains vendor information by fuel rod array. Abbreviations are as follows: Westinghouse (W), Combustion Engineering (CE), Siemens Power Corporation (SPC), Babcock and Wilcox (BW), and Framatome Cogema Fuels (FCF).

Table 2.2-2 BWR Fuel Assembly Characteristics

Characteristic	Fuel Class			
	7×7	8×8	9×9	10×10
Base Fuel Type[a]	SPC, GE	SPC, GE	SPC, GE	SPC, GE, ABB
Max Initial Enrichment (wt% ^{235}U)	4.5	4.5	4.5	4.5
Number of Fuel Rods	48 49	59 60 61 62 63 64[b]	72 74[c,d] 76 79 80	91[c] 92[c] 96[c,d] 100[d]
Max Assembly Average Burnup (MWd/MTU)	60,000	60,000	60,000	60,000
Peak Average Rod Burnup (MWd/MTU)	62,500	62,500	62,500	62,500
Min Cool Time (years)	4	4	4	4
Min Average Enrichment (wt% ^{235}U)	1.3	1.3	1.3	1.3
Max Weight (lb) per Storage Location	704	704	704	704
Max Decay Heat (Watts) per Storage Location	402	402	402	402

- Each BWR fuel assembly may have a zirconium-based alloy channel up to 120 mil thick.
- Assembly weight includes the weight of the channel.
- Maximum initial enrichment is the peak planar-average enrichment.
- Water rods may occupy more than one fuel lattice location. Fuel assembly to contain nominal number of water rods for the specific assembly design.
- All enrichment values are nominal preirradiation fabrication values.
- Spacers may be used to axially position fuel assemblies to facilitate handling.

[a] Indicates assembly vendor/type referenced for fuel input data. Fuel acceptability for loading is not restricted to the indicated vendor/type provided that the fuel assembly meets the limits listed in Table 6.4-2. Table 6.2-2 contains vendor information by fuel rod array. Abbreviations are as follows: General Electric/Global Nuclear Fuels (GE), Exxon/Advanced Nuclear Fuels/Siemens Power Corporation (SPC).

[b] May be composed of four subchannel clusters.

[c] Assemblies may contain partial-length fuel rods.

[d] Composed of four subchannel clusters.

Chapter 3

Chapter 3
Structural Evaluation

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This DLF is applied to the end drop acceleration of 60g, which is the bounding load to potentially result in the buckling of the fuel rod. The product of 60g × DLF (= 14.6g) is well below the vertical acceleration corresponding to the first buckling mode shape, 37.9g as computed in this section. This indicates that the time duration of the impact of the fuel onto the fuel assembly base is of sufficiently short nature that buckling of the fuel rod cannot occur.

An effective cross-sectional property is used in the model to consider the properties of the fuel pellet and the fuel cladding. The modulus of elasticity (EX) for the fuel pellet has a nominal value of 26.0×10^6 psi [25]. To be conservative, only 50 percent of this value is used in the evaluation. The EX for the fuel pellet was, therefore, taken to be 13.0×10^6 psi. The value of EX (10.47×10^6 psi) was used for the irradiated zirconium alloy cladding [26]. The analysis contained in Reference 27 used an elastic modulus of 12.1×10^6 psi, which is bounded by the above value. Reference 28 information shows that there is no additional reduction of the ductility of the cladding due to extended burnup in the 45,000 – 50,000 MWd/MTU range.

The bounding dimensions and physical data (minimum clad thickness, maximum rod length and minimum number of support grids) for the MAGNASTOR fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396

The cladding thickness is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches.

The elevation of the grids, measured from the bottom of the fuel assembly, are: 2.3, 33.0, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 (inches).

The effective cross-sectional properties (EI_{eff}) for the beam are computed by adding the value of EI for the cladding and the pellet, where:

E = modulus of elasticity (lb/in^2)
I = cross-sectional moment of inertia (in^4)

The lowest frequency for the extentional mode shape was computed to be 219.0 Hz. The first mode shape corresponds to a frequency of 25.9 Hz. Using the expression for the DLF previously discussed, the DLF is computed to be 0.240 ($\beta = 8.44$).

120 Micron Cladding Oxide Layer Thickness Evaluation

The buckling calculation used the same model employed for the mode shape calculation. The load that would potentially buckle the fuel rod in the end drop is due to the deceleration of the

rod. This loading was implemented by applying a 1g acceleration in the direction that would result in compressive loading of the fuel rod. The acceleration required to buckle the fuel rod is computed to be 37.3g, which is much higher than the calculated effective g-load (14.3g) due to the 60g end drop. Therefore, the fuel rods with a 120 micron cladding oxide layer do not buckle in the 60g end drop event.

3.8.2 BWR Fuel Rod Evaluation

The evaluation of the BWR fuel rod is based on the following representative sample of BWR fuel rods:

Fuel Assembly	Cladding Diameter (in)	Cladding Thickness (in)	Cladding Material	Pellet Diameter (in)	Rod Length (in)
GE 7x7	0.563	0.032	Zirc-2	0.487	158.15
GE 7x7	0.563	0.032	Zirc-2	0.487	163.42
GE 8x8-2	0.483	0.032	Zirc-2	0.410	158.67
GE 8x8-2	0.483	0.032	Zirc-2	0.410	163.42
GE 8x8-4	0.484	0.032	Zirc-2	0.410	163.42
GE 8x8-4	0.484	0.032	Zirc-2	0.410	163.42
GE 9x9-2	0.441	0.028	Zirc-2	0.376	163.42
GE 10x10-2	0.378	0.024	Zirc-2	0.322	163.42

The location of the lateral constraints in the BWR fuel are: 0.00 in, 22.88 in, 43.03 in, 63.18 in, 83.33 in, 103.48 in, 122.3 in, 143.78 in, and 163.42 in.

For the PWR fuel rod the largest ratio of unsupported length (L) to radius of gyration of the cladding cross section (r) [13] is

$$L/r = \frac{30.7}{0.5 \times \sqrt{(.434/2)^2 + (.388/2)^2}} = 211$$

The ratio (L/r) for a BWR fuel rod is

$$L/r = \frac{22.88}{0.5 \times \sqrt{(.378/2)^2 + (.330/2)^2}} = 182$$

The analysis presented in Section 3.8.1 is bounding for both PWR and BWR fuel rods, because the (L/r) for the PWR fuel rod is larger than the (L/r) for the BWR fuel rod. Therefore, no further evaluation of the BWR fuel rod is required.

3.9 References

1. 10 CFR 72, Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Fuel, High Level Radioactive Waste and Reactor-Related Greater than Class C Waste," US Nuclear Regulatory Commission, Washington, DC.
2. American National Standard for Radioactive Materials N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," American National Standard Institute, Inc., Washington, DC, 1993.
3. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Washington, D.C., 1980.
4. ANSI/ANS-57.9-1992, American National Standard Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type), American Nuclear Society, La Grange Park, IL, May 1992.
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6. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, "Class 1 Components," American Society of Mechanical Engineers, New York, NY, 2001 Edition with 2003 Addenda.
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8. ASME Boiler and Pressure Vessel Code, Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits," The American Society of Mechanical Engineers, New York, NY, 2001, with 2003 Addenda.
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11. "Steel Structures Design and Behavior," C.G. Salmon and J.E. Johnson, Harper and Row Publishers, New York, NY, Second Edition, 1980.
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3.10.8 Basket Pin-Socket Connection Evaluation for Concrete Cask Tip-Over Accident Condition

This section describes the structural evaluation for the pin-socket connections for the PWR and BWR fuel baskets using the LS-DYNA program for the hypothetical concrete cask tip-over accident.

As shown in Figure 3.10.8-1, a three-dimensional half-symmetry finite element model of the BWR fuel basket is used for the evaluation. The BWR basket is considered to be the bounding configuration based on slightly higher shear stresses at the pin-socket connections determined by the quasi-static analyses for BWR and PWR baskets for the tip-over accident as documented in Section 3.7.2. The model is constructed using the identical methods and boundary conditions for the models used for the basket stability evaluation as presented in Section 3.10.6. This model corresponds to the 0-degree basket orientation, which is associated with the maximum shear forces at the pin-socket connection, and represents a half cross-section of the loaded BWR basket, the canister shell and the concrete cask steel liner. The LS-DYNA program is used to perform transient (time history) analyses to simulate the side impact condition of the cask tip-over accident. Three cases are considered in the analysis:

Case No.	Gap between Flats at Tube Connors	Dynamic Properties Used for the Tubes and Pins
1	0.03 inch	Yes
2	0.06 inch	Yes
3	0.06 inch	No

Note that the maximum allowable gap between the flats at the tube corners is 0.03 inch. Cases 2 and 3 conservatively use a gap of 0.06 inch. To maximize the effect of these gaps, only the tube dimension for the "Interior Tubes" in the model is reduced to reflect the gap. The "Interior Tubes" correspond to the tubes not attached to the side or corner support weldments. Piecewise linear plastic properties are used in the model for the SA 537 Class 1 carbon steel for the tubes and pins. The dynamic (strain rate) effect on the properties is considered for Cases 1 and 2.

To more accurately solve for the stress/strain at the pin-socket connections, the pin-socket connection associated with the largest shear load is modeled with a finer mesh as shown in Figure 3.10.8-1 and Figure 3.10.8-2.

The calculated maximum plastic strains at this pin-socket connection are shown in the following table. See Figure 3.10.8-2 for typical locations of the maximum strains in the pin and the upper and lower tube sockets.

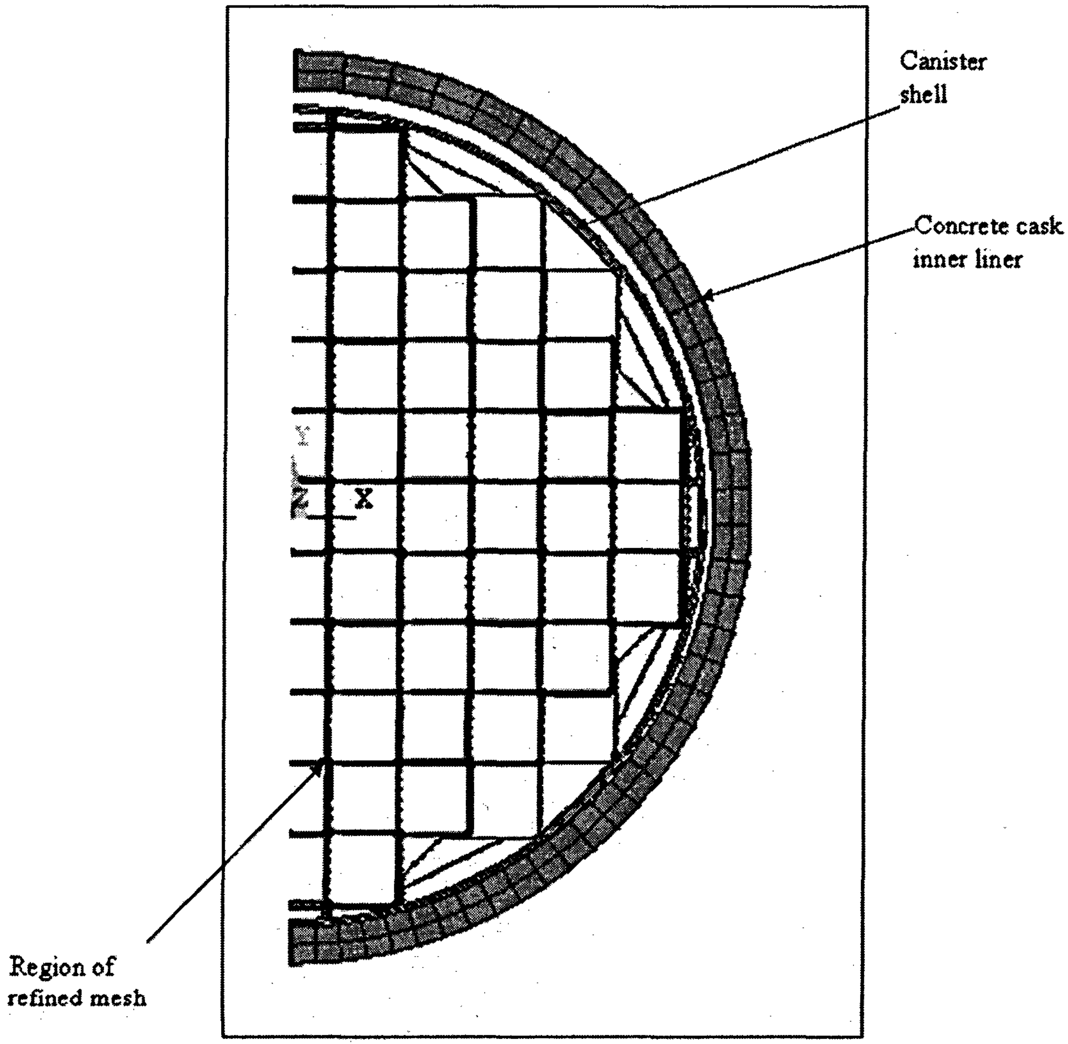
	Maximum Plastic Strain		
Case No.	Upper Tube Socket	Lower Tube Socket	Pin
1	7.1%	9.0%	6.2%
2	9.8%	8.9%	8.2%
3	9.3%	7.6%	7.4%

The comparison of results for Cases 1 and 2 in the above table indicates that the effect of gap size at the tube corners on the plastic strain of the pin and tube socket is not significant.

The calculated maximum plastic strains are similar for the analysis using dynamic (strain rate sensitive) properties (Case 2) and the analysis using static properties (Case 3). The maximum plastic strain for Case 3 is slightly lower due to the fact that the dynamic properties employed in the analysis for Case 2 are conservatively established. For a given strain, the stress for the dynamic properties (Case 2) is lower than the stress used for the static properties (Case 3).

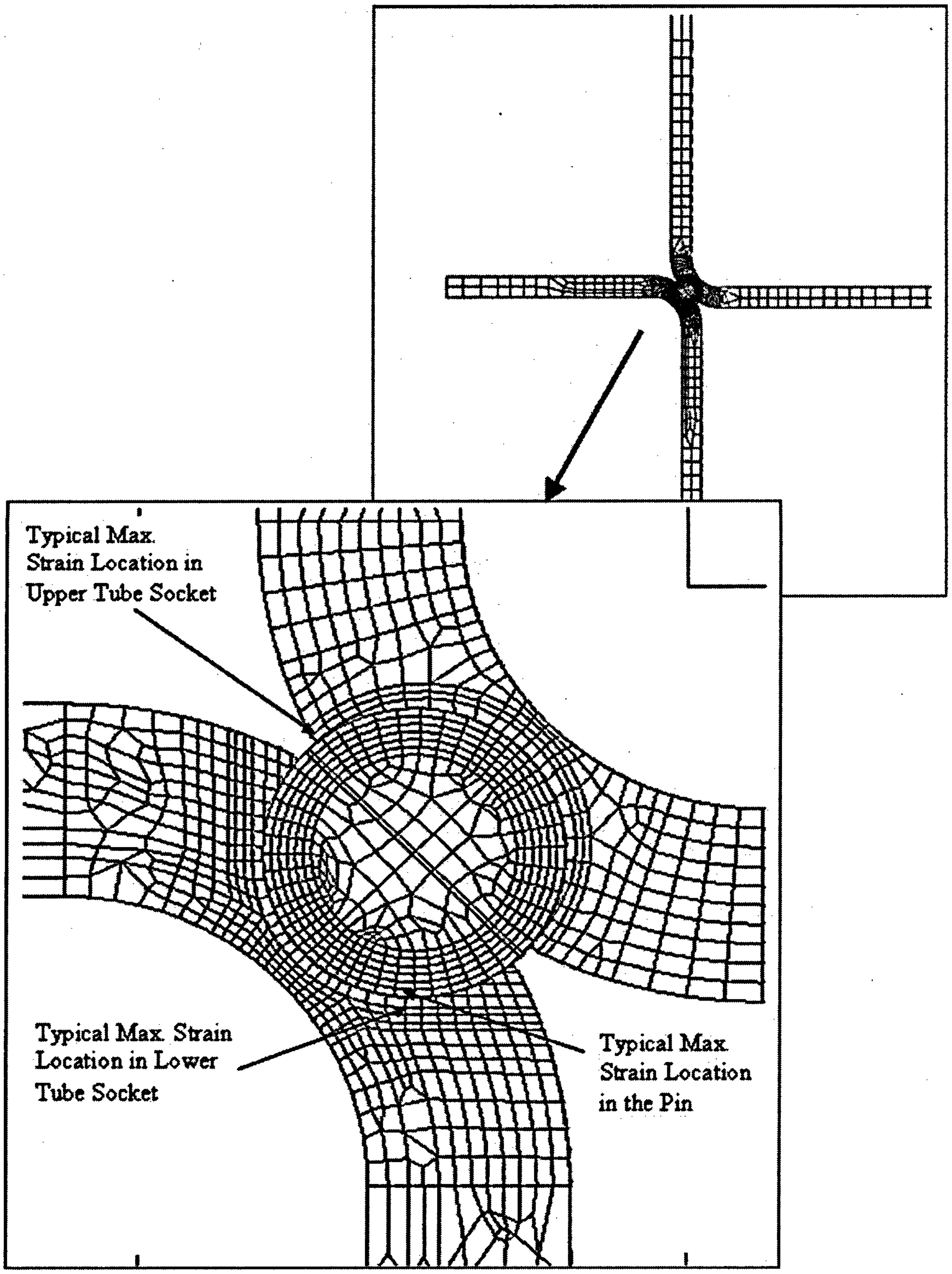
As shown in the table, the calculated maximum plastic strains at the pin-socket connection for all of the analyzed cases are well below the material ultimate true strain of the SA 537 carbon steel (20%). Also note that the analysis results indicate that the plastic strain in the tube socket or pin occurs in localized regions and the majority of the material remains elastic. The pins remain completely enclosed in the tube sockets and allow transfer of shear and bearing loads between adjacent fuel tubes during the cask tip-over accident. Therefore, the pin-socket connections of the BWR and PWR fuel baskets are structurally adequate for the concrete cask tip-over accident condition.

Figure 3.10.8-1 Half-Symmetry Finite Element Model for the Basket Pin-Socket Connection Evaluation



Elements for the fuel assemblies are not shown.

Figure 3.10.8-2 Region with the Refined Mesh at the Pin-Socket Connection



Chapter 6

Table 6.2-1 PWR Fuel Assembly Characteristics

Fuel ID			CE14H1	CE16H1	WE14H1	WE15H1	WE15H2	WE17H1	WE17H2	BW15H1	BW15H2	BW15H3	BW15H4	BW17H1
No. Fuel Rods			176	236	179	204	204	264	264	208	208	208	208	264
Base Fuel Type[a]			CE,SPC	CE	W,SPC	W,SPC	W,SPC	W,SPC	W,SPC	BW,FCF	BW,FCF	BW,FCF	BW,FCF	BW,FCF
Pitch	Max	(in)	0.5800	0.5063	0.5560	0.5630	0.5630	0.4960	0.4960	0.5680	0.5680	0.5680	0.5680	0.5020
	Min	(in)	0.5800	0.5063	0.5560	0.5630	0.5630	0.4960	0.4960	0.5680	0.5680	0.5680	0.5680	0.5020
Fuel Pellet OD	Max	(in)	0.3805	0.3250	0.3674	0.3669	0.3570	0.3232	0.3088	0.3686	0.3735	0.3742	0.3622	0.3252
	Min	(in)	0.3700	0.3250	0.3444	0.3565	0.3570	0.3225	0.3030	0.3686	0.3735	0.3707	0.3622	0.3232
Fuel Rod OD	Max	(in)	0.4400	0.3820	0.4240	0.4240	0.4170	0.3740	0.3600	0.4300	0.4300	0.4280	0.4140	0.3790
	Min	(in)	0.4400	0.3820	0.4000	0.4220	0.4170	0.3720	0.3600	0.4300	0.4300	0.4280	0.4140	0.3770
Fuel Clad Thick.	Max	(in)	0.0310	0.0250	0.0300	0.0300	0.0265	0.0225	0.0250	0.0265	0.0250	0.0245	0.0220	0.0240
	Min	(in)	0.0260	0.0250	0.0162	0.0242	0.0265	0.0205	0.0225	0.0265	0.0250	0.0230	0.0220	0.0220
Guide Tube OD	Max	(in)	1.115	0.980	0.481	0.544	0.484	0.482	0.482	0.493	0.493	0.493	0.493	0.420
	Min	(in)	1.115	0.970	0.481	0.484	0.484	0.482	0.480	0.493	0.493	0.493	0.493	0.420
Guide Tube Thick.	Max	(in)	0.040	0.035	0.034	0.017	0.017	0.015	0.016	0.016	0.015	0.014	0.014	0.020
	Min	(in)	0.036	0.035	0.017	0.015	0.017	0.014	0.015	0.016	0.015	0.014	0.014	0.018
Active Fuel Length	Max	(in)	137.0	150.0	145.2	144.0	144.0	144.0	144.0	144.0	144.0	144.0	144.0	144.0
	Min	(in)	134.0	150.0	142.0	144.0	144.0	144.0	144.0	144.0	144.0	144.0	144.0	143.0
Fuel Mass	Max	(MTU)	0.4167	0.4463	0.4188	0.4720	0.4469	0.4740	0.4327	0.4858	0.4988	0.5006	0.4690	0.4799
	Min	(MTU)	0.3854	0.4463	0.3599	0.4457	0.4469	0.4720	0.4166	0.4858	0.4988	0.4913	0.4690	0.4707

- Fuel assembly characteristics represent cold, unirradiated, nominal fuel dimension.
- An instrument tube may be located in the center of the assembly. The instrument tube may have slightly different diameter and thickness than the guide tubes. As the instrument tube is limited to one per assembly, dimensional variations have no significant effect on system reactivity and are not listed here.
- Guide tubes may contain "dashpots" near the bottom of the active fuel region narrowing from the listed tube dimension. Stainless steel rod inserts may be installed to displace "dashpot water." Fuel assemblies containing these stainless steel rod inserts are addressed as a fuel assembly containing a nonfuel insert.
- Assemblies may contain unenriched axial blankets.

[a] Indicates assembly and/or nuclear steam supply system (NSSS) vendor/type referenced for fuel input data. Fuel acceptability for loading is not restricted to the indicated vendor provided that the fuel assembly meets the limits listed in Table 6.4-1. Abbreviations are as follows: Westinghouse (W), Combustion Engineering (CE), Siemens Power Corporation (SPC), Babcock and Wilcox (BW), and Framatome Cogema Fuels (FCF).

Table 6.2-2 BWR Fuel Assembly Characteristics

Fuel ID			B7_48A	B7_49A	B7_49B	B8_59A	B8_60A	B8_60B	B8_61B	B8_62A	B8_63A	B8_64A	B8_64B[a]
No. Fuel Rods			48	49	49	59	60	60	61	62	63	64	64
Base Fuel Type[b]			SPC	GE	GE	GE	GE	GE	GE	SPC	SPC	GE	GE
Pitch	Max	(in)	0.7380	0.7380	0.7380	0.6400	0.6417	0.6400	0.6400	0.6417	0.6420	0.6420	0.6090
	Min	(in)	0.7380	0.7380	0.7380	0.6400	0.6378	0.6378	0.6400	0.6400	0.6400	0.6420	0.6090
Fuel Pellet OD	Max	(in)	0.4900	0.4880	0.4910	0.4160	0.4110	0.4140	0.4140	0.4160	0.4195	0.4195	0.3913
	Min	(in)	0.4900	0.4770	0.4910	0.4160	0.4095	0.4095	0.4140	0.4045	0.4045	0.4195	0.3913
Fuel Rod OD	Max	(in)	0.5700	0.5700	0.5630	0.4930	0.4843	0.4843	0.4830	0.4843	0.4930	0.4840	0.4576
	Min	(in)	0.5700	0.5630	0.5630	0.4930	0.4840	0.4830	0.4830	0.4830	0.4840	0.4840	0.4576
Fuel Clad Thick.	Max	(in)	0.0360	0.0370	0.0320	0.0340	0.0320	0.0320	0.0300	0.0360	0.0360	0.0273	0.0290
	Min	(in)	0.0360	0.0320	0.0320	0.0340	0.0315	0.0300	0.0300	0.0290	0.0273	0.0273	0.0290
No. Water Rods			1	0	0	5	1	4	3	2	1	0	0
Water Rod OD		(in)	0.5700	0.0000	0.0000	0.4930	1.2598	0.4843	0.4830	0.5910	0.4930	0.0000	0.0000
Water Rod Thick.		(in)	Solid	N/A	N/A	0.0340	0.0394	N/A	N/A	N/A	0.0340	N/A	N/A
Active Fuel Length	Max	(in)	144.0	146.0	150.0	150.0	150.0	150.0	150.0	150.0	150.0	150.0	150.0
	Min	(in)	144.0	144.0	150.0	150.0	145.2	145.2	150.0	145.2	144.0	150.0	150.0
Fuel Mass	Max	(MTU)	0.1981	0.2034	0.2115	0.1828	0.1815	0.1841	0.1872	0.1921	0.1985	0.2017	0.1755
	Min	(MTU)	0.1981	0.1916	0.2115	0.1828	0.1744	0.1744	0.1872	0.1758	0.1772	0.2017	0.1755

- Assemblies may contain unenriched axial blankets.
- Water rods may occupy more than one lattice location. Square water rods are not modeled.
- Assembly lattice structure may be surrounded by a zirconium alloy channel up to 120 mil thickness.

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[a] Composed of four subchanneled clusters.

[b] Indicates assembly vendor/type referenced for fuel input data. Fuel acceptability for loading is not restricted to the indicated vendor/type provided that the fuel assembly meets the limits listed in Table 6.4-2. Abbreviations are as follows: General Electric/Global Nuclear Fuels (GE), Exxon/Advanced Nuclear Fuels/Siemens Power Corporation (SPC).

Table 6.2-2 BWR Fuel Assembly Characteristics (cont.)

Fuel ID			B9_72A	B9_74A	B9_76A	B9_79A	B9_80A	B10_91A	B10_92A	B10_96A[a]	B10_100A[a]
No. Fuel Rods			72	74	76	79	80	91	92	96	100
Base Fuel Type[b]			GE	GE	SPC	SPC	GE	SPC	GE	ABB	ABB
No. Partial Length Rods[c]			N/A	8	N/A	N/A	N/A	8	14	12	N/A
Pitch	Max	(in)	0.5720	0.5720	0.5720	0.5720	0.5720	0.5100	0.5100	0.4880	0.4880
	Min	(in)	0.5720	0.5660	0.5720	0.5660	0.5720	0.5100	0.5100	0.4880	0.4880
Fuel Pellet OD	Max	(in)	0.3740	0.3760	0.3750	0.3760	0.3565	0.3420	0.3455	0.3224	0.3224
	Min	(in)	0.3740	0.3565	0.3530	0.3565	0.3565	0.3420	0.3455	0.3224	0.3224
Fuel Rod OD	Max	(in)	0.4330	0.4410	0.4430	0.4410	0.4230	0.3957	0.4040	0.3780	0.3780
	Min	(in)	0.4330	0.4240	0.4170	0.4240	0.4230	0.3957	0.4040	0.3780	0.3780
Fuel Clad Thick.	Max	(in)	0.0260	0.0300	0.0310	0.0300	0.0295	0.0239	0.0260	0.0243	0.0243
	Min	(in)	0.0260	0.0239	0.0209	0.0239	0.0295	0.0239	0.0260	0.0243	0.0243
No. Water Rods			1	2	5	2	1	1	2	0	0
Water Rod OD		(in)	1.7160	0.9843	0.4430	0.4409	0.4230	1.5300	0.8080	0.0000	0.0000
Water Rod Thick.		(in)	N/A	0.0276	0.0120	0.0280	0.0200	N/A	0.0300	0.0310[d]	0.0340[d]
Active Fuel Length	Max	(in)	150.0	150.0	150.0	150.0	150.0	150.0	150.0	150.0	150.0
	Min	(in)	150.0	150.0	150.0	145.2	150.0	150.0	150.0	150.0	150.0
Fuel Mass	Max	(MTU)	0.1803	0.1873	0.1914	0.2000	0.1821	0.1906	0.1966	0.1787	0.1861
	Min	(MTU)	0.1803	0.1684	0.1696	0.1740	0.1821	0.1906	0.1966	0.1787	0.1861

- Assemblies may contain unenriched axial blankets.
- Water rods may occupy more than one lattice location. Square water rods are not modeled.
- Assembly lattice structure may be surrounded by a zirconium alloy channel up to 120 mil thickness.

[a] Composed of four subchanneled clusters.

[b] Indicates assembly vendor/type referenced for fuel input data. Fuel acceptability for loading is not restricted to the indicated vendor/type provided that the fuel assembly meets the limits listed in Table 6.4-2. Abbreviations are as follows: General Electric/Global Nuclear Fuels (GE), Exxon/Advanced Nuclear Fuels/Siemens Power Corporation (SPC).

[c] Assembly may contain partial length rods. If partial length rods are used, see location sketch in Figure 6.2-1.

[d] Thickness indicated is the subchannel thickness.

Table 6.4-1 PWR Fuel Basket Allowable Loading

Assembly Type	No. of Fuel Rods	No. of Guide Tubes[a]	Max Pitch (inch)	Min Clad OD (inch)	Min Clad Thick. (inch)	Max Pellet OD (inch)	Max Active Length (inch)	Max Load (MTU)	Max. Initial Enrichment (wt % ^{235}U) Soluble Boron 1500 ppm	Soluble Boron 1750 ppm	Soluble Boron 2000 ppm	Soluble Boron 2250 pm	Soluble Boron 2500 ppm
BW15H1	208	17	0.568	0.43	0.0265	0.3686	144.0	0.4858	3.8%	4.1%	4.4%	4.7%	5.0%
BW15H2	208	17	0.568	0.43	0.025	0.3735	144.0	0.4988	3.7%	4.1%	4.4%	4.7%	5.0%
BW15H3	208	17	0.568	0.428	0.023	0.3742	144.0	0.5006	3.7%	4.0%	4.3%	4.7%	4.9%
BW15H4	208	17	0.568	0.414	0.022	0.3622	144.0	0.4690	3.9%	4.2%	4.6%	4.9%	5.0%
BW17H1	264	25	0.502	0.377	0.022	0.3252	144.0	0.4799	3.8%	4.1%	4.4%	4.7%	5.0%
CE14H1	176	5	0.58	0.44	0.026	0.3805	137.0	0.4167	4.6%	4.9%	5.0%	5.0%	5.0%
CE16H1	236	5	0.5063	0.382	0.025	0.325	150.0	0.4463	4.5%	4.9%	5.0%	5.0%	5.0%
WE14H1	179	17	0.556	0.40	0.0162	0.3674	145.2	0.4188	4.7%	5.0%	5.0%	5.0%	5.0%
WE15H1	204	21	0.563	0.422	0.0242	0.3669	144.0	0.4720	3.9%	4.2%	4.6%	4.9%	5.0%
WE15H2	204	21	0.563	0.417	0.0265	0.357	144.0	0.4469	4.0%	4.4%	4.8%	5.0%	5.0%
WE17H1	264	25	0.496	0.372	0.0205	0.3232	144.0	0.4740	3.8%	4.1%	4.5%	4.8%	5.0%
WE17H2	264	25	0.496	0.36	0.0225	0.3088	144.0	0.4327	4.0%	4.4%	4.8%	5.0%	5.0%

- Assembly characteristics represent cold, unirradiated, nominal configurations.
- Specified soluble boron concentrations are independent of whether a fuel assembly contains a nonfuel insert.

[a] Combined number of guide and instrument tubes.

Table 6.4-2 BWR Fuel Basket Allowable Loading

Assembly Type	Number of Fuel Rods	Number of Partial Length Rods	Max Pitch (inch)	Min Clad OD (inch)	Min Clad Thick. (inch)	Max Pellet OD (inch)	Max Active Length (inch)	Max Loading (MTU)	87-Assy Max Enrichment (wt % ^{235}U)	82-Assy Max Enrichment (wt % ^{235}U)
B7_48A	48	N/A	0.7380	0.5700	0.03600	0.4900	144.0	0.1981	4.1%	4.5%
B7_49A	49	N/A	0.7380	0.5630	0.03200	0.4880	146.0	0.2034	3.9%	4.5%
B7_49B	49	N/A	0.7380	0.5630	0.03200	0.4910	150.0	0.2115	3.9%	4.5%
B8_59A	59	N/A	0.6400	0.4930	0.03400	0.4160	150.0	0.1828	4.0%	4.5%
B8_60A	60	N/A	0.6417	0.4840	0.03150	0.4110	150.0	0.1815	3.9%	4.5%
B8_60B	60	N/A	0.6400	0.4830	0.03000	0.4140	150.0	0.1841	3.9%	4.5%
B8_61B	61	N/A	0.6400	0.4830	0.03000	0.4140	150.0	0.1872	3.9%	4.5%
B8_62A	62	N/A	0.6417	0.4830	0.02900	0.4160	150.0	0.1921	3.9%	4.5%
B8_63A	63	N/A	0.6420	0.4840	0.02725	0.4195	150.0	0.1985	3.8%	4.5%
B8_64A	64	N/A	0.6420	0.4840	0.02725	0.4195	150.0	0.2017	3.9%	4.5%
B8_64B[a]	64	N/A	0.6090	0.4576	0.02900	0.3913	150.0	0.1755	3.7%	4.4%
B9_72A	72	N/A	0.5720	0.4330	0.02600	0.3740	150.0	0.1803	3.8%	4.5%
B9_74A	74[b]	8	0.5720	0.4240	0.02390	0.3760	150.0	0.1873	3.7%	4.4%
B9_76A	76	N/A	0.5720	0.4170	0.02090	0.3750	150.0	0.1914	3.6%	4.3%
B9_79A	79	N/A	0.5720	0.4240	0.02390	0.3760	150.0	0.2000	3.7%	4.5%
B9_80A	80	N/A	0.5720	0.4230	0.02950	0.3565	150.0	0.1821	3.9%	4.5%
B10_91A	91[b]	8	0.5100	0.3957	0.02385	0.3420	150.0	0.1906	3.8%	4.5%
B10_92A	92[b]	14	0.5100	0.4040	0.02600	0.3455	150.0	0.1966	3.8%	4.5%
B10_96A[a]	96[b]	12	0.4880	0.3780	0.02430	0.3224	150.0	0.1787	3.7%	4.4%
B10_100A[a]	100	N/A	0.4880	0.3780	0.02430	0.3224	150.0	0.1861	3.7%	4.5%

Note:

- Assembly characteristics represent cold, unirradiated, nominal configurations.
- Maximum channel thickness allowed is 120 mils (nominal).

[a] Composed of four subchannel clusters.

[b] Assemblies may contain partial length fuel rods. Partial length rod assemblies are evaluated by removing partial length rods from the lattice. This configuration bounds an assembly with full length rods and combinations of full and partial length rods.

Chapter 7

7.1 Confinement Boundary

The welded TSC is the confinement vessel for the PWR or BWR spent fuel assembly contents.

The confinement boundary of the TSC consists of the TSC shell, bottom plate, TSC closure lid, closure ring, the redundant vent and drain port covers, and the welds that join these components. The confinement boundary is shown in Figure 7.1-1. The confinement boundary does not incorporate bolted closures or mechanical seals. The confinement boundary welds are described in Table 7.1-1.

7.1.1 Confinement Vessel

The TSC consists of three principal components: the TSC shell, bottom plate, and closure lid. The TSC shell is a right circular cylinder constructed of rolled Type 304/304L (dual certified) stainless steel plate with the edges of the plate joined by full penetration welds. It is closed at the bottom end by a circular plate joined to the shell by a full penetration weld. The TSC has two lengths to accommodate different fuel lengths. The TSC shell is helium leak tested following fabrication.

After loading, the TSC is closed at the top by a closure lid fabricated from Type 304 stainless steel. It is joined to the TSC shell using a field-installed groove weld. The closure lid-to-TSC shell weld is analyzed, installed, and examined in accordance with ASME Code Case N-595-4 [3]. This closure lid-to-TSC shell weld is a partial penetration weld progressively examined at the root, midplane, and final surface by liquid penetrant (PT) examination. Following NDE of the closure lid-to-TSC shell weld, the TSC cavity is reflooded and the TSC vessel is hydrostatically pressure tested as described in the Operating Procedures of Chapter 9 and the Acceptance Test Program of Chapter 10. The acceptance criteria for the test are no leakage and no loss of pressure during the minimum 10-minute test duration.

After successful completion of the hydrostatic pressure test, the Type 304 stainless steel closure ring is installed in the TSC-to-closure lid weld groove, and welded to both the closure lid and the TSC shell. The closure ring welds are inspected by PT examination of the final weld surfaces. The closure ring provides the double weld redundant sealing of the confinement boundary, as required by 10 CFR 72.236(e).

The closure lid incorporates drain and vent penetrations, which provide access to the TSC cavity for canister draining, drying and helium backfilling operations during TSC closure and preparation for placement into storage. The design of the penetrations incorporates features to provide adequate shielding for the operators during these operations and closure welding.

Following final helium backfill and pressurization, the vent and drain port penetrations are closed with Type 304 stainless steel inner port covers that are partial-penetration welded in place. Each inner port cover weld is helium leak tested. Each inner port cover weld final surface is then PT examined. A second (outer) port cover is then installed and welded to the closure lid at each of the ports to provide the double weld redundant sealing of the confinement boundary. The outer port cover weld final surfaces are inspected by PT examination.

Prior to sealing, the TSC cavity is backfilled and pressurized with helium. The minimum helium purity level of 99.995% (minimum) specified in the Operating Procedures maintains the quantity of oxidizing contaminants to less than one mole per canister for all loading conditions. Based on the maximum empty canister free volume of 10,400 liters and the design basis helium density (Section 4.4.4), an empty canister would contain approximately 2,000 moles of gases. Conservatively, assuming that all of the impurities in the helium are oxidants, a maximum of 0.1 moles of oxidants could exist in the largest canister during storage. By limiting the amount of oxidants to less than one mole, the recommended limits for preventing cladding degradation found in the PNL-6365 [4] are satisfied.

The thermal analysis of the loaded TSC is based, in part, on heat transfer from the fuel to the TSC shell by convection within the TSC. The provision of a specific density of high-purity helium, which ensures the establishment of internal convection in the TSC, also ensures that a positive pressure exists within the TSC during the design life of the system. The maintenance of a positive helium pressure eliminates any potential for in-leakage of air into the TSC cavity during storage operations.

The closure welds completed in the field are not helium leakage tested. Interim Staff Guidance (ISG)-18 [5] provides that an adequate confinement boundary is established for stainless steel spent fuel storage canisters that are closed using a closure weld that meets the guidance of ISG-15 [6]. The TSC closure weld meets the ISG-15 guidance in that the analysis of the weld considers a stress reduction factor of 0.8. The weld is qualified and performed in accordance with the ASME Code, Section IX requirements [7]; and the weld is dye penetrant examined after the root, midplane, and final surface passes. The final surfaces of the welds joining the closure ring to the closure lid and shell, and joining the redundant port covers to the closure lid are PT examined.

During fabrication, the TSC shell and bottom plate welds are volumetrically inspected and the shell assembly is shop helium leakage tested to the leaktight criteria of 1×10^{-7} ref cm^3/sec, or 2×10^{-7} cm^3/sec (helium), in accordance with ANSI N14.5 [8] using the evacuated envelope test method. A minimum test sensitivity of 1×10^{-7} cm^3/sec (helium) is required.

Based on the shop helium leakage testing of the TSC shell, bottom plate and the joining welds; the design analyses and qualifications of the closure lid and port cover welds; the performance of a TSC field hydrostatic pressure test of the closure lid-to-TSC shell weld; and the multiple NDE performed on all of the confinement boundary welds, the loaded TSC is considered and analyzed as a leaktight configuration.

The confinement boundary details at the top of the TSC are shown in Figure 7.1-1. The closure is welded by qualified welders using weld procedures qualified in accordance with ASME Code Section IX. Over its 50-year design life, the leaktight TSC precludes the release of radioactive contents to the environment and the entry of air, or water, that could potentially damage the cladding of the stored spent fuel.

7.1.2 Confinement Penetrations

Two penetrations fitted with quick-disconnect fittings are provided in the TSC closure lid for operational functions during system loading and sealing operations. The drain port accesses a drain tube that extends into a sump located in the bottom plate. The vent port extends to the underside of the closure lid and accesses the top of the TSC cavity.

After the completion of the closure lid-to-TSC shell weld, TSC pressure test, closure ring welding and cavity draining, the vent and drain penetrations are utilized for drying the TSC internals and contents, and for helium backfilling and pressurizing the TSC. After backfilling to a specific helium density, both penetrations are closed with redundant port covers welded to the closure lid. As presented for storage, the TSC has no exposed or accessible penetrations, has no mechanical closures, and does not employ seals to maintain confinement.

7.1.3 Seals and Welds

The confinement boundary welds consist of the field-installed welds that close and seal the TSC, and the shop welds that join the bottom plate to the TSC and that join the rolled plates that form the TSC shell. The TSC shell may incorporate both longitudinal and circumferential weld seams in joining the rolled plates. No elastomer or metallic seals are used in the confinement boundary of the TSC.

All cutting, machining, welding, and forming of the TSC vessel are performed in accordance with Section III, Article NB-4000 of the ASME Code, unless otherwise specified in the approved fabrication drawings and specifications. Code alternatives are listed in Table 2.1-2.

Weld procedures, welders, and welding machine operators shall be qualified in accordance with ASME Code, Section IX. Refer to Chapter 10 for the acceptance criteria for the TSC weld visual inspections and nondestructive examinations (NDE).

The loaded TSC is closed using field-installed welds. The closure lid to TSC shell weld is liquid penetrant examined at the root, at the midplane level and the final surface. After the completion of TSC hydrostatic pressure testing, the closure ring is installed and welded to the TSC shell and closure lid. The final surface of each of the closure ring welds is liquid penetrant examined. Following draining, drying, and helium backfilling operations, the vent and drain ports are closed with redundant port covers that are welded in place. The final surface of each port cover to closure lid weld is liquid penetrant examined.

Shop and field examinations of TSC confinement boundary welds are performed by personnel qualified in accordance with American Society of Nondestructive Testing Recommended Practice No. SNT-TC-1A [9]. Weld examinations are documented in written reports.

7.1.4 Closure

The closure of the TSC consists of the welded closure lid, the welded closure ring, and the welded redundant vent and drain port covers. There are no bolted closures or mechanical seals in the confinement boundary.

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a hydrogen concentration in the TSC that approaches or exceeds the Lower Flammability Limit (LFL) for hydrogen of four percent.

Thus, it is reasonable to conclude that small amounts of combustible gases, primarily hydrogen, may be produced during TSC loading or unloading operations as a result of a chemical reaction between the aluminum neutron absorber in the fuel basket and the spent fuel pool water. The generation of combustible gases stops when the water is removed from the TSC and the aluminum surfaces are dry.

8.10.3 Evaluation of the Operating Procedures

This section evaluates the operating procedures to identify the potential for galvanic reactions, corrosion or flammable gas formation to occur during planned operations. As described in this section, no potential chemical, galvanic, or other reactions have been identified for MAGNASTOR. The use of a dry inert helium atmosphere in storage inhibits galvanic and corrosion events and flammable gas formation. Monitoring for levels of hydrogen approaching the Lower Flammability Limit (LFL) is performed during TSC closure lid root pass welding, and during closure lid weld removal operations in preparation for TSC unloading. If hydrogen levels exceed 60% of LFL (i.e., 2.4%H_2), these operations are stopped and corrective actions are taken until the H_2 levels are reduced to acceptable levels. Therefore, no adverse conditions, such as the ignition of flammable or explosive quantities of combustible gases, can result during any phase of TSC operations.

8.10.3.1 Evaluation of Loading Operations

After the TSC is removed from the pool and during TSC closure operations, the cavity water level is lowered (approximately 70 gallons are removed) to ensure that the closure lid-to-TSC shell weld area is dry during welding. The lowering of the cavity water level will not expose fuel rod cladding to an air environment. As there is limited clearance between the inside diameter of the TSC shell and the outside diameter of the closure lid, it is possible that gases released from a chemical reaction inside the TSC could accumulate beneath the lid.

The aluminum surfaces of the neutron absorber panels oxidize when exposed to air, react chemically in an aqueous solution, and may react galvanically when in contact with stainless steel or carbon steel. The reaction of aluminum in water, which results in hydrogen generation, proceeds as follows.

$$2\ Al + 3\ H_20 \Rightarrow Al_2O_3 + 3\ H_2$$

The aluminum oxide (Al_2O_3) produces the dull, light gray film that is present on the surface of bare aluminum when it reacts with the oxygen in air or water. The formation of the thin oxide

film is a self-limiting reaction as the film isolates the aluminum metal from the oxygen source, acting as a barrier to further oxidation. The oxide film is stable in pH neutral (passive) solutions, but is soluble in borated PWR spent fuel pool water. The oxide film dissolves at a rate dependent upon the pH of the water, the exposure time of the aluminum in the water, and the temperatures of the aluminum and water.

PWR spent fuel pool water is a boric acid and demineralized water solution. BWR spent fuel pool water does not contain boron and typically has a neutral pH (approximately 7.0). The pH, water chemistry and water temperature vary from pool to pool. Since the reaction rate is largely dependent upon these variables, it may vary considerably from pool to pool. Thus, the generation rate of combustible gas (hydrogen) that could be considered representative of spent fuel pools in general is very difficult to accurately calculate, but the reaction rate would be less in the neutral pH BWR pool.

The MAGNASTOR basket configurations incorporate carbon steel fuel tubes and support components that are coated with electroless nickel. The coating protects the carbon steel during the comparatively short time that the TSC is immersed in, or contains, water. The coating is nonreactive with the pool water and does not off-gas or generate gases as a result of contact with the pool water. Consequently, there are no flammable gases generated by the coating and no flammable gases generated by the materials of the coated components.

To ensure safe loading of the TSC, the loading procedure described in Chapter 9 provides for the monitoring of hydrogen gas before and during the root pass welding operations that join the closure lid to the TSC shell. The monitoring system is capable of detecting hydrogen at 60% of the Lower Flammability Limit (LFL) for hydrogen (i.e., 2.4% H_2). The hydrogen detector is connected to the cavity volume so as to detect hydrogen prior to initiation of welding. The hydrogen concentration is monitored during the root pass welding operation. The welding operation is stopped upon the detection of hydrogen in a concentration exceeding 2.4%. Hydrogen gas concentrations exceeding 2.4% are removed by flushing air, nitrogen, argon or helium into the region below the closure lid or by evacuating the hydrogen using a vacuum pump.

The vacuum pump exhausts to a system or area where hydrogen flammability is not an issue. Once the root pass weld is completed, there is no further likelihood of a combustible gas burn because the ignition source is isolated from the potential source of combustible gases, and hydrogen gas monitoring is stopped.

Hydrogen is not expected to be detected prior to, or during, the welding operations. During the completion of the closure lid to TSC shell root pass, the hydrogen gas detector accesses the vent

port and is used to monitor the hydrogen gas levels. Following closure lid welding and TSC hydostatic testing, the TSC is drained. Once the TSC is dry, no combustible gases form within the TSC.

8.10.3.2 Evaluation of Unloading Operations

The TSC is dried and backfilled with helium immediately prior to final closure welding operations, thereby eliminating all oxidizing gases and water. Therefore, it is not expected that the TSC will contain any combustible gases during the time period of storage. To ensure the safe, wet unloading of the TSC, the unloading procedure described in Chapter 9 provides for monitoring for hydrogen gas during closure lid weld cutting/removal operations.

The principal steps in opening the TSC are the removal of the vent and drain port cover welds, and the removal of the closure lid weld. The welds are expected to be removed by cutting or grinding. Following removal of the vent and drain port covers, the TSC is sampled for radioactive gases, vented, flushed with nitrogen gas, and cooled down with water using the vent and drain ports. Prior to cutting the closure lid weld, the cavity water level is lowered to permit removal of the closure lid weld in a dry environment, and the cavity gas volume is sampled for hydrogen gas levels ≥2.4% using a hydrogen gas detector connected to the vent port. If unacceptable hydrogen levels are detected during closure lid weld removal operations, weld removal operations are terminated and the cavity is flushed with air, nitrogen, argon or helium, or the cavity is evacuated with a vacuum pump.

8.10.3.3 Conclusions

The steps taken to monitor for the presence of hydrogen will ensure that combustion of any hydrogen gas does not occur due to either closure lid welding or lid removal operations. Based on this evaluation, which results in no identified reactions, it is concluded that MAGNASTOR operating controls and procedures for loading and unloading the TSC presented in Chapter 9 are adequate to minimize the occurrence of hazardous conditions.

8.11 Cladding Integrity

Intact fuel is spent nuclear fuel that is not damaged fuel, as defined below. To be classified as intact, fuel must meet the criteria for both intact cladding and structural integrity. An intact fuel assembly can be handled using normal handling methods, and any missing fuel rods are replaced by solid filler rods that displace a volume equal to, or greater than, that of the original fuel rod.

Damaged fuel is spent nuclear fuel that includes any of the following conditions that result in either compromise of cladding confinement integrity or reconfiguration of fuel assembly geometry:

1) The fuel contains known or suspected cladding defects greater than a pinhole leak or a hairline crack that have the potential for release of significant amounts of fuel particles.
2) The fuel assembly:
   a) is damaged in such a manner as to impair its structural integrity;
   b) has missing or displaced structural components such as grid spacers;
   c) is missing fuel pins that have not been replaced by filler rods that displace a volume equal to, or greater than, that of the original fuel rod;
   d) cannot be handled using normal handling methods.
3) The fuel is no longer in the form of an intact fuel assembly and consists of, or contains, debris such as loose pellets, rod segments, etc.

Fuel cladding integrity is maintained in storage by limiting the fuel cladding temperature to less than 400°C for normal and transfer conditions and to less than 570°C for off-normal and accident events. The analyses providing the maximum calculated fuel cladding temperatures for normal conditions and off-normal or accident events are provided in Chapter 4. The thermal analyses presented in Chapter 4 also provide the calculated temperature history for loading operations. The operating procedures and specific operational completion times have been determined and defined to preclude excessive temperature cycles of the fuel cladding exceeding 65°C and cladding temperatures in excess of 400°C. These temperature limits preclude thermally induced fuel rod cladding deterioration by limiting fuel rod cladding hoop stress and reducing the potential for the reorientation of hydrides. Thus, the stored fuel and its cladding integrity are maintained intact and may be retrieved, using normal means of handling.

Normal and accident condition thermal transients experienced by the MAGNASTOR canister, basket and contained fuel are controlled and introduce insignificant thermal loading and material stress to fuel rod cladding. Normal condition cooldown transients during cask operations may be

introduced during vacuum drying when the canister dryness criteria are not met within the prescribed heat load-dependent time limit. If the dryness criteria are not met, the canister is backfilled to 7 bar gauge with helium, and the canister is cooled by the annulus cooling water system or by returning the canister to the pool as stated in Section 1.3.1.4. This backfill with helium may be performed when the temperatures in the mid to upper regions of the fuel basket are in the range of 700°F and the fuel local to the bottom plate is in the range of 250°F. Noting the significant difference in mass between helium and fuel, i.e., approximately five orders of magnitude, helium is heated with little temperature change to the fuel – the basket, canister bottom plate and shell mass add heat to the helium in combination with the fuel – reducing the thermal influence of the initial helium fill on the fuel cladding. Following the helium backfill, the canister is cooled by the annulus cooling water system or returned to the pool. Water in contact with the canister wall provides more effective heat transfer than the air boundary when the transfer cask is sitting in a cask processing area outside the pool. Although this pool water boundary provides a more effective heat transfer path, the influence of the canister submergence in the pool does not produce a thermal shock or significant through wall gradient to the fuel rod cladding.

Investigation of the canister unloading sequence presented in Section 9.3 leads to similar conclusions as those for the introduction of helium gas discussed above. When the canister is first prepared for unloading and the port covers are removed, nitrogen gas is initially cycled through the canister for a minimum of 10 minutes to flush the radioactive gases from the canister. This gas cycling is similar to the helium backfill. Although nitrogen has a higher thermal capacitance than helium (about a factor of 10), when compared to the mass of the metal canister, basket and fuel, the influence of the nitrogen gas on the thermal gradient response in the fuel cladding remains insignificant. Following the nitrogen flush, water is introduced into the canister at a maximum rate of 8 gpm. The maximum flow rate is based on reflood thermal hydraulic analyses of a bounding canister configuration. The bounding maximum flow rate, water temperature and pressure are defined in step 14 of Section 9.3, "Wet Unloading a TSC." The water initially introduced into the canister flashes to steam in the drain tube and on contact with the bottom plate. Steam in the cavity permits additional heat to be removed from the basket and fuel in a smooth transition without introducing thermal shock through wall stresses. Once water is permitted to form on the canister bottom plate, the canister starts to fill at a maximum rate of 8 gpm. Addition of water at 8 gpm permits the water to rise in the canister at a maximum rate of 0.8 inch per minute. Thermal hydraulic analyses results show thermal cladding temperature radial gradients are less than 1°F during the reflooding of the canister. Such a small increase is consistent with the gradual cooling process created by the initial steam condition followed by water. The axial temperature gradient along the fuel assembly is actually larger than

the radial gradient. However, in the fuel axial direction, thermal stresses are not developed since the fuel cladding is free to expand in the axial direction. The combination of initial nitrogen purge, followed by the cooling transition of the steam created in the canister cavity, provides a relatively smooth transition to water cooling and insignificant thermal stress in the fuel rod cladding.

There are no evaluated normal conditions or off-normal events that result in damage to the fuel cladding or the TSC that preclude retrieval of the TSC for transport and ultimate disposal.

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33. Decontaminate the top of the transfer cask and TSC closure lid to allow installation of the welding equipment. Decontaminate external surfaces of the transfer cask and remove the bottom protective cover, if installed.
34. Insert the drain line with a female quick-connector attached through the drain port opening and into the basket drain port sleeve. Remove the female quick-disconnect and any contaminated water displaced from the cavity.
35. Torque the drain tube connector to the drain opening to the value specified in Table 9.1-2. Verify quick-disconnect is installed and properly torqued in the vent port opening.
36. Install a venting device to the vent port quick-disconnect to prevent combustible gas or pressure buildup below the closure lid.
37. At the discretion of the user, establish foreign material exclusion controls to prevent objects from being dropped into the annulus or TSC.
38. Install the welding system, including supplemental shielding, to the top of the closure lid.

    Note: At the discretion of the user, supplemental shielding may be installed around the transfer cask to reduce operator dose. Use of supplemental shielding shall be evaluated to ensure its use does not adversely affect the safety performance of MAGNASTOR.

39. Connect a suction pump to the drain port quick-disconnect and verify venting through the vent port quick-disconnect.
40. Operate the suction pump to remove approximately 70 gallons of water from the TSC. Disconnect the suction pump.

    Note: The radiation level will increase as water is removed from the TSC cavity, since shielding is being removed.

    Note: Fuel rods shall not be exposed to air during the 70-gallon pumpdown.

41. Attach a hydrogen detector to the vent line. Ensure that the vent line does not interfere with the operation of the weld machine.
42. Sample the gas volume below the closure lid and observe hydrogen detector for H_2 concentration prior to commencing closure lid welding operations. Monitor H_2 concentration in the TSC until the root pass of the closure lid-to-shell weld is completed.

    Note: If H_2 concentration exceeds 2.4% prior to or during root pass welding operations, immediately stop welding operations. Evacuate the TSC gas volume by connecting a vacuum pump to the vent port and evacuating the gas volume, or purge the gas volume with helium, nitrogen or argon gas. Verify H_2 levels are <2.4% prior to restarting welding operations.

43. Install shims into the closure lid-to-TSC shell gap, as necessary, to establish a uniform gap for welding. Tack weld the closure lid, as required.

44. Operate the welding equipment to complete the closure lid-to-TSC shell root pass weld in accordance with the approved weld procedure.
45. Remove the H_2 detector from the vent line while ensuring the vent line remains installed.
46. Perform visual and liquid penetrant (PT) examinations of the root pass and record the results.
47. Operate the welding equipment to perform the closure lid-to-shell weld to the midplane between the root and final weld surfaces. Perform visual and PT examinations for the midplane weld pass, and record the results.
48. Complete the final pass of the closure lid weld, perform final visual and PT examinations and record the results.
49. Perform the hydrostatic testing of the TSC as follows:
    a. Connect a drain line to the vent port and the pressure test system to the drain port.
    b. Refill the TSC with clean water until water is observed flowing from the vent port drain line. Close the vent line isolation valve. Ensure compliance with the boron concentration requirements of LCO 3.2.1.
    c. Pressurize the TSC to 130 (+5, -0) psig and isolate the TSC.
    d. Monitor the TSC pressure for a minimum of 10 minutes and visually examine the closure lid-to-TSC shell weld for leakage of water.
    e. The hydrostatic test is acceptable if there is no observed pressure drop or visible water leakage from the closure lid weld during the test.
    f. Remove the pressure test system from the drain port and the drain line from the vent line. Reinstall the vent line to the vent port to prevent pressurization of the TSC.
50. Install and tack closure ring in position in the closure lid-to-TSC shell weld groove.
51. Weld the closure ring to the TSC shell and to the closure lid. Perform visual and PT examinations of the final surfaces of the welds and record the results.
52. Remove the water from the TSC using one of the following methods: drain down using a suction pump with a helium or nitrogen pressurized cover gas; or blow down using pressurized helium or nitrogen gas. Ensure the totalizer in the drain line is reset to zero.

    Note: If pressurized helium drying is to be used to remove residual moisture, helium must be used to blow down the TSC. This will preclude the presence of any significant amount of residual oxidizing gases in the cavity after completion of helium drying and backfill operations.

    Note: Fuel rods shall not be exposed to air during canister draining operations.
53. Connect a drain line with or without suction pump to the drain port connector.
54. Connect a regulated nitrogen or helium gas supply to the vent port guide connector. Set gas regulator to 30 (+10, -25) psig.

55. Open gas supply valve and start suction pump, if used, and drain water from the TSC until water ceases to flow out of the drain line. Close gas supply valve and stop suction pump.
56. Record the time at the completion of the draining of the TSC. Record the volume of water drained from the TSC (V_{TSC}) as measured by the totalizer.
57. At the option of the user, disconnect suction pump, close discharge line isolation valve, and open gas supply line. Pressurize TSC to 25 (+5, -10) psig and open discharge line isolation valve to blow down the TSC. Repeat blow down operations until no significant water flows out of the drain line.
58. Disconnect the drain line and gas supply line from the drain and vent port quick-disconnects.
59. Dry the TSC cavity using one of the methods described in Step 60 and Step 61 (vacuum drying) or in Step 62 and Step 63 (pressurized helium drying). Ensure time durations established for vacuum drying are not exceeded so that fuel cladding temperatures are maintained below 752°F.

    Note: At the option of the user, the drain and/or vent port quick-disconnects can be removed and replaced temporarily with suitable straight-through fittings to increase flow area cross-section and to reduce resistance to gas flow. The quick-disconnect fittings must be reinstalled and torqued prior to final helium backfill.
60. Vacuum dry the TSC using the vacuum drying system as follows.
    a. Connect the vacuum drying system to the vent and drain port openings.
    b. Operate the vacuum pump until a vapor pressure of < 10 torr is achieved in the TSC. The time durations of the first vacuum drying phase shall be in accordance with the time limits of Table 9.1-3.
    c. Isolate the vacuum pump from the TSC and turn off the vacuum pump. Observe the vacuum gauge connected to the TSC for an increase in pressure for a minimum period of 10 minutes. If the TSC pressure is ≤ 10 torr at the end of 10 minutes, the TSC is dry of free water in accordance with LCO 3.1.1.

    Note: If vacuum drying times greater than those defined in Table 9.1-3 are required to dry the TSC, the TSC shall be backfilled with helium to 7 bar, gauge, and cooled by the annulus circulating water system, or by placement in the spent fuel pool for a 12-hour (+1,-0) period. After the cooling period, drying operations can continue for the times indicated in Table 9.1-4. Drying cycles may be continued until the TSC cavity passes the dryness verification per LCO 3.1.1, Condition A.

    Note: If the annulus cooling system becomes inoperable during the vacuum drying operational sequence, backfill the TSC to 7 bar, absolute, and place the TSC

into auxiliary cooling (i.e., placement of the transfer cask with the TSC into the spent fuel pool, or start the auxiliary air cooling system) and maintain the auxiliary cooling until the annulus cooling system is operable.

61. Upon satisfactory completion of the dryness verification, evacuate the TSC cavity to a pressure of ≤ 3 torr. Isolate the vacuum pump, and backfill and pressurize the TSC cavity with 99.995% (minimum) pure helium as follows:
    a. Determine the free volume of the TSC (V_{TSC}) per Step 56.
    b. Multiply the V_{TSC} free volume by the helium loading value per unit volume (L_{helium}) to determine required helium mass (M_{helium}) to be backfilled into the cavity.
    c. Set the helium bottle regulator to 100 (+5,-0) psig.
    d. Connect the helium backfill system to the vent port and reset the mass-flow meter to zero
    e. Slowly open the helium supply valve and backfill the TSC with the required helium mass (M_{helium}) in accordance with LCO 3.1.1.
62. Dry the TSC using the pressurized helium drying system as follows:
    a. Connect the inlet helium gas line for the pressurized helium drying system to the drain port.
    b. Connect the helium discharge line to the vent port.
    c. Using a helium supply system, charge the drying system with 99.995% (minimum) pure helium to a pressure of 70 (±5) psig.
    d. Start the gas circulation pump, or equivalent, and initiate dry helium gas circulation.
    e. Continue helium circulation through the TSC until the measured discharged gas dew point temperature at the vent port meets the dryness criteria of LCO 3.1.1.
63. Stop helium circulation and backfill the TSC with 99.995% (minimum) pure helium per LCO 3.1.1 until the cavity is backfilled with the required helium mass (M_{helium}).
64. Disconnect the vacuum drying system and the helium backfill system, or the pressurized helium drying system, from the vent and drain openings.
65. Install and weld the inner port cover on the drain port opening.
66. Install and weld the inner port cover on the vent port opening.
67. Perform helium leak test on each of the inner port cover welds.
68. Perform visual and PT examinations of the final surface of the port cover welds and record the results.
69. Install and weld the outer port cover on the drain port opening. Perform visual and PT examinations of the final weld surface and record the results.

70. Install and weld the outer port cover on the vent port opening. Perform visual and PT examinations of the final weld surface and record the results.
71. Using an appropriate crane, remove the weld machine and supplemental shield.
72. Drain the TSC/transfer cask annulus by stopping water flow to the annulus and connecting one or more drain lines to the lower annulus fill ports. Once the annulus is drained, deflate the top and bottom annulus seals. Note the time the annulus circulating system flow is terminated.

    Note: The time duration of the sequence of operations from stopping annulus circulating water systems through completion of TSC transfer into the concrete cask shall not exceed 19 hours. If the TSC transfer to the vertical concrete cask cannot be completed in the defined time period, the transfer operation will be suspended and the TSC shall be cooled by the annulus circulating water system for a minimum of 12 hours prior to restarting TSC transfer operations. The second, and subsequent, TSC transfer evolution times are limited to 12 hours prior to returning the TSC into cooling.
73. Remove the lock pins and move the transfer cask retaining blocks inward into their functional position. Reinstall the lock pins.
74. Install the six swivel hoist rings into the six threaded holes in the closure lid if TSC transfer is to be performed by two sets of redundant slings. Torque the hoist rings to the manufacturer's recommended value.

    Note: Alternative site-specific TSC lifting systems and equipment may be used for lowering and lifting the TSC in the transfer cask. The lifting system design must comply with the user's heavy load program and the applicable requirements of ANSI N14.6, NUREG-0612, and/or ASME/ANSI B30.1, as appropriate.
75. Complete final decontamination of the transfer cask exterior surfaces. Final TSC contamination surveys may be performed after TSC transfer following Step 21 in Section 9.1.2 when TSC surfaces are more accessible.
76. Proceed to Section 9.1.2.

9.1.2 Transferring the TSC to the Concrete Cask

This section describes the sequence of operations required to complete the transfer of a loaded TSC from the transfer cask into a concrete cask, and preparation of the concrete cask for movement to the ISFSI pad.

1. Position an empty concrete cask with the lid assembly removed in the designated TSC transfer location.

   Note: The concrete cask can be positioned on the ground, or on a deenergized air pad set, roller skid, heavy-haul trailer, rail car, or transfer cart. The transfer location can be in a truck/rail bay inside the loading facility or an external area accessed by the facility cask handling crane.
2. Inspect all concrete cask openings for foreign objects and remove if present; install supplemental shielding in four outlets.
3. Install a four-legged sling set to the lifting points on the transfer adapter.
4. Using the crane, lift the transfer adapter and place it on top of the concrete cask ensuring that the guide ring sits inside the concrete cask lid flange. Remove the sling set from the crane and move the slings out of the operational area.
5. Connect a hydraulic supply system to the hydraulic cylinders of the transfer adapter.
6. Verify the movement of the connectors and move the connector tees to the fully extended position.
7. Connect the lift yoke to the crane and engage the lift yoke to the transfer cask trunnions. Ensure all lines, temporary shielding and work platforms are removed to allow for the vertical lift of the transfer cask.

   Note: The minimum ambient air temperature (either in the facility or external air temperature, as applicable for the handling sequence) must be ≥ 0°F for the use of the transfer cask.
8. Raise the transfer cask and move it into position over the empty concrete cask.
9. Slowly lower the transfer cask into the engagement position on top of the transfer adapter to align with the door rails and engage the connector tees.
10. Following set down, remove the lock pins from the shield door lock tabs.
11. Install a stabilization system for the transfer cask, if required by the facility heavy load handling or seismic analysis programs.
12. Disengage the lift yoke from the transfer cask trunnions and move the lift yoke from the area.
13. As appropriate to the TSC lifting system being used, move the lifting system to a position above the transfer cask. If redundant sling sets are being used, connect the sling sets to the crane hook.
14. Using the TSC lifting system, lift the TSC slightly (approximately ½-1 inch) to remove the TSC weight from the shield doors.

    Note: The lifting system operator must take care to ensure that the TSC is not lifted such that the retaining blocks are engaged by the top of the TSC.

15. Open the transfer cask shield doors with the hydraulic system to provide access to the concrete cask cavity.
16. Using the cask handling crane in slow speed (or other approved site-specific handling system), slowly lower the TSC into the concrete cask cavity until the TSC is seated on the pedestal.

    Note: The transfer adapter and the standoffs in the concrete cask will ensure the TSC is appropriately centered on the pedestal within the concrete cask.

    Note: The completion of the transfer of the TSC to the vertical concrete cask (i.e., set-down of the TSC on the vertical concrete cask pedestal) completes the TSC transfer evolution time limits from Step 70 in Section 9.1.1.
17. When the TSC is seated, disconnect the slings (or other handling system) from the lifting system, and lower the sling sets through the transfer cask until they rest on top of the TSC.
18. Retrieve the lift yoke and engage the lift yoke to the transfer cask trunnions.
19. Remove the seismic/heavy load restraints from the transfer cask, if installed.
20. Close the shield doors using the hydraulic system and reinstall the lock pins into the shield door lock tabs.
21. Lift the transfer cask from the top of the concrete cask and return it to the cask preparation area for next fuel loading sequence or to its designated storage location.
22. Disconnect hydraulic supply system from the transfer adapter hydraulic cylinders.
23. Remove redundant sling sets, swivel hoist rings, or other lifting system components from the top of the TSC, if installed.
24. Verify all equipment and tools have been removed from the top of the TSC and transfer adapter.
25. Connect the transfer adapter four-legged sling set to the crane hook and lift the transfer adapter off the concrete cask. Place the transfer adapter in its designated storage location and remove the slings from the crane hook. Remove supplemental shielding from outlets.

    Note: If the optional low profile concrete cask is used, proceed to Step 26. If the standard concrete cask is provided, proceed to Step 38.
26. Install three swivel hoist rings and the three-legged sling set on the concrete cask shield ring.
27. Using the crane, lift the shield ring and place it into position inside of the concrete cask top flange.
28. Remove the three-legged sling and swivel hoist rings.
29. Using the designated transport equipment, move the loaded concrete cask out of the low clearance work area or truck/rail bay.
30. Install the three swivel hoist rings into the three threaded holes and attach the three-legged sling set to the shield ring.

31. Using an external or mobile crane, lift and remove the shield ring. Place the shield ring in position for the next loading sequence or return it to its designated storage location.
32. Install four swivel hoist rings in the threaded holes of the concrete cask extension using the manufacturer-specified torque.
33. Install the four-legged sling set and attach to the crane hook.

    Note: A mobile crane of sufficient capacity may be required for concrete cask extension and lid installations performed outside the building.
34. Perform visual inspection of the top of the concrete cask and verify all equipment and tools have been removed.

    Note: Take care to minimize personnel access to the top of the unshielded loaded concrete cask due to shine from the TSC.
35. Lift the concrete cask extension and move it into position over the concrete cask, ensuring alignment of the two anchor cavities with their mating lift anchor embedment.
36. Lower the concrete cask extension into position and remove the sling set from the crane hook.
37. Remove the four swivel hoist rings and cables from the concrete cask extension.

    Note: If concrete cask transport is to be performed by a vertical cask transporter, proceed to Step 38. If transport is to be performed using air pads in conjunction with a flat-bed transporter, proceed to Step 40.
38. Install the lift lugs into the anchor cavities of the concrete cask extension, or directly on top of the lifting embedment for the standard concrete cask.
39. Install the lift lug bolts through each lift lug and into the threaded holes in the embedment base. Torque each of the lug bolts to the value specified in Table 9.1-2.
40. Install three swivel hoist rings into the concrete cask lid and attach the three-legged sling set. Attach the lifting sling set to the crane hook.
41. Install the weather seal on the concrete cask lid flange. Lift the concrete cask lid and place it in position on the top of the flange.
42. Remove the sling set and swivel hoist rings and install the concrete cask lid bolts. Torque to the value specified in Table 9.1-2.
43. Move the loaded concrete cask into position for access to the site-specific transport equipment.
44. Proceed to Section 9.1.3.

9.1.3 Transporting and Placing the Loaded Concrete Cask

The section describes the general procedures for moving a loaded concrete cask to the ISFSI pad using either a vertical cask transporter (Step 1 through Step 9) or a flat-bed transport vehicle (Steps 10 through 17). Steps following Step 17 are performed for all concrete casks.

Vertical Cask Transporter

1. Using the vertical cask transporter lift fixture or device, engage the two concrete cask lifting lugs.
2. Lift the loaded concrete cask and move it to the ISFSI pad following the approved onsite transport route.

   Note: Ensure vertical cask transporter lifts the concrete cask evenly using the two lifting lugs.

   Note: Do not exceed the maximum lift height for a loaded concrete cask of 24 inches.
3. Move the concrete cask into position over its intended ISFSI pad storage location. Ensure the surface under the concrete cask is free of foreign objects and debris.

   Note: The spacing between adjacent loaded concrete casks must be at least 15 feet.
4. Using the vertical transporter, slowly lower the concrete cask into position.
5. Disengage the vertical transporter lift connections from the two concrete cask lifting lugs. Move the cask transporter from the area.
6. Detorque and remove the lift lug bolts from each lifting lug, if the lugs are to be reused.

   Note: At the option of the user, the lift lugs may be left installed during storage operations.
7. Lift out and remove the concrete cask lift lugs. Store the lift lugs for the next concrete cask movement.
8. Install the lug bolts through the extension base (or through the cover plate for the standard concrete cask) and into the threaded holes. Torque each bolt to the value specified in Table 9.1-2.
9. For the casks with extensions containing anchor cavities, install the weather seal and cover plates. Install the bolts and washers and torque to the value specified in Table 9.1-2.

Flat-bed Transport Vehicle Loaded with the Closed Concrete Cask

10. Move the transport vehicle with the closed concrete cask to a position adjacent to the ISFSI pad.
11. If required, install a bridging plate to cover the gap between the vehicle and the ISFSI pad.
12. If not already installed, insert four deflated air pads into the four inlets.

13. Attach a restraining device around the concrete cask and connect to a tow vehicle suitable for pushing or pulling the concrete cask off of the transport vehicle.
14. Using an air supply and an air pad controller, inflate the air pads.
15. Verify the ISFSI pad surface in the storage location is free of foreign objects and debris.
16. Using the tow vehicle, move the concrete cask into its position on the storage pad.

    Note: The center-to-center spacing of loaded concrete casks shall be a minimum of 15 feet.

17. Lower the concrete cask into position by deflating and removing the four air pads.

All Concrete Casks

18. If optional temperature monitoring is implemented, install the temperature monitoring devices in each of the four outlets of the concrete cask and connect to the site's temperature monitoring system.
19. Install inlet and outlet screens to prevent access by debris and small animals.

    Note: Screens may be installed on the concrete cask prior to TSC loading to minimize operations personnel exposure.

20. Scribe and/or stamp the concrete cask nameplate to indicate the loading date. If not already done, scribe or stamp any other required information.
21. Perform a radiological survey of the concrete cask within the ISFSI array to confirm dose rates comply with ISFSI administrative boundary and site boundary dose limits.
22. Initiate a daily temperature monitoring program or daily inspection program of the inlet and outlet screens to verify continuing effectiveness of the heat removal system.

Chapter 10

Specification section, which follows. The presence, areal density (effectiveness) and uniform distribution of ^{10}B in the borated MMC material will be verified by neutron transmission testing. Visual inspections of the sheets of borated MMC material will be based on Aluminum Association recommendations, i.e., blisters and/or widespread rough surface conditions such as die chatter or porosity will not be acceptable, but local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

10.1.6.4.3 Borated Aluminum

Borated aluminum material is a direct chill cast metallurgy product with a uniform fine dispersion of discrete boron particles in a matrix of aluminum. Borated aluminum material is a metallurgically bonded matrix, low porosity product. Borated aluminum is credited with an effectiveness of 90% of the specified minimum areal density of ^{10}B in the borated aluminum material based on acceptance testing of the material as described in the Specification section, which follows. The presence, areal density and uniform distribution of ^{10}B in the borated aluminum material will be verified by neutron transmission testing. Visual inspections of the sheets of borated aluminum material will be based on Aluminum Association recommendations, i.e., blisters and/or widespread rough surface conditions such as die chatter or porosity will not be acceptable, but local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

10.1.6.4.4 Thermal Conductivity Testing of Neutron Absorber Material

Thermal conductivity qualification testing of the neutron absorber materials shall conform to ASTM E1225 [15], ASTM E1461 [16], or an equivalent method. The testing shall be performed at room temperature on test coupons taken from production material. Note that thermal conductivity increases slightly with temperature increases.

- Sampling will initially be one test per lot and may be reduced if the first five tests meet the specified minimum thermal conductivity. Additional tests may be performed on the material from a lot whose test result does not meet the required minimum value, but the lot will be rejected if the mean value of the tests does not meet the required minimum value.
- Upon completion of 25 tests of a single type of neutron absorber material having the same aluminum alloy matrix and boron content (in the same compound), further testing may be terminated if the mean value of all of the test results minus two standard deviations meets the specified minimum thermal conductivity. Similarly, testing may be terminated if the matrix of the material changes to an alloy with a larger coefficient of

thermal conductivity, or if the boron compound remains the same, but the boron content is reduced.

In the Chapter 4 thermal analyses, the neutron absorber is conservatively evaluated as a 0.125-in nominal thickness sheet (0.1-in thick boron composite core with 0.0125-in thick aluminum face plates - Boral) for the PWR fuel basket and a 0.10-in nominal thickness sheet (0.075-in thick boron composite core with 0.0125-in thick aluminum face plates - Boral) for the BWR fuel basket. The required minimum thermal conductivities for the MAGNASTOR absorbers are as follows.

Minimum Effective Thermal Conductivity - BTU/(hr-in-°F)

Fuel Basket Type	Radial		Axial	
	100°F	500°F	100°F	500°F
PWR	4.565	4.191	4.870	4.754
BWR	4.687	4.335	5.054	5.017

Neutron absorber sheets of borated MMC material or borated aluminum will have higher effective coefficients of thermal conductivity than the Boral sheets evaluated due to their larger aluminum alloy content. The neutron absorber thermal acceptance criterion will be based on the nominal sheet thickness.

Additional thermal conductivity qualification testing of neutron absorber material is not required if certified quality-controlled test results (from an NAC approved supplier) that meet the specified minimum thermal conductivity are available as referenced documentation.

10.1.6.4.5 Acceptance Testing of Neutron Absorber Material by Neutron Transmission

Acceptance testing shall be performed to ensure that neutron absorber material properties for sheets in a given production run are in compliance with the materials requirements for the MAGNASTOR fuel baskets and that the process is operating in a satisfactory manner.

Proposed alternatives to neutron absorber material acceptance testing may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate the following.

- The proposed alternatives would provide an acceptable level of quality and safety, or compliance with the specified requirements of Section 10.1.6.4.5 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.
- Requests for alternatives shall be submitted in accordance with 10 CFR 72.

Statistical tests may be run to augment findings relating to isotopic content, impurity content, or uniformity of the ^{10}B distribution.

- For neutron absorber materials credited with 90% effectiveness, determination of neutron absorber material acceptance shall be performed by neutron transmission testing of a statistical sample of finished product or test coupons taken from each lot of material to verify the presence, uniform distribution and the minimum areal density of ^{10}B. The definition of lot and associated sampling and testing terminology is provided in this section.
- For neutron absorber materials credited with 75% effectiveness, determination of neutron absorber material acceptance may be performed by either neutron transmission or wet chemistry testing of a statistical sample of finished product or test coupons taken from each lot of material to verify the adequacy of ^{10}B content. The wet chemistry method shall demonstrate the repeatability and correlation of the B_4C content relative to neutron transmission testing. The definition of lot and associated sampling and testing terminology is as provided previously in this section.
- Based on the MAGNASTOR required minimum effective areal density of ^{10}B – 0.036 g/cm^2 for the PWR basket and 0.027 g/cm^2 for the BWR basket – and the credit taken for the ^{10}B for the criticality analyses, i.e., 75% for Boral and 90% for borated aluminum alloys and for borated metal matrix composites, a required minimum areal density for the as-manufactured neutron absorber sheets is established.
- Test locations/coupons shall be well distributed throughout the lot of material, particularly in the areas most likely to contain variances in thickness, and shall not contain unacceptable defects that could inhibit accurate physical and test measurements.
- The neutron absorber sampling plan shall be selected to demonstrate a 95/95 statistical confidence level in the neutron absorber material and shall be implemented in accordance with written and approved procedures. The sampling plan shall require that each of the first 50 sheets of neutron absorber material from a lot, or a coupon taken therefrom, be tested. Thereafter, coupons shall be taken from 10 randomly selected sheets from each set of 50 sheets. This 1 in 5 sampling plan shall continue until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion), or a process change. A measured value less than the required minimum areal density of ^{10}B during the reduced inspection results in a rejection, along with rejection of other contiguous sheets, and mandates a return to 100% inspection for the next 50 sheets. The coupons are indelibly marked and recorded for identification. This identification will be used to document the neutron absorber material test results, which become part of the quality record documentation package.

- Neutron transmission testing of the final product or the coupons shall compare the results with those for calibrated standards composed of a homogeneous ^{10}B compound. Other calibrated standards may be used, but those standards must be shown to be equivalent to a homogeneous standard.
- An NAC approved facility with a neutron source and neutron transmission detection capability shall be selected to perform the described tests. The tests will ensure that the neutron absorption capacity of the material tested is equal to, or higher than, the reference calibration standard value and will verify the uniformity of boron distribution.
- For each neutron absorber sheet, the acceptance criterion is established from a statistical analysis of the test results for the lot of which it was a part. The minimum ^{10}B areal density is determined by reducing the nominal measured areal density by 3 standard deviations, based on the number of neutrons counted, to account for statistical variations in testing. This minimum ^{10}B areal density is converted to volume density by dividing by the neutron absorber thickness at the test location. Then, the lower tolerance limit of ^{10}B volume density – defined as the mean value ^{10}B volume density for the measurements, less K times the standard deviation, where K is the one-sided tolerance limit factor for a normal distribution with 95% probability and 95% confidence – is determined. The minimum neutron absorber sheet thickness is calculated as the minimum specified value of ^{10}B areal density divided by the lower tolerance limit of ^{10}B volume density. Then, the lower tolerance limit of ^{10}B volume density and the minimum sheet thickness are compared to the specified minimum values to determine the acceptability of each lot of neutron absorber material.
- Neutron absorber sheets thinner than the larger of the calculated minimum sheet thickness or the drawing defined minimum sheet thickness, shall be considered non-conforming, but local depressions totaling no more than 0.5% of the area of the sheet are acceptable if the thickness at their location is ≥ 90% of the drawing specified minimum sheet thickness. Any lot of material not meeting the neutron transmission testing acceptance criteria shall be rejected, and no rejected neutron absorber sheet shall be used.
- All neutron absorber material acceptance verification will be conducted in accordance with the NAC International Quality Assurance Program. The neutron absorber material supplier shall control manufacturing in accordance with the key process controls via a documented quality assurance system (approved by NAC), and the designer shall verify conformance by reviewing the manufacturing records.

10.1.6.4.6 Qualification Testing of Neutron Absorber Material

Qualification tests for each MAGNASTOR System neutron absorber material and its set of manufacturing processes shall be performed at least once to demonstrate acceptability and durability based on the critical design characteristics, previously defined in this section.

Proposed alternatives to neutron absorber material qualification testing may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate the following.

- The proposed alternatives would provide an acceptable level of quality and safety, or compliance with the specified requirements of Section 10.1.6.4.6 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.
- Requests for alternatives shall be submitted in accordance with 10 CFR 72.

The licensed service life will include a range of environmental conditions associated with short-term transfer operations, normal storage conditions, as well as off-normal and accident storage events. Additional qualification testing is not required for a neutron absorber material previously qualified, i.e., reference can be provided to prior testing with the same, or similar, materials for similar design functions and service conditions.

- Qualification testing is required for: (1) neutron absorber material specifications not previously qualified; (2) neutron absorber material specifications previously qualified, but manufactured by a new supplier; and (3) neutron absorber material specifications previously qualified, but with changes in key process controls. Key process controls for producing the neutron absorber material used for qualification testing shall be the same as those to be used for commercial production.
- Qualification testing shall demonstrate consistency between lots (2 minimum).
- Environmental conditions qualification will be verified by direct testing or by validation by data on the same, or similar, material, i.e., the neutron absorber material is shown to not undergo physical changes that would preclude the performance of its design functions. Conditions encountered by the neutron absorber material may include: short-term immersion in water, exposure to chemical, temperature, pressure, and gamma and neutron radiation environments. Suppliers' testing has shown the durability of the three types of neutron absorber materials that may be used in the MAGNASTOR system by demonstrating that the neutron absorber materials will not incur significant damage due to the pressure, temperature, radiation, or corrosion environments or the short-term water immersion that may occur in the loading and storage of spent fuel.

- Mechanical testing of the neutron absorber materials is not required, since the only related design requirement is that the material have a strength at least equivalent to that of 1100 series aluminum at 700°F that is sufficient to maintain its form. Verification will be by review of supplier-provided mechanical properties. Thermal conductivity qualification testing shall be as previously described in this section.
- The uniformity of the boron carbide distribution in the material shall be verified by neutron transmission testing of a statistically significant number of measurements of the areal density at locations distributed throughout the test material production run, i.e., from the ends and the middle. The designer shall define the allowable difference between the measured maximum and minimum ^{10}B areal densities.
- Testing to determine the minimum ^{10}B areal density shall be performed for the test material production run at a minimum of 25 distributed locations on each sheet of material. One standard deviation of the sampling shall be less that 5% of the sample mean. The ^{10}B areal density testing may be by the neutron transmission method similar to that described previously in this section for Acceptance Testing or by chemical analysis.
- A material qualification report verifying that all design requirements are satisfied shall be prepared.
- Key manufacturing process controls in the form of a complete specification for materials and process controls shall be developed for the neutron absorber material by the supplier and approved by NAC to ensure that the product delivered for use is consistent with the qualified material in all respects that are important to the material's design function.
- Major changes in key manufacturing processes for neutron absorber material shall require a complete program of qualification testing prior to the use of the material produced by the changed process. Neutron absorber material process changes defined as major changes include those that: (1) reduce the neutron absorber material thermal conductivity; (2) increase the material porosity; (3) reduce the material strength; (4) increase the boron carbide content of the material; (5) change the matrix alloy; or (6) adversely affect the uniform distribution of the ^{10}B in the material.
- Minor neutron absorber material processing changes may be determined to be acceptable on the basis of engineering review without additional qualification testing, if such changes do not adversely affect the particle bonding microstructure, i.e., the durability or the uniformity of the boron carbide particle distribution, which is the neutron absorber effectiveness.

10.1.7 Thermal Tests

Thermal acceptance testing of the MAGNASTOR system following fabrication and construction is not required. Continued effectiveness of the heat-rejection capabilities of the system may be monitored during system operation using a remote temperature-monitoring system.

The heat-rejection system consists of convection air cooling where air flow is established and maintained by a chimney effect, with air moving from the lower inlets to the upper outlets. Since this system is passive, and air flow is established by the decay heat of the contents of the TSC, it is sufficient to ensure by inspection that the inlet and outlet screens are clear and free of debris that could impede air flow. Because of the passive design of the heat-rejection system, no thermal testing is required.

10.1.8 Cask Identification

Each TSC and concrete cask shall be marked with a model number and an identification number. Each concrete cask will additionally be marked for empty weight and date of loading. Specific marking instructions are provided on the license drawings for these system components.

Chapter 13

4.0 DESIGN FEATURES

4.1 Design Features Significant to Safety

4.1.1 Criticality Control

a) Minimum ^{10}B loading in the neutron absorber material:

Neutron Absorber Type	Required Minimum Effective Areal Density (^{10}B g/cm^2)		% Credit Used in Criticality Analyses	Required Minimum Actual Areal Density (^{10}B g/cm^2)	
	PWR Fuel	BWR Fuel		PWR Fuel	BWR Fuel
Borated Aluminum Alloy	0.036	0.027	90	0.04	0.03
Borated MMC	0.036	0.027	90	0.04	0.03
Boral	0.036	0.027	75	0.048	0.036

b) *Acceptance and qualification testing of neutron absorber material shall be in* accordance with Section 10.1.6.

c) Soluble boron concentration in the PWR fuel pool and water in the TSC shall *be in accordance with* LCO 3.2.1, *with a minimum water temperature* 5-10°F higher than the minimum needed to ensure solubility.

4.1.2 Alternatives to Neutron Absorber Material Testing

Proposed alternatives to neutron absorber material acceptance and qualification testing may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of Section 10.1.6 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

4.1.3 Fuel Cladding Integrity

The licensee shall ensure that fuel oxidation and the resultant consequences are precluded during canister loading and unloading operations.

(continued)

4.2 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2001 Edition with Addenda through 2003, Section III, Subsection NB, is the governing Code for the design, material procurement, fabrication, and testing of the TSC.

The ASME Code, 2001 Edition with Addenda through 2003, Section III, Subsection NG, is the governing Code for the design, material procurement, fabrication and testing of the spent fuel baskets

The American Concrete Institute Specifications ACI-349 and ACI-318 govern the CONCRETE CASK design and construction, respectively.

The American National Standards Institute ANSI N14.6 (1993) and NUREG-0612 govern the TRANSFER CASK design, operation, fabrication, testing, inspection, and maintenance.

4.2.1 Alternatives to Codes, Standards, and Criteria

Table 2.1-2 of the FSAR lists approved alternatives to the ASME Code for the design, procurement, fabrication, inspection and testing of MAGNASTOR SYSTEM TSCs and spent fuel baskets.

4.2.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria

Proposed alternatives to ASME Code, Section III, 2001 Edition with Addenda through 2003, including alternatives authorized in Table 2.1-2 of the FSAR, may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of ASME Code, Section III, Subsections NB and NG, 2001 Edition with Addenda through 2003, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

(continued)

4.3 Site-Specific Parameters and Analyses

This section presents site-specific parameters and analytical bases that must be verified by the MAGNASTOR SYSTEM user. The parameters and bases presented in Section 4.3.1 are those applied in the design bases analysis.

4.3.1 Design Basis Specific Parameters and Analyses

The design basis site-specific parameters and analyses that require verification by the MAGNASTOR SYSTEM user are:

a. A temperature of 100°F is the maximum average yearly temperature. The three-day average ambient temperature shall be ≤106°F.

b. The allowed temperature extremes, averaged over a three-day period, shall be ≥–40°F and ≤133°F.

c. The analyzed flood condition of 15 fps water velocity and a depth of 50 ft of water (full submergence of the loaded cask) are not exceeded.

d. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the fuel tank of the cask handling equipment used to move the loaded CONCRETE CASK onto or from the ISFSI site contains no more than 50 gallons of fuel.

e. In cases where engineered features (i.e., berms, shield walls) are used to ensure that requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category on a site-specific basis.

f. The TRANSFER CASK shall not be operated and used when surrounding air temperature is < 0°F.

g. The CONCRETE CASK shall not be lifted by the lifting lugs with surrounding air temperatures < 0°F.

h. Loaded CONCRETE CASK lifting height limit ≤24 inches.

(continued)

4.4 TSC Handling and Transfer Facility

The TSC provides a leaktight confinement boundary and is evaluated for normal and off-normal handling loads. A handling and transfer facility is not required for TSC and TRANSFER CASK handling and transfer operations within a 10 CFR 50 licensed facility.

Movements of the TRANSFER CASK and TSC outside of a 10 CFR 50 licensed facility are not permitted unless a TSC TRANSFER FACILITY is designed, operated, fabricated, tested, inspected, and maintained in accordance with the following requirements. These requirements do not apply to handling heavy loads under a 10 CFR 50 license.

The permanent or stationary weldment structure of the TSC TRANSFER FACILITY shall be designed to comply with the stress limits of ASME Code, Section III, Subsection NF, Class 3 for linear structures. All compression loaded members shall satisfy the buckling criteria of ASME Code, Section III, Subsection NF.

The reinforced concrete structure of the facility shall be designed in accordance with ACI-349 and the factored load combinations set forth in ACI-318 for the loads defined in Table 4-1 shall apply. TRANSFER CASK and TSC lifting devices installed in the handling facility shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with NUREG-0612, Section 5.1.

If mobile load lifting and handling equipment is used at the facility, that equipment shall meet the guidelines of NUREG-0612, Section 5.1, with the following conditions:

a. The mobile lifting device (i.e., crane) shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6 (1)(a), and shall be capable of stopping and holding the load during a design earthquake event;

b. The mobile lifting device shall contain ≤50 gallons of flammable liquid during operation inside the ISFSI;

c. Mobile cranes are not required to meet the guidance of NUREG-0612, Section 5.1.6(2) for new cranes;

d. The mobile lifting device shall conform to the requirements of ASME B30.5, “Mobile and Locomotive Cranes”;

e. Movement of the TSC or CONCRETE CASK in a horizontal orientation is not permitted.

Table 4-1 Load Combinations and Service Condition Definitions for the TSC Handling and Transfer Facility Structure

Load Combination	ASME Section III Service Condition for Definition of Allowable Stress	Note
D* D + S	Level A	All primary load bearing members must satisfy Level A stress limits
D + M + W'[1] D + F D + E D + Y	Level D	Factor of safety against overturning shall be ≥ 1.1, if applicable.

D	=	Crane hook dead load
D*	=	Apparent crane hook dead load
S	=	Snow and ice load for the facility site
M	=	Tornado missile load of the facility site[1]
W'	=	Tornado wind load for the facility site[1]
F	=	Flood load for the facility site
E	=	Seismic load for the facility site
Y	=	Tsunami load for the facility site

1. Tornado missile load may be reduced or eliminated based on a Probabilistic Risk Assessment for the facility site.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

5.1 Administrative Programs

The following programs shall be established, implemented, and maintained:

5.1.1 Radioactive Effluent Control Program

A program shall be established and maintained to:

1. Implement the requirements of 10 CFR 72.44 (d) or 10 CFR 72.126, as appropriate.
2. Provide limits on the surface contamination of the TSC and TRANSFER CASK, and verification of meeting those limits prior to removal of the loaded TSC and/or TRANSFER CASK from the 10 CFR 50 structure.
3. Provide an effluent monitoring program, as appropriate, if surface contamination limits are greater than the values specified in Regulatory Guide 1.86.

5.1.2 TSC Loading, Unloading, and Preparation Program

A program shall be established and maintained to implement the FSAR, Chapter 9 requirements for loading fuel and components into the TSC, unloading fuel and components from the TSC, and preparing the TSC and CONCRETE CASK for storage. The requirements of the program for loading and preparing the TSC shall be completed prior to removing the TSC from the 10 CFR 50 structure. The program shall provide for evaluation and control of the following FSAR requirements during the applicable operation:

- Verify that no TRANSFER CASK handling or CONCRETE CASK handling using the lifting lugs occurs when the ambient temperature is < 0°F.
- The water temperature of a water-filled, or partially filled, loaded TSC shall be shown by analysis and/or measurement to be less than boiling at all times.
- Verify that the drying time, cavity vacuum pressure, and component and gas temperatures ensure that the fuel cladding temperature limit of 400°C is not exceeded during TSC preparation activities, and that the TSC is adequately dry.
- Verify that the helium backfill purity and mass assure adequate heat transfer and preclude fuel cladding corrosion.
- The inner port cover welds to the closure lid at the vent port and at the drain port shall be qualified in accordance with the procedures in Section 9.1.1.

continued

- Verify that the time to complete the transfer of the TSC from the TRANSFER CASK to the CONCRETE CASK and from a CONCRETE CASK to another CONCRETE CASK assures that the fuel cladding temperature limit of 400°C is not exceeded.
- The surface dose rates of the CONCRETE CASK are adequate to allow proper storage and to assure consistency with the offsite dose analysis.
- The equipment used to move the loaded CONCRETE CASK onto or from the ISFSI site contains no more than 50 gallons of flammable liquid.

This program will control limits, surveillances, compensatory measures and appropriate completion times to assure the integrity of the fuel cladding at all times in preparation for and during LOADING OPERATIONS, UNLOADING OPERATIONS, TRANSPORT OPERATIONS, TRANSFER OPERATIONS and STORAGE OPERATIONS, as applicable.

5.1.3 Transport Evaluation Program

A program that provides a means for evaluating transport route conditions shall be developed to ensure that the design basis impact g-load drop limits are met. For lifting of the loaded TRANSFER CASK or CONCRETE CASK using devices that are integral to a structure governed by 10 CFR 50 regulations, 10 CFR 50 requirements apply. This program evaluates the site-specific transport route conditions and controls, including the transport route road surface conditions; road and route hazards; security during transport; ambient temperature; and equipment operability and lift heights. The program shall also consider drop event impact g-loading and route subsurface conditions, as necessary.

5.1.4 ISFSI Operations Program

A program shall be established to implement FSAR requirements for ISFSI operations.

At a minimum, the program shall include the following criteria to be verified and controlled:

a. Minimum CONCRETE CASK center-to-center spacing.

b. ISFSI pad parameters (i.e., thickness, concrete strength, soil modulus, reinforcement, etc.) are consistent with the FSAR analyses.

c. Maximum CONCRETE CASK lift heights ensure that the g-load limits analyzed in the FSAR are not exceeded.