7. CONFINEMENT

7.1 Confinement Boundary

The 24PT1-DSC is a high integrity stainless steel welded vessel that provides confinement of radioactive materials, encapsulates the fuel in a helium atmosphere and provides biological shielding during 24PT1-DSC closure and transfer and storage operations. The 24PT1-DSC is designed to maintain confinement of radioactive material within the limits of 10CFR 72.104(a), 10CFR 72.106(b) and 10CFR 20 under normal, off-normal, and credible accident conditions. Chapter 3 concludes that the design including the helium atmosphere within the 24PT1-DSC will adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage. The design ensures that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage.

The cylindrical shell, and the inner top and bottom cover plates form the confinement boundary for the spent fuel. The vent and siphon cover plate welds and the vent and siphon block weld are also included in the confinement boundary. The outer top and bottom cover plate's function as redundantly welded barriers for confining radioactive material within the 24PT1-DSC. The dimensions and material descriptions for the confinement boundary assemblies and the redundantly welded barriers are discussed in Chapter 1. The components important to safety are identified in Chapter 2.

7.1.1 Confinement Vessel

The cylindrical shell and inner shell to bottom cover plate welds are made during fabrication of the 24PT1-DSC and are fully compliant to ASME Section III [7.1], Subsection NB. The vent and siphon block weld is also made during fabrication. The inner top cover plate weld is made after fuel loading. This weld is fully compliant to ASME Code Case N-595-1. Both top plug penetrations (siphon and vent ports) and cover plates are welded after drying operations are complete.

Stringent design and fabrication requirements ensure that the confinement function of the 24PT1-DSC is maintained. The shell and inner bottom cover plate are pressure tested in accordance with the ASME Code, Section III, Subarticle NB-6300. This pressure test is performed after installation of the inner bottom cover plate and may be performed concurrently with the leak test, provided the requirements of NB-6300 are met.

Following the pressure test, a leak test of the shell assembly, including the inner bottom cover plate, is performed in accordance with ANSI N14.5 [7.2] and the ASME Code, Section V, Article 10. These tests are typically performed at the fabricator. The acceptance criteria for the test is "leaktight" as defined in ANSI N14.5-1997 [7.2].

The process involved in leak testing the 24PT1-DSC involves temporarily sealing the shell from the top end. The gas filled envelope and evacuated envelope testing methodologies have the required nominal test sensitivity for leaktight construction and are used for leak testing. A helium mass spectrometer is used to detect any leakage as defined in ANSI N14.5 [7.2].

During final drying and sealing operations of the 24PT1-DSC, the top closure confinement welds are applied to confine radioactive materials within the cavity. The inner top cover plate is welded to the shell using automated welding equipment. Once the 24PT1-DSC has been vacuum dried, backfilled with helium and both top plug penetrations welded, the outer top cover plate is lowered onto the 24PT1-DSC. The outer top cover plate is also welded in place using automated welding equipment. The outer top cover plate to shell weld acts as a redundant barrier for confining radioactive material within the 24PT1-DSC throughout its service life.

Leak testing of the 24PT1-DSC inner top cover plate to shell weld, vent and siphon cover plate and vent/siphon block to shell welds is performed by pulling a vacuum between the inner and outer top cover plates through a test port in the outer top cover plate and monitoring for helium in accordance with the requirements of ANSI N14.5 [7.2].

7.1.2 Confinement Penetrations

All penetrations in the 24PT1-DSC confinement boundary are welded closed. The 24PT1-DSC is designed and tested to be "leaktight" as described above.

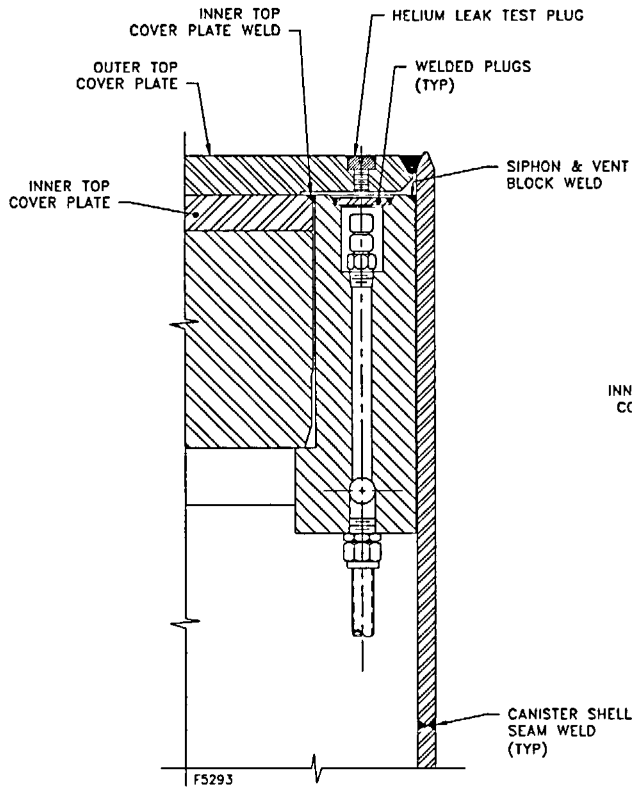
7.1.3 Seals and Welds

The welds made during fabrication of the 24PT1-DSC that affect the confinement boundary include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are inspected (radiographic or ultrasonic inspection, and liquid penetrant inspection) according to the requirements of Subsection NB of the ASME Code. The vent and siphon block weld is also made during fabrication and is liquid penetrant inspected in accordance with Subsection NB of the ASME Code.

The welds applied to the vent and siphon port covers and the inner and outer top cover plates (including test plug) during closure operations, define the confinement boundary at the top end of the 24PT1-DSC. These welds are applied using a multiple-layer technique with multi-level PT in accordance with Subsection NB of the ASME Code and Code Case N-595-1. This effectively eliminates any pinhole leak which might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Figure 7.1-1 provides a graphic representation of the confinement boundary welds.

7.1.4 Closure

The 24PT1-DSC is closed entirely by welding and thus, no closure devices are utilized for confinement.



NDE REQUIREMENTS FOR
CONFINEMENT BOUNDARY WELDS

CONFINEMENT WELD	NDE REQUIREMENTS	WELD SIZE
INNER TOP COVER PLATE	MULTI-LEVEL PT	PARTIAL PENETRATION
INNER BOTTOM COVER PLATE	RT OR UT AND PT	FULL PENETRATION
LONGITUDINAL AND CIRCUMFERENTIAL SHELL SEAMS	RT AND PT	FULL PENETRATION
TOP PLUGS	MULTI-LEVEL PT	PARTIAL PENETRATION
SIPHON & VENT BLOCK WELD	MULTI-LEVEL PT	PARTIAL PENETRATION

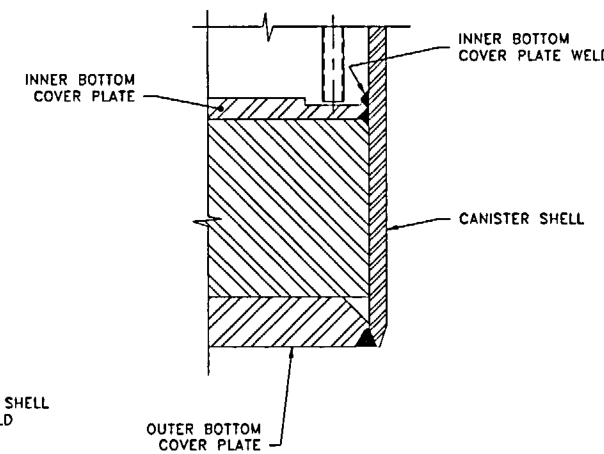


Figure 7.1-1
<u>24PT1-DSC Confinement Boundary Welds</u>

7.2 Requirements for Normal Conditions of Storage

The 24PT1-DSC shell is designed and tested to be "leaktight" to prevent the leakage of radioactive materials. No discernable undetected leakage is credible and the dose at the controlled area boundary from atmospheric release is negligible.

7.2.1 Release of Radioactive Material

Analyses for determining the annual dose equivalent to an individual located at the site boundary or outside the controlled area resulting from releases of radioactive material are not required as defined by NRC Spent Fuel Project Office Interim Staff Guidance-5 (ISG-5) [7.3], since the 24PT1-DSC is designed to be leaktight. Analyses required for determining the annual dose equivalent based on direct radiation for normal, off-normal, and accident conditions are discussed in Chapter 10.

7.2.2 Pressurization of Confinement Vessel

The design provides for drying and evacuation of the 24PT1-DSC interior as part of the loading operations. The design is acceptable for the pressures that may be experienced during these operations as discussed in Chapter 4. On completion of fuel loading, the gas fill of the 24PT1-DSC interior is at a pressure level that will maintain a non-reactive environment for at least the 40 year storage life of the 24PT1-DSC interior under normal, off-normal, and accident conditions.

7.3 Confinement Requirements for Hypothetical Accident Conditions

7.3.1 Fission Gas Products

The 24PT1-DSC confinement boundary is designed and tested to be "leaktight" to prevent the leakage of radioactive materials. The analyses presented in Chapter 11 demonstrate that the confinement boundary is not compromised following hypothetical accident conditions. Therefore, estimating the maximum quantity of fission gas products is not required per ISG-5 [7.3].

7.3.2 Release of Contents

The 24PT1-DSC confinement boundary is designed and tested to be "leaktight" to prevent the leakage of radioactive materials. The analyses presented in Chapter 11 demonstrate that the confinement boundary is not compromised following hypothetical accident conditions. Therefore, confinement analyses for the release of radioactive materials are not required per ISG-5 [7.3].

7.4 Supplemental Data

7.4.1 Confinement Monitoring Capability

The Advanced NUHOMS® System is a self-contained passive system that does not produce routine, solid, liquid or gaseous effluents. Effluent processing systems, or monitoring for airborne or liquid radioactivity, are not required to protect personnel or the environment during storage conditions. Since the 24PT1-DSC is closed entirely by welding, a closure monitoring system is not required as discussed in NRC ISG-5 [7.3].

7.4.2 References

[7.1] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994, including exceptions allowed by Code Case N-595-1.

[7.2] American National Standards Institute, ANSI N14.5-1997, Leakage Tests on Packages for Shipment of Radioactive Materials.

[7.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-5, Revision 1, Confinement Evaluation.

8. OPERATING PROCEDURES

This Chapter presents the operating procedures for the Advanced NUHOMS® System described in previous chapters and shown on the drawings in Chapter 1. The procedures include preparation of the 24PT1-DSC and fuel loading, closure of the DSC, transport to the ISFSI, transfer into the AHSM, monitoring operations, and retrieval from the AHSM. The Advanced NUHOMS® System transfer equipment and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed. Standard fuel and cask handling operations performed under the plant's 10CFR 50 operating license are described in less detail. The licensee may revise existing operational procedures and new ones may be developed according to the requirements of the plant, provided that the system requirements specified in Chapter 12 are met.

These generic Advanced NUHOMS® System procedures have been developed to minimize the amount of time required to complete the subject operations, to minimize personnel exposure, and to assure that all operations required for 24PT1-DSC loading, closure, transfer, storage, and unloading are performed safely. Plant specific ISFSI procedures are to be developed by each licensee in accordance with the requirements of 10CFR 20, 10CFR 50 or 10CFR 72, as applicable. The generic procedures presented here are provided as a guide for the preparation of plant specific procedures and serve to demonstrate how the Advanced NUHOMS® System operations are to be accomplished. Procedures should include identification of potential hazards and mitigating actions applicable to site specific configurations. These procedures are not intended to be limiting in that the licensee may decide that alternate acceptable means are available to accomplish the same operational objective.

These guidelines provide operational sequences to ensure that occupational and site dose limits, as determined in Chapter 10, can be maintained.

8.1 Procedures for Loading the 24PT1-DSC and Transfer to the AHSM

The following sections outline the typical operating procedures for loading the 24PT1-DSC into the Transfer Cask and for transferring the loaded canister to the AHSM.

8.1.1 Narrative Description

The following steps describe the recommended generic operating procedures for the Advanced NUHOMS® System. A list of instrumentation used during loading operations is provided in Table 8.1-1. Flowcharts of Advanced NUHOMS® System loading operations are provided in Figure 8.1-1. A pictorial representation of key phases of this process is provided in Figure 8.1-2.

8.1.1.1 Transfer Cask and 24PT1-DSC preparation

1. Prior to placement in dry storage, the candidate fuel assemblies are to be visually examined to insure that no known or suspected gross cladding breaches exist. Pinholes and hairline cracks are acceptable. Visual verification of fuel integrity may also be accomplished in conjunction with existing plant records. The assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Chapter 12.
2. Prior to being placed in service, the transfer cask is to be cleaned or decontaminated as necessary to insure an acceptable surface contamination level.
3. Place the transfer cask in the vertical position in the cask decon area using the cask handling crane and the transfer cask lifting yoke (the design basis and criteria for the yoke is addressed in Chapter 1).
4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.
5. Remove the cask top cover plate and examine the cask cavity for any physical damage and ready the cask for service.
6. Examine the 24PT1-DSC for any physical damage which might have occurred since the receipt inspection was performed. The 24PT1-DSC is to be cleaned and any loose debris removed. Verify that *bottom* fuel spacers are present in all fuel assembly slots.

   Failed fuel cans, if required for loading damaged fuel assemblies, should be placed into the 24PT1-DSC basket prior to lowering the 24PT1-DSC/TC into the spent fuel pool. The failed fuel cans shall be inspected prior to placement in the 24PT1-DSC. This inspection shall ensure cleanliness and determine if any damage has occurred to the failed fuel cans since receipt inspection. The failed fuel can cover is to be removed from the failed fuel can for installation after fuel loading.
7. Using a crane, lower the 24PT1-DSC into the cask cavity by the internal lifting lugs and rotate the 24PT1-DSC to match the cask alignment marks.

8. Place the top shield plug onto the 24PT1-DSC. Examine the top shield plug to ensure a proper fit.

9. Fill the cask/24PT1-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask/24PT1-DSC annulus by pressurizing the seal with compressed air.

NOTE: A transfer cask/24PT1-DSC annulus pressurization tank filled with demineralized water may be connected to the top vent port of the transfer cask via a hose to provide a positive head above the level of water in the cask/24PT1-DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the cask/24PT1-DSC annulus, provided a positive head is maintained at all times.

10. Position the cask lifting yoke and engage the cask lifting trunnions and the rigging cables to the 24PT1-DSC top shield plug. Adjust the rigging cables as necessary to obtain even cable tension. Remove the top shield plug and place clear of the cask.

11. Fill the 24PT1-DSC cavity with water from the fuel pool or an equivalent source.

12. Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.

13. Move the scaffolding away from the cask as necessary.

14. Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting hooks. Reinspect the lifting hooks to insure that they are properly positioned on the cask trunnions.

15. Optionally, secure a sheet of suitable material to the bottom of the cask to minimize the potential for ground-in contamination. This step may be done prior to initial placement of the cask in the decontamination area.

16. Prior to the cask being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask/24PT1-DSC volume. If the water placed in the 24PT1-DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

8.1.1.2 24PT1-DSC Fuel Loading

1. Lift the cask/24PT1-DSC and position them over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR 50 cask handling procedures.

2. Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool surface. As the cask is lowered into the pool, spray the exterior

surface of the cask with demineralized water to minimize surface adhesion of contaminated particles.

NOTE: The introduction of demineralized water may dilute fuel pool boron concentration.

3. Place the cask in the location of the fuel pool designated as the cask loading area.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean demineralized water if it is raised out of the fuel pool.

5. Prior to insertion of a spent fuel assembly into the 24PT1-DSC, the identity of the assembly shall be verified by two individuals using an underwater video camera. Read and record the fuel assembly identification number from the fuel assembly and check this identification number against the 24PT1-DSC loading plan that indicates which fuel assemblies are acceptable for dry storage. *Damaged fuel assemblies to be loaded must be loaded in the 24PT1-DSC locations where a failed fuel can has been installed.*

6. Move a candidate fuel assembly (including NFAH where applicable) from a fuel rack in accordance with the plant's 10CFR 50 fuel handling procedures. *Damaged fuel assemblies are to be placed in their assigned 24PT1-DSC locations where a failed fuel can has been placed.*

7. Position the fuel assembly for insertion into the selected 24PT1-DSC storage cell and load the fuel assembly. Repeat Steps 5 through 7 for each SFA loaded. After the 24PT1-DSC has been fully loaded, check and record the identity and location of each fuel assembly.

8. After all the SFAs have been placed into the 24PT1-DSC and their identities verified, fuel spacers are placed above each intact fuel assembly (damaged fuel cans incorporate an integral *top* fuel spacer). *Failed fuel can covers are to be placed onto any failed fuel cans in the 24PT1-DSC.* Position the lifting yoke and the top shield plug and lower the shield plug onto the 24PT1-DSC.

CAUTION: Verify that all the lifting height restrictions specified in Chapter 12 can be met in the following steps which involve lifting of the transfer cask.

9. Visually verify that the top shield plug is properly seated onto the 24PT1-DSC.

10. Position the lifting yoke and verify that it is properly engaged with the cask trunnions.

11. Raise the transfer cask to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

12. Inspect the top shield plug to reverify that it is properly seated onto the 24PT1-DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 11 and 12 as necessary.

13. Continue to raise the cask from the pool and spray the exposed portion of the cask with demineralized water.

14. Drain any excess water from the top of the shield plug back to the fuel pool.

15. Check the radiation levels at the center of the top shield plug and around the perimeter of the cask.

16. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with demineralized water.

17. Move the cask with loaded 24PT1-DSC to the cask decon area.

8.1.1.3 <u>24PT1-DSC Drying and Backfilling</u>

1. Check the radiation levels along the perimeter of the cask. Temporary shielding may be installed as necessary to minimize personnel exposure. Liquid neutron shield, if left unfilled for weight reduction, shall be filled.

2. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.

3. Disengage the rigging cables from the top shield plug, remove the eyebolts. Disengage the lifting yoke from the trunnions and move it clear of the cask.

4. Decontaminate the exposed surfaces of the 24PT1-DSC shell perimeter and remove the inflatable cask/24PT1-DSC annulus seal.

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the 24PT1-DSC shell. Take swipes around the outer surface of the shell and check for smearable contamination in accordance with Chapter 12 limits.

6. Install the automated welding machine onto the inner top cover plate and place the inner top cover plate with the automated welding machine onto the 24PT1-DSC. Verify proper fit-up of the inner top cover plate with the shell.

7. Check radiation levels along the surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

8. Connect the vacuum drying system (VDS) to the 24PT1-DSC and use the liquid pump to drain approximately 60 gallons to the fuel pool. This will lower the water level about four inches below the bottom of the shield plug.

9. Disconnect the VDS from the 24PT1-DSC.

10. Cover the cask/24PT1-DSC annulus to prevent debris and weld splatter from entering the annulus.

11. Continuous hydrogen monitoring during the welding of the inner cover plate is required [8.1]. Insert a piece ¼" tygon tube (or equivalent) of sufficient length through the vent port such that it terminates just below the shield plug. Connect the tube to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the 24PT1-DSC cavity during welding of the inner cover plate. The 24PT1-DSC internal pressure is to be maintained at atmospheric pressure during welding of the inner top cover plate.

12. Ready the automated welding machine and tack weld the inner top cover plate to the 24PT1-DSC shell. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.1]. If this limit is exceeded, stop all welding operations and purge the 24PT1-DSC cavity with 2-3 psig helium (or other inert medium) via the ¼" tubing to reduce hydrogen concentration safely below the 2.4% limit. Complete the inner top cover plate weldment and remove the automated welding machine.

13. Perform dye penetrant weld examination of the inner top cover plate weld.

14. If required to limit elastic deformation of inner top cover plate, place the strongback so that it sits on the inner top cover plate and is oriented such that:

    - The siphon and vent ports are accessible.
    - The strongback stud holes line up with the cask lid bolt holes.

15. Lubricate the studs and, using a crossing pattern, adjust the strongback studs to snug tight ensuring approximately even pressure on the cover plate.

16. Connect the VDS to the siphon and vent ports.

17. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

18. Engage the nitrogen or helium supply and open the valve on the vent port and allow compressed gas to force the water from the 24PT1-DSC cavity through the siphon port.

19. Once the water stops flowing from the 24PT1-DSC, close the siphon port and disengage the gas source.

20. Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.

21. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the 24PT1-DSC cavity. The cavity pressure should be reduced in steps to approximately 100 torr, 50 torr, 25 torr, 15 torr, 10 torr, 5 torr, and 3 torr. This staged drawdown will prevent ice blockage of the evacuation path. After pumping down to each level, the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 torr or less as specified in Chapter 12.

CAUTION: Verify time duration limitation for vacuum drying is in accordance with Chapter 12 requirements.

22. Open the valve to the vent port and fill the 24PT1-DSC cavity with helium.

NOTE: Helium gas introduced into the DSC shall be welding grade (>99% purity).

23. Pressurize the 24PT1-DSC with helium in accordance with Chapter 12 requirements.

24. Perform a helium sniff test on the inner top cover plate and vent/siphon block.

25. If a leak is found, repair the weld in accordance with the Code of Construction. Repressurize the 24PT1-DSC and repeat the helium sniff test.

26. Once no leaks are detected, depressurize the 24PT1-DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

27. Re-evacuate the 24PT1-DSC cavity using the VDS. The cavity pressure should be reduced in steps to approximately 10 torr, 5 torr, and 3 torr. After pumping down to each level, the pump is valved off and the cavity pressure is monitored. When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 torr or less in accordance with Chapter 12 requirements.

28. Open the valve on the vent port and allow helium to flow into the cavity to pressurize the 24PT1-DSC in accordance with the limits specified in Chapter 12.

29. Close the valves on the helium source.

NOTE: If during drying and backfilling the system is inadvertently vented, re-evacuation and backfilling with helium will be required.

8.1.1.4 24PT1-DSC Sealing Operations

1. Disconnect the VDS from the 24PT1-DSC. Seal weld the prefabricated covers over the vent and siphon ports and perform a dye penetrant weld examination.

2. Install the automated welding machine onto the outer top cover plate and place the outer top cover plate with the automated welding system onto the 24PT1-DSC. Verify proper fit up of the outer top cover plate.

3. Tack weld the outer top cover plate to the 24PT1-DSC shell. Place the outer top cover plate weld root pass. Perform dye penetrant examination of the root pass weld.

4. Remove the outer top cover plate leak test port plug.

5. Perform a helium leak test of the inner top cover plate and vent/siphon port by establishing a vacuum between the inner and outer cover plate. The acceptance criteria for this leak test is defined in Chapters 7 and 12. If a leak is found, repair and retest until acceptance criteria is met.

6. Replace outer top cover plate leak test port plug and seal weld.

7. Weld out the outer top cover plate to the shell and perform dye penetrant examination on the weld surface.

8. Open the cask drain port valve and remove the remaining water from the cask/24PT1-DSC annulus.

9. Remove the automated welding machine from the 24PT1-DSC.

10. Rig the cask top cover plate and lower the cover plate onto the cask.

11. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

8.1.1.5 Transfer Cask Downending and Transport to ISFSI

1. Verify liquid neutron shield, if used, is filled. Re-attach the transfer cask lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.

CAUTION: Verify that the requirements of Chapter 12 lifting controls are met prior to the next step.

2. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid onto the transfer trailer.

3. The transfer trailer should be positioned so that the cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

4. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

5. Move the crane while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

6. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.

7. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

8. Inspect the trunnions to insure that they are properly seated onto the skid and install the trunnion tower closure plates.

8.1.1.6 24PT1-DSC Transfer to the AHSM

CAUTION: Verify that the requirements of Chapter 12 lifting controls are met prior to the next step. *The maximum lifting height and ambient temperature requirements must be met during transfer from the fuel building to the AHSM.*

1. Using a suitable heavy haul tractor, transfer the loaded cask from the plant's fuel building to the ISFSI along the designated transfer route.

2. Prior to aligning the cask, remove the AHSM door using a portable crane, inspect the cavity of the AHSM, remove any debris and prepare the AHSM to receive a 24PT1-DSC. The doors on adjacent AHSMs should remain in place.

3. Inspect the AHSM air inlet and outlet to ensure that they are clear of debris. Inspect the screens on the air inlet and outlet for damage.

4. Position the transfer trailer to within a few feet of the AHSM.

5. Check the position of the trailer to ensure the centerline of the AHSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Using a portable crane, unbolt and remove the cask top cover plate.

7. Back the trailer to within a few inches of the AHSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer and extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer. Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the AHSM. Using optical survey equipment and the alignment marks on the cask and the AHSM, adjust the position of the cask until it is properly aligned with the AHSM.

9. Using the skid positioning system, fully insert the cask into the AHSM access opening docking collar.

10. Secure the cask to the front wall embedments of the AHSM using the cask restraints.

11. After the cask is docked with the AHSM, verify the alignment of the transfer cask using the optical survey equipment.

12. Position the hydraulic ram behind the cask in horizontal alignment with the cask and level the ram. Remove the bottom ram access cover plate. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the 24PT1-DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the grapple ring.

14. Recheck all alignment marks and ready all systems for transferring the 24PT1-DSC into the AHSM.

15. Activate the hydraulic ram to initiate insertion of the 24PT1-DSC into the AHSM. Stop the ram when the 24PT1-DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the AHSM. Replace the bottom ram access cover plate.

18. Using the skid positioning system, disengage the cask from the AHSM access opening.

19. Install the 24PT1-DSC seismic restraint.

20. Install the AHSM door using a portable crane and secure it in place.

21. Replace the transfer cask top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

22. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

23. Close and lock the ISFSI access gate and activate ISFSI security measures.

24. Adjust the seismic restraint on the 24PT1-DSC one week following initial placement.

8.1.1.7 Monitoring Operations

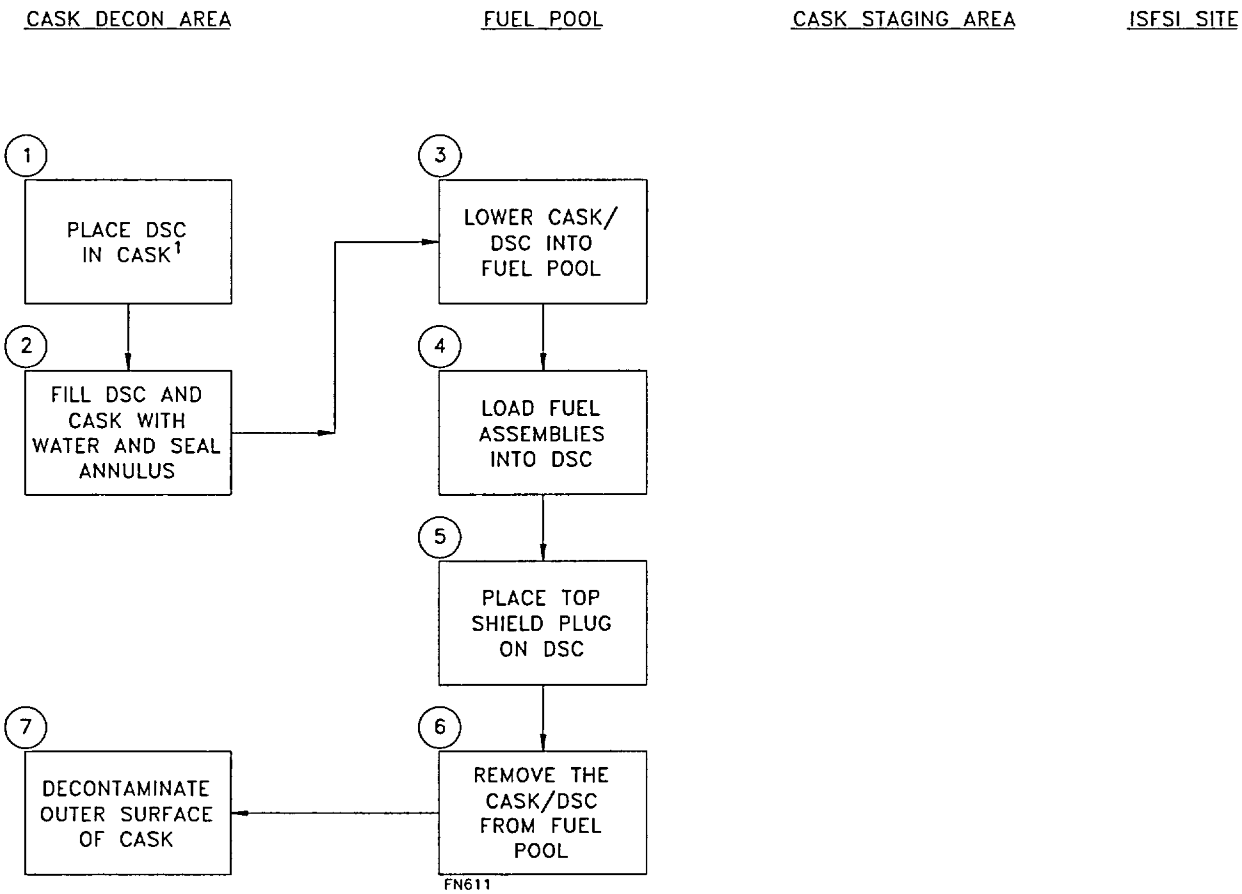
1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform a daily visual surveillance of the AHSM air inlets and outlets to insure that no debris is obstructing the AHSM vents in accordance with Chapter 12 requirements.

3. Perform a temperature measurement for each AHSM on a daily basis in accordance with Chapter 12 requirements.

Table 8.1-1
Instrumentation Used During Advanced NUHOMS® System Loading Operations

Instruments	Function
1. Gross Gamma/Beta/Neutron Detectors	Measure doses to ensure operations are performed to maintain ALARA doses.
2. Pressure and Vacuum Gauges	Measure helium, air, water and vacuum pressures inside 24PT1-DSC
3. Hydraulic Pressure Gauges and Ram Pressure Relief Valves	Measure and limit hydraulic ram force applied to 24PT1-DSC
4. Optical Survey Equipment	Align cask and ram with AHSM
5. Helium Mass Spectrometer	Measure helium leakage rate



[1] INCLUDING FAILED FUEL CANS, AS REQUIRED.

Figure 8.1-1
Advanced NUHOMS® System Loading Operations Flow Chart

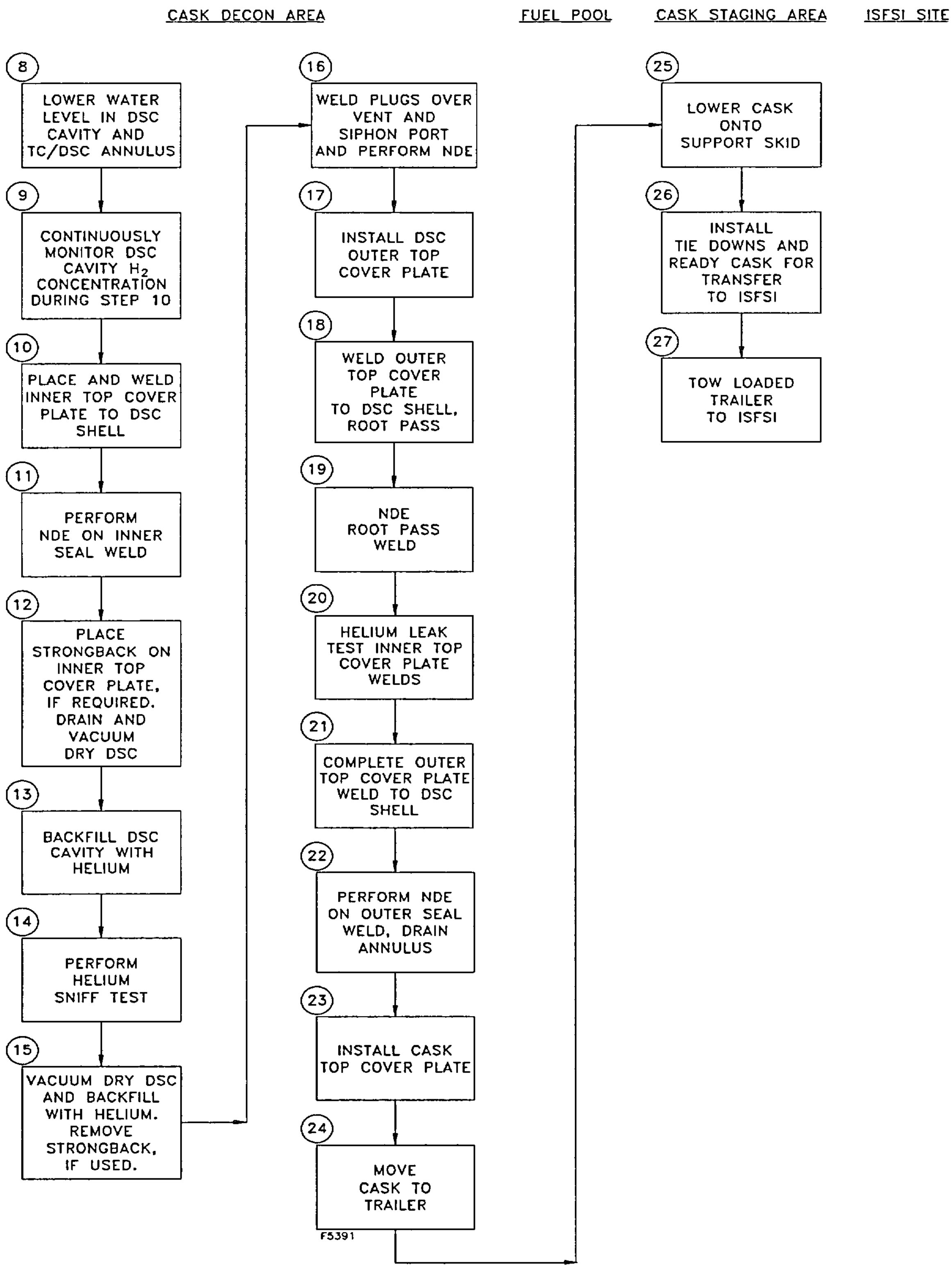


Figure 8.1-1
Advanced NUHOMS® System Loading Operations Flow Chart (continued)

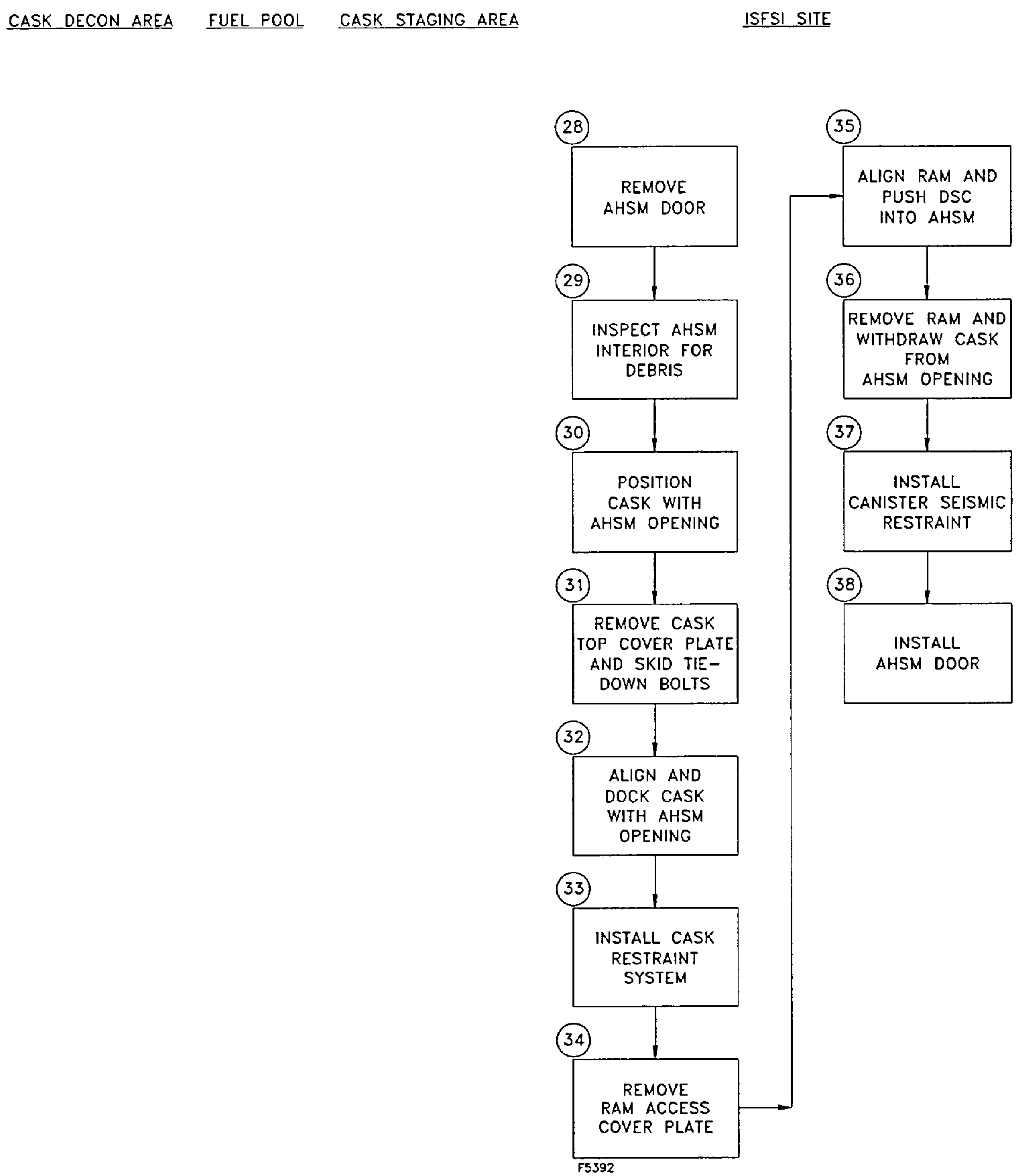


Figure 8.1-1
Advanced NUHOMS® System Loading Operations Flow Chart (concluded)

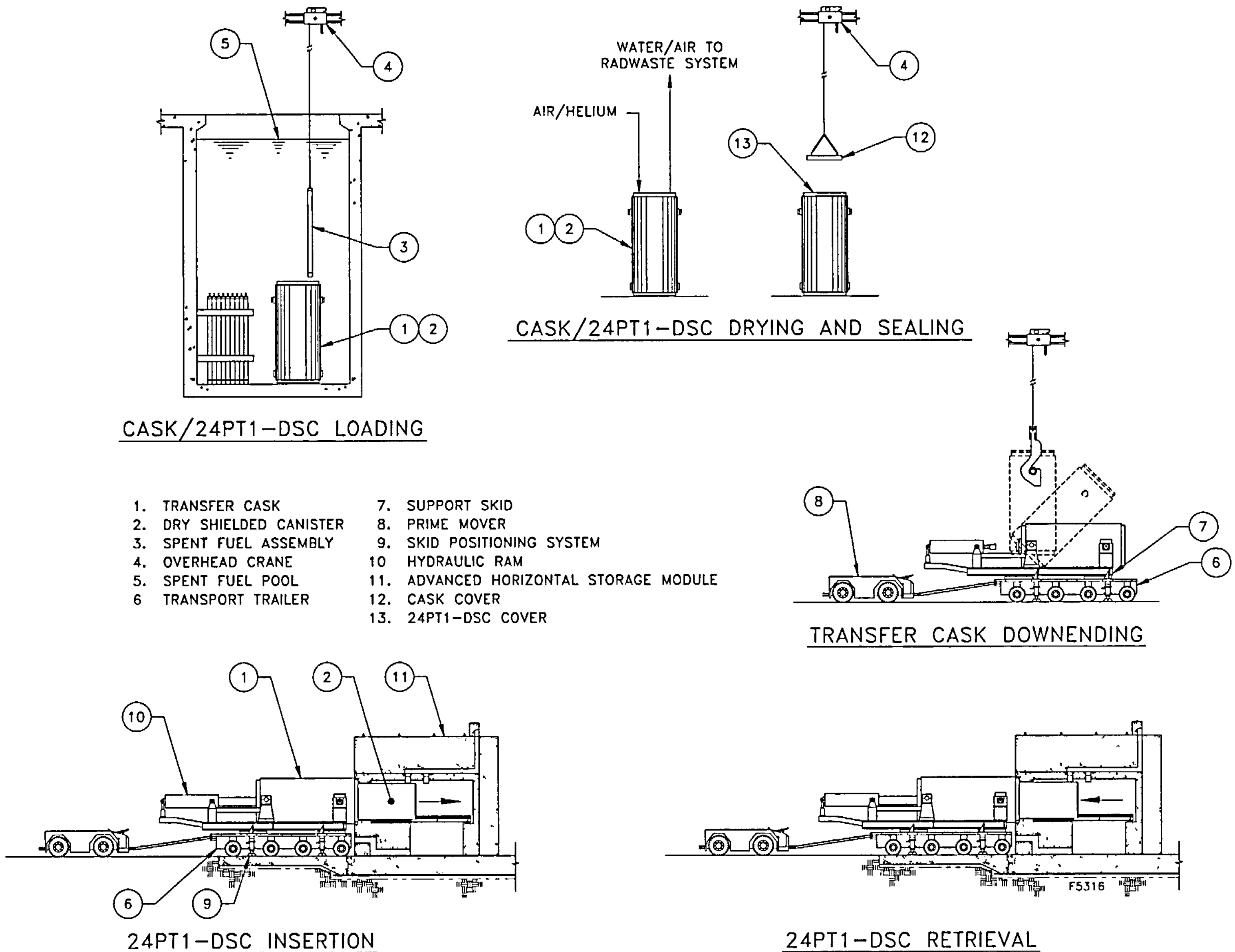


Figure 8.1-2
Cask Loading Operations

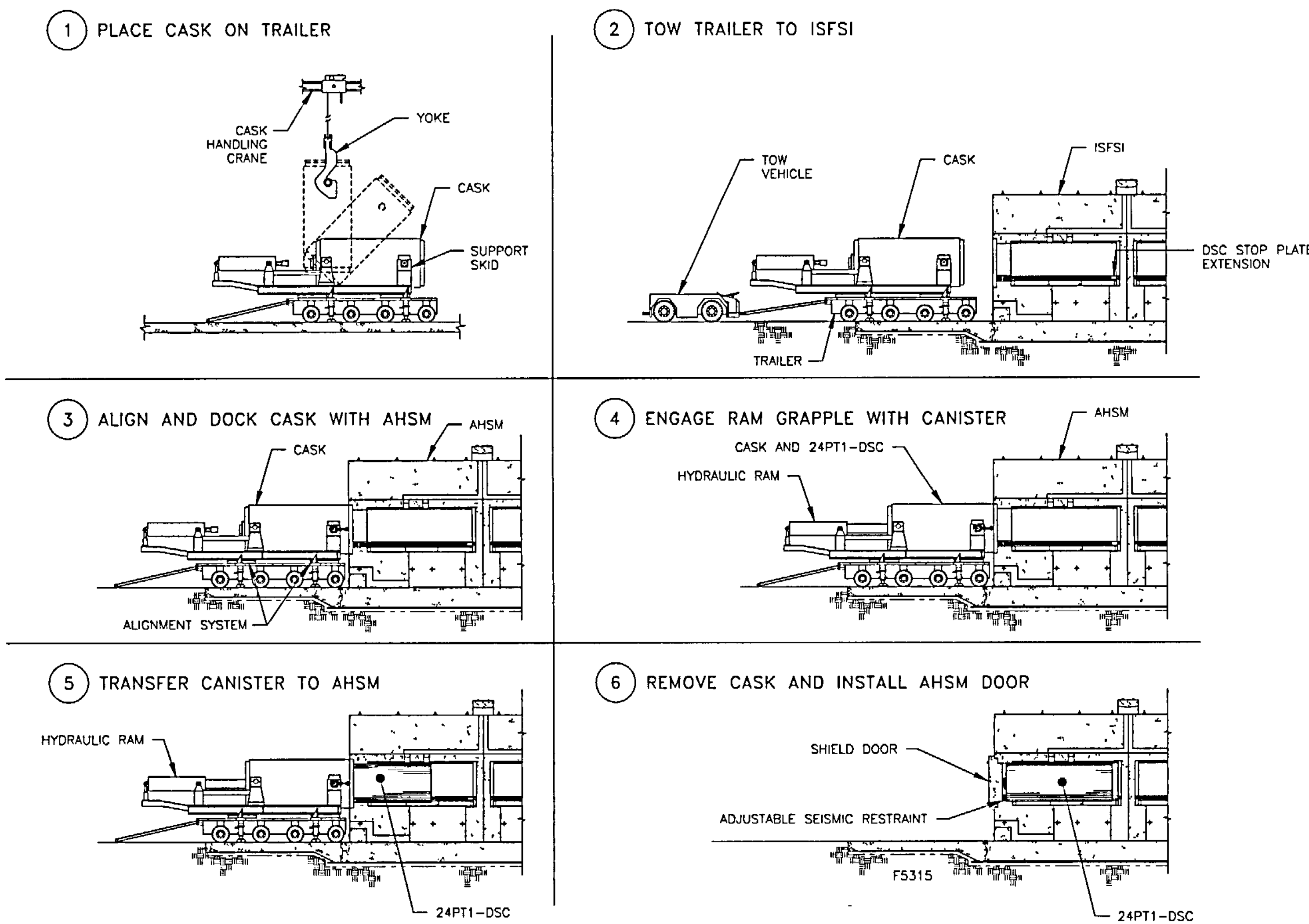


Figure 8.1-2
Cask Loading Operations
(concluded)

8.2 Procedures for Unloading the 24PT1-DSC

The following section outlines the procedures for retrieving the 24PT1-DSC from the AHSM and for removing the fuel assemblies from the 24PT1-DSC. These procedures are provided as a guide and are not intended to be limiting if the licensee determines that alternate means are available to accomplish the same operational objective. A process flow diagram for the Advanced NUHOMS® System retrieval is presented in Figure 8.2-1.

8.2.1 24PT1-DSC Retrieval from the AHSM

CAUTION: Verify that the requirements of Chapter 12 lifting controls are met prior to the next step. The maximum lifting height and ambient temperature requirements must be met during transfer from the AHSM to the fuel building.

1. Ready the transfer cask, transfer trailer, and support skid for service and tow the trailer to the AHSM.
2. Remove AHSM door and seismic restraint. Remove the transfer cask top cover plate. Back the trailer to within a few inches of the AHSM.
3. Using the skid positioning system align the transfer cask with the AHSM and position the skid until the transfer cask is docked with the AHSM access opening.
4. Using optical survey equipment verify alignment of the transfer cask with respect to the AHSM. Install the transfer cask restraints.
5. Install and align the hydraulic ram with the transfer cask.
6. Extend the ram through the transfer cask into the AHSM until it is inserted in the 24PT1-DSC grapple ring.
7. Activate the arms on the ram grapple mechanism to engage the grapple ring.
8. Retract ram and pull the 24PT1-DSC into the transfer cask.
9. Retract the ram grapple arms.
10. Disengage the ram from the transfer cask.
11. Replace the cask ram access cover plate and remove the transfer cask restraints.
12. Using the skid positioning system, disengage the transfer cask from the AHSM.
13. Install the transfer cask top cover plate and ready the trailer for transport.
14. Replace the door and seismic restraint on the AHSM.

8.2.2 Removal of Fuel from the 24PT1-DSC

When the 24PT1-DSC has been removed from the AHSM, there are several potential options for off-site shipment of the fuel. These options include, but are not limited to, shipping the 24PT1-DSC with fuel assemblies or removing the fuel from the 24PT1-DSC as described below. It is preferred to ship the 24PT1-DSC intact to a reprocessing facility, monitored retrievable storage facility or permanent geologic repository in a compatible shipping cask, such as the MP187, licensed under 10CFR 71. However, there are several reasons why it may be necessary to remove fuel assemblies from the 24PT1-DSC. These include off-site transport in a transport cask requiring an alternate canister configuration, return of fuel assemblies to a spent fuel pool, or placement of fuel assemblies in a different 24PT1-DSC. Other reasons might include removing fuel assemblies at the end of service life or for inspection following an accident as discussed in Chapter 12.

If it becomes necessary to remove fuel from the 24PT1-DSC prior to off-site shipment, there are two basic options available at the ISFSI or reactor site. The fuel assemblies could be removed and reloaded into a shipping cask using dry transfer techniques, or if the applicant so desires, the initial fuel loading sequence could be reversed and the plant's spent fuel pool utilized. Procedures for unloading the 24PT1-DSC in a fuel pool are presented here, however wet or dry unloading procedures are essentially identical to those of 24PT1-DSC loading through the weld removal process (beginning of preparation to placement of the transfer cask in the fuel pool). Prior to opening the 24PT1-DSC, the following operations are to be performed.

1. The transfer cask may now be transported to the cask handling area inside the plant's fuel handling building.

2. Position and ready the trailer for access by the crane.

3. Attach the lifting yoke to the crane hook.

4. Engage the lifting yoke with the trunnions of the transfer cask.

5. Visually inspect the yoke lifting hooks to insure that they are properly aligned and engaged onto the transfer cask trunnions.

6. Lift the transfer cask approximately one inch off the trunnion supports. Visually inspect the yoke lifting hooks to insure that they are properly positioned on the trunnions.

7. Move the crane in a horizontal motion while simultaneously raising the crane hook vertically and lift the transfer cask off the trailer. Move the transfer cask to the cask decontamination area.

8. Lower the transfer cask into the cask decontamination area in the vertical position.

9. Wash the transfer cask to remove any dirt which may have accumulated during the 24PT1-DSC unloading and transfer operations.

10. Place scaffolding around the transfer cask so that any point on the surface of the transfer cask is easily accessible to handling personnel.

11. Unbolt the transfer cask top cover plate.

12. Connect the rigging cables to the transfer cask top cover plate and lift the cover plate from the transfer cask. Set the transfer cask cover plate aside and disconnect the lid lifting cables.

13. Install temporary shielding to reduce personnel exposure as required. Fill the transfer cask/24PT1-DSC annulus with clean demineralized water and seal the annulus.

The process of unloading the 24PT1-DSC is similar to that used for loading. Operations that involve opening the 24PT1-DSC described below are to be carefully controlled in accordance with plant procedures. These operations are to be performed under the site's standard health physics guidelines for welding, grinding, and handling of potentially highly contaminated equipment. These are to include the use of prudent housekeeping measures and monitoring of airborne particulates. Procedures may require personnel to perform the work using respirators or supplied air.

If fuel needs to be removed from the 24PT1-DSC, precautions must be taken for the presence of damaged or oxidized fuel and to prevent radiological exposure to personnel during this operation. If degraded fuel is suspected, additional measures appropriate for the specific conditions are to be planned, reviewed, and implemented to minimize exposures to workers and radiological releases to the environment. A sampling of the atmosphere within the 24PT1-DSC should be taken prior to inspection or removal of fuel.

If the work is performed outside the fuel handling building, a tent may be constructed over the work area which may be kept under a negative pressure to control airborne particulates. Any radioactive gas release will be Kr-85, which is not readily captured. Whether the krypton is vented through the plant stack or allowed to be released directly depends on the plant operating requirements.

Following opening of the 24PT1-DSC, it is filled with demineralized or pool water prior to placement in the spent fuel pool to prevent a sudden inrush of pool water. Parameters related to reflooding the 24PT1-DSC cavity are addressed in Chapter 3. Place transfer cask into the pool. Fuel unloading procedures will be governed by the plant operating license under 10CFR 50. The generic procedures for these operations are as follows:

1. Locate the siphon and vent port using the indications on the top cover plate. Place a portable drill press on the top of the 24PT1-DSC. Align the drill over the siphon port.

2. Place an exhaust hood or tent over the 24PT1-DSC, if necessary. The exhaust should be filtered or routed to the site radwaste system.

3. Drill a hole through the top cover plate to expose the siphon port quick connect.

4. Drill a second hole through the top cover plate to expose the vent port quick connect.

   CAUTION:

   (a) The water fill rate must be regulated during this reflooding operation to ensure that the DSC vent pressure does not exceed 20.0 psig.

   (b) Provide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations to ensure that a safety limit of 2.4% hydrogen concentration is not exceeded. Purge with 2-3 psig helium (or any other inert medium) as necessary to maintain the hydrogen concentration safely below this limit.

5. Obtain a sample of the 24PT1-DSC atmosphere (confirm acceptable hydrogen concentration). Fill the 24PT1-DSC with water from the fuel pool through the siphon port with the vent port open and routed to the plant's off-gas system.

6. Place welding blankets around the transfer cask and scaffolding.

7. Using plasma arc-gouging, a mechanical cutting system or other suitable means, remove the seal weld from the outer top cover plate and 24PT1-DSC shell. A fire watch should be placed on the scaffolding with the welder, as appropriate. The exhaust system should be operating at all times.

8. The material or waste from the cutting or grinding process should be treated and handled in accordance with the plant's low level waste procedures unless determined otherwise.

9. Remove the top of the tent, if necessary.

10. Remove the exhaust hood, if necessary.

11. Remove the outer top cover plate.

12. Reinstall tent and temporary shielding, as required. Remove the seal weld from the inner top cover plate to the shell in the same manner as the outer cover plate. Remove the inner top cover plate. Remove any remaining excess material on the inside shell surface by grinding.

13. Clean the transfer cask surface of dirt and any debris which may be on the transfer cask surface as a result of the weld removal operation. Other procedures which

are required for the operation of the transfer cask should take place at this point as necessary.

14. Engage the yoke onto the trunnions, install eyebolts into the top shield plug and connect the rigging cables to the eyebolts.

15. Visually inspect the lifting hooks of the yoke to insure that they are properly positioned on the trunnions.

16. The transfer cask should be lifted just far enough to allow the weight of the transfer cask to be distributed onto the yoke lifting hooks. Inspect the lifting hooks to insure that they are properly positioned on the trunnions.

17. Install suitable protective material onto the bottom of the transfer cask to minimize cask contamination. Move the transfer cask to the spent fuel pool.

18. Prior to lowering the transfer cask into the pool, adjust the pool water level, if necessary, to accommodate the volume of water which will be displaced by the transfer cask during the operation.

19. Position the transfer cask over the cask loading area in the spent fuel pool.

20. Lower the transfer cask into the pool. As the transfer cask is being lowered, the exterior surface of the transfer cask should be sprayed with clean demineralized water.

21. Disengage the lifting yoke from the transfer cask and lift the top shield plug from the 24PT1-DSC.

22. Remove top spacers or failed fuel can covers. Remove the fuel from the 24PT1-DSC and place the fuel into the spent fuel racks. Failed fuel may also be removed along with its failed fuel can (without removal of the failed fuel can cover), if desired.

23. Lower the top shield plug onto the empty DSC.

24. Visually verify that the top shield plug is properly positioned.

25. Engage the lifting yoke onto the cask trunnions.

26. Visually verify that the yoke lifting hooks are properly engaged with the cask trunnions.

27. Lift the transfer cask by a small amount and verify that the lifting hooks are properly engaged with the trunnions.

28. Lift the transfer cask to the pool surface. Prior to raising the top of the transfer cask above the water surface, stop vertical movement and inspect the top shield plug to ensure that it is properly positioned. If the top shield plug is not properly seated, lower the transfer cask back to the fuel pool and reposition the plug.

29. Spray the exposed portion of the transfer cask with demineralized water.

30. Drain any excess water from the top of the top shield plug into the fuel pool.

31. Lift the transfer cask from the spent fuel pool. As the transfer cask is rising out of the pool, spray the transfer cask with demineralized water.

32. Move the transfer cask to the cask decontamination area.

33. Check radiation levels around the perimeter of the transfer cask. The transfer cask exterior surface should be decontaminated if necessary.

34. Place scaffolding around the transfer cask so that any point along the surface of the transfer cask is easily accessible to personnel.

35. Ready the VDS.

36. Connect the VDS to the vent port with the system open to atmosphere. Also connect the VDS to the siphon port and connect the other end of the system to the liquid pump. The pump discharge should be routed to the plant radwaste system or the spent fuel pool.

37. Open the valves on the vent port and siphon port of the VDS.

38. Activate the liquid pump.

39. Once the water stops flowing, deactivate the pump.

40. Close the valves on the VDS.

41. Disconnect the VDS from the vent and siphon ports.

42. Seal DSC cavity, if required.

43. Decontaminate the 24PT1-DSC, as necessary, and handle in accordance with low-level waste procedures. Alternatively, the 24PT1-DSC may be repaired for reuse.

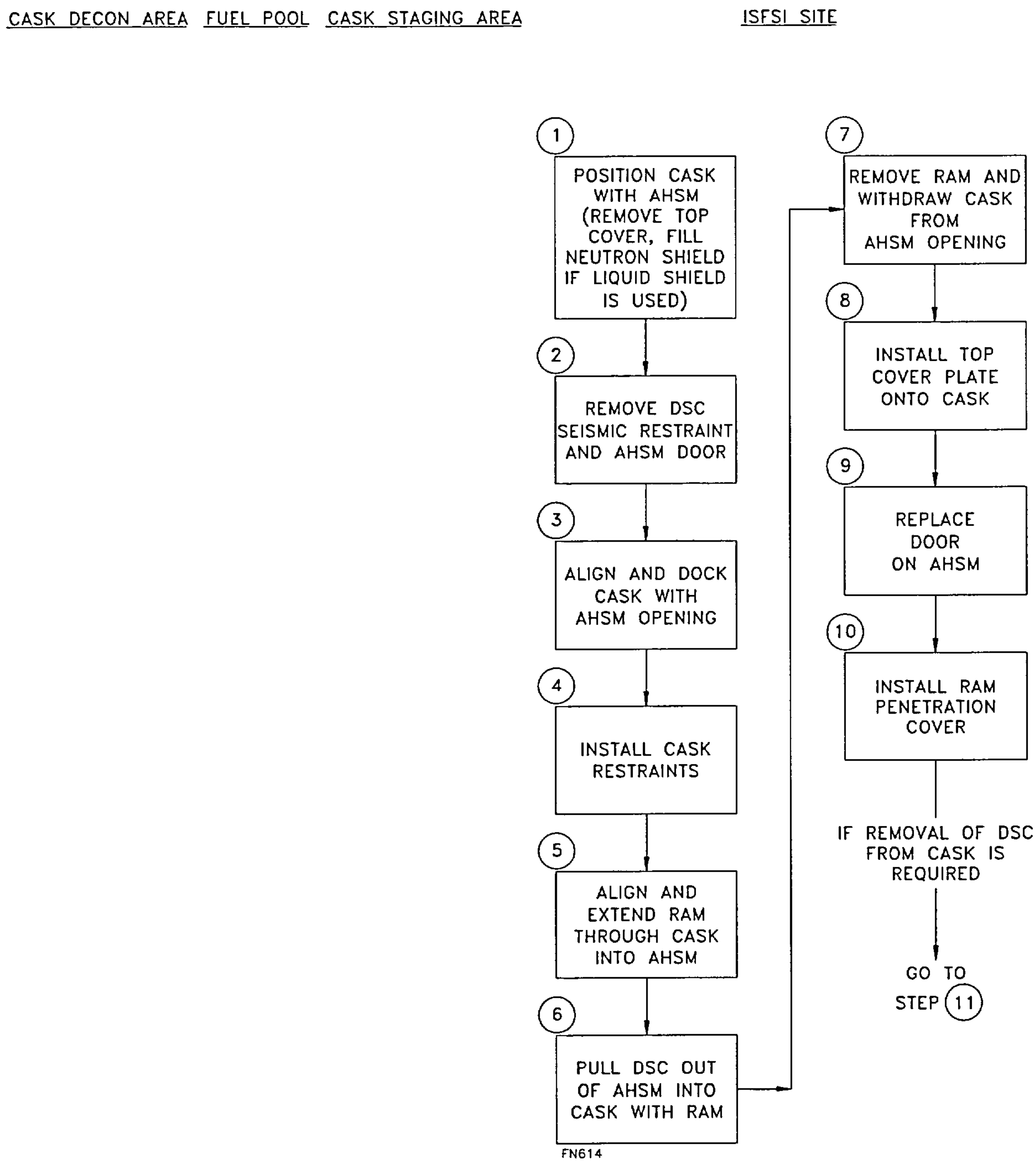


Figure 8.2-1
Advanced NUHOMS® System Retrieval Operations Flow Chart

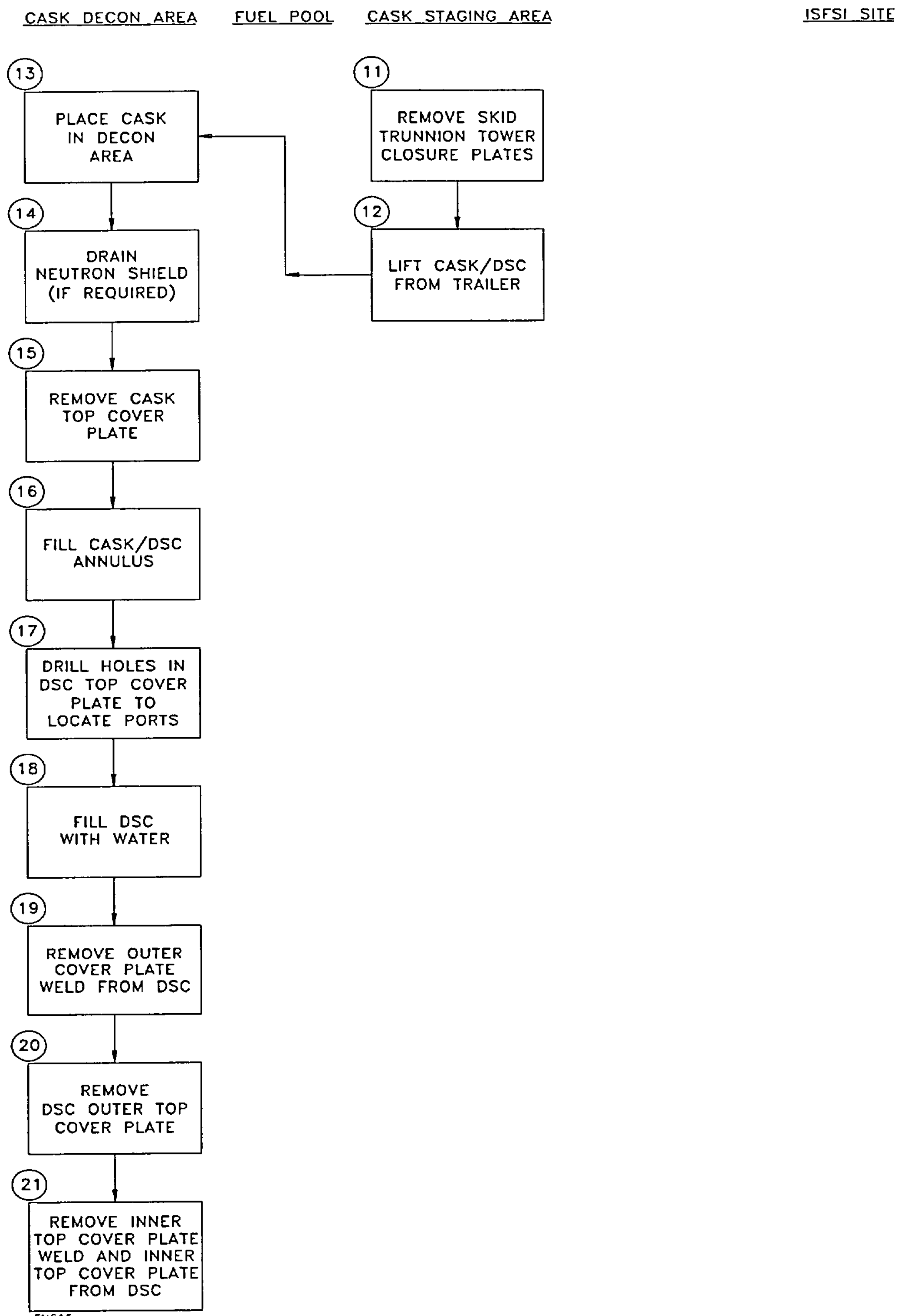


Figure 8.2-1
Advanced NUHOMS® System Retrieval Operations Flow Chart (continued)

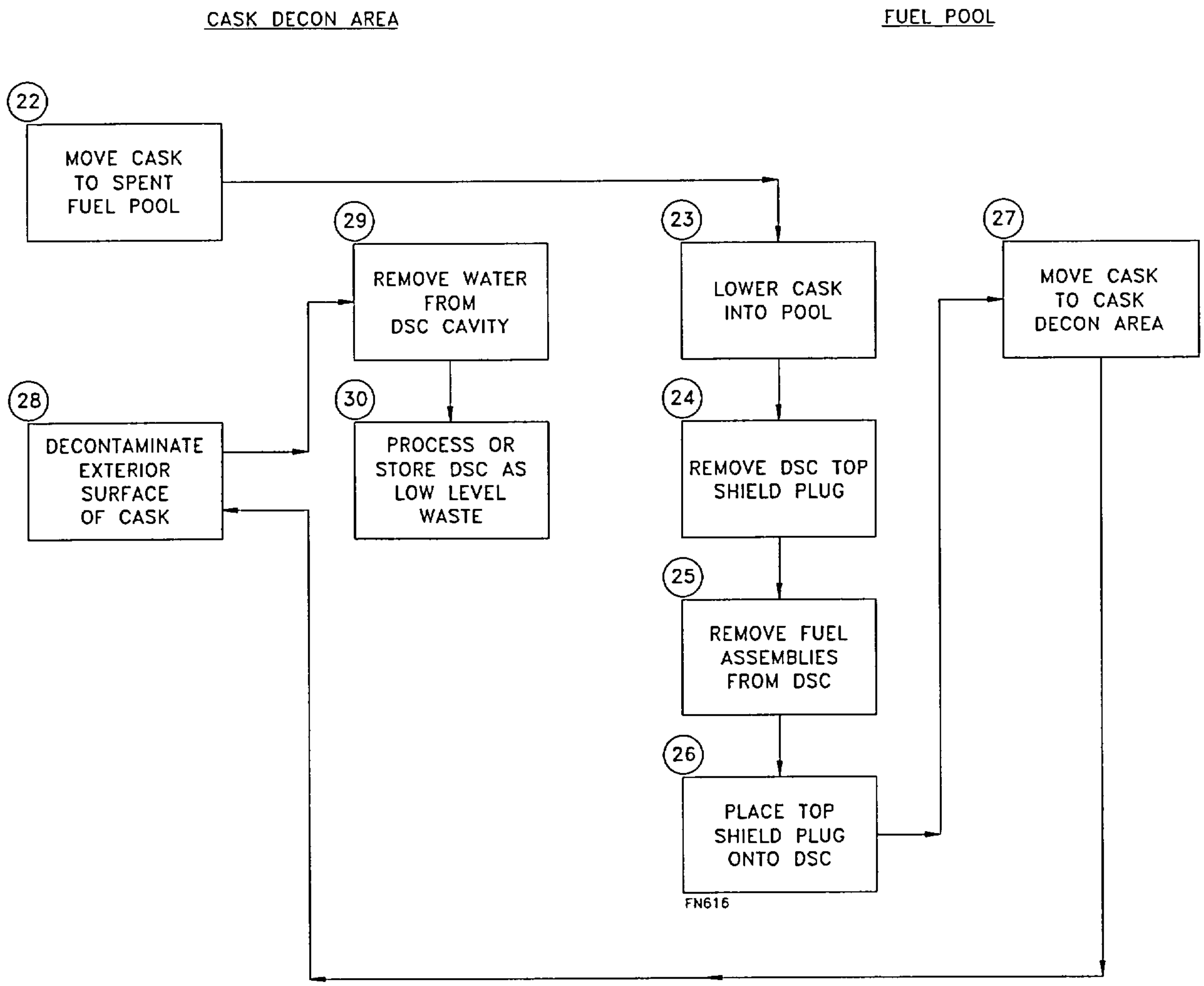


Figure 8.2-1
Advanced NUHOMS® System Retrieval Operations Flow Chart (concluded)

8.3 Supplemental Information

8.3.1 Other Operating Systems

The Advanced NUHOMS® System is a passive storage system and requires no operating systems other than those systems used in transferring the 24PT1-DSC to and from the AHSM.

8.3.1.1 Component/Equipment Spares

As discussed in Chapter 11, the Advanced NUHOMS® ISFSI is designed to withstand all postulated design basis events. Therefore, no storage component or equipment spares are required for the Advanced NUHOMS® System.

8.3.2 Operation Support System

The Advanced NUHOMS® System is a self contained passive system and requires no effluent processing systems during storage conditions.

8.3.2.1 Instrumentation and Control System

There are no instrumentation and control systems used during storage conditions, except for the AHSM temperature monitoring. The instrumentation and controls necessary during 24PT1-DSC loading, closure and transfer are described in Section 8.1.

8.3.2.2 System and Component Spares

Other than spares for the AHSM temperature monitoring, there are no instrumentation or control systems used during storage conditions; thus, no other system and component spare parts are required.

8.3.3 Control Room and/or Control Areas

There are no control room or control areas for the Advanced NUHOMS® System.

8.3.4 Analytical Sampling

The only analytical sampling used with the Advanced NUHOMS® System is the continuous monitoring [8.1] of the hydrogen concentration in the 24PT1-DSC cavity during welding of the inner top cover plate.

8.3.5 References

[8.1] U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards, "Safety Evaluation of VECTRA Technologies' Response to Nuclear Regulatory Commission Bulletin 96-04 for NUHOMS®-24P and NUHOMS®-7P Dry Spent Fuel Storage System," November 1997 (Dockets 72-1004, 72-3, 72-4, 72-8, and 72-14).

9. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

9.1 Acceptance Criteria

A comprehensive pre-operational testing program has been carried out at Carolina Power and Light's Robinson plant NUHOMS® ISFSI. This program was developed jointly and conducted by Carolina Power and Light, the Department of Energy, the Electric Power Research Institute and NUTECH (now *TN*). These pre-operational testing results [9.1] are applicable to the Advanced NUHOMS® System because of the similarity of system operations. These results provide sufficient data to demonstrate that the analytical methods described in this SAR provide conservative thermal and radiological results.

Prior to operation of the ISFSI for a particular plant, the licensee should perform functional tests of the in-plant operations, the on-site transfer operations, and 24PT1-DSC insertion and retrieval (operations at the ISFSI). These tests are intended to verify that the storage system components (e.g., 24PT1-DSC, AHSM, transfer cask, transfer equipment, etc.) operate safely and effectively. Such a program has been successfully completed for the NUHOMS® ISFSIs at Duke Power Company's Oconee Nuclear Station, Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant, Toledo Edison's Davis Besse Nuclear Station and Pennsylvania Power and Light's Susquehanna Nuclear Station.

The discussion provided below focuses on requirements for the 24PT1-DSC and the AHSM. Acceptance tests for the transfer cask are addressed by the safety analysis report under which the OS197 transfer cask is fabricated (C of C 72-1004 [9.9]).

9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the 24PT1-DSC and the AHSM conform to the drawings and specifications. The visual inspections include verifying dimensions and the application of specified coatings and that the 24PT1-DSC and AHSM are clean and free of defects. Visual inspections are performed in accordance with the requirements and acceptance criteria specified by the codes applicable to the associated components.

Upon arrival at the site, the 24PT1-DSCs and AHSMs are again inspected to ensure that they have not been damaged during shipment. Conditions which are not in conformance with the drawings and specifications will be repaired or evaluated, in accordance with 10CFR 72.48, for the effect of the condition on the safety function of the components.

9.1.2 Structural

The structural analyses performed on the 24PT1-DSC and AHSM are presented in Chapter 3. To ensure that the 24PT1-DSC and AHSM can perform their design function, all structural materials are chemically and physically tested to ensure that the required properties are met. The reinforced concrete AHSM is designed to meet the requirements of ACI 349-97 [9.11].

Base materials and welds are examined in accordance with the applicable codes (e.g., ASME, ASTM, AWS). All welding is performed using qualified processes and qualified personnel according to the applicable code requirements (e.g., ASME, AWS). NDE requirements for welds are specified on the drawings provided in Chapter 1. Weld-related NDE is performed in accordance with written and approved procedures. NDE personnel are qualified in accordance with SNT-TC-1A [9.3].

The confinement welds on the 24PT1-DSC are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB [9.2] as modified by Code Case N-595-1 as discussed in Chapter 3 and 7.

24PT1-DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

24PT1-DSC lifting lugs are provided for empty DSC handling operations only. These operations are performed away from plant safety related equipment; therefore these lifting lugs are not subject to load testing requirements of ANSI N14.6 [9.4] for heavy loads.

9.1.3 Leak Tests and Hydrostatic Pressure Tests

24PT1-DSC leakage tests are performed on the confinement system at the Fabricator's facility and during 24PT1-DSC closure operations. These tests are usually performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved. The 24PT1-DSC confinement boundary is tested to be "leaktight" as defined in ANSI N14.5 [9.5]. Personnel performing the leakage tests, *both at the fabricator and the loading site,* are qualified in accordance with SNT-TC-1A [9.3].

The 24PT1-DSC is pressure tested in accordance with ASME Code Case N-595-1.

9.1.4 Components

Components that comprise the Advanced NUHOMS® System and which perform an important-to-safety function are described in Chapter 2. The Advanced NUHOMS® System important-to-safety components do not include any active components requiring testing or any additional testing beyond that described in Sections 9.1.1, 9.1.2 and 9.1.3 above.

9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no valves performing a function important to safety. The Advanced NUHOMS® System design does not utilize valves, rupture discs, or fluid transport devices.

9.1.4.2 Gaskets

There are no gaskets performing a function important to safety. The Advanced NUHOMS® System design does not utilize gasket devices.

9.1.4.3 Miscellaneous

No other components of the Advanced NUHOMS® System require testing, except as discussed in this chapter.

9.1.5 Shielding Integrity

The analyses performed to ensure shielding integrity over the lifetime of the Advanced NUHOMS® System are presented in Chapter 5. The gamma and neutron shielding materials used are limited to concrete AHSM components and carbon steel shield plugs in the 24PT1-DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria.

External dose rate surveys are performed at loading to ensure that Technical Specification radiation dose limits are not exceeded for each 24PT1-DSC and AHSM. Details of the shielding materials are discussed in Chapter 5.

9.1.6 Thermal Acceptance

The thermal analysis performed for the Advanced NUHOMS® System is discussed in Chapter 4. This analysis demonstrates that the system will provide adequate thermal performance during storage. Thermal acceptance will be confirmed by monitoring AHSM concrete temperatures. This is performed in accordance with the associated Technical Specification requirements (See Chapter 12). Monitoring of AHSM thermal performance continues throughout the service life of the AHSM in accordance with Chapter 12 requirements. No testing to determine heat removal capability is required.

9.1.7 Neutron Absorber Tests

This section describes the tests, process controls, and measurements relied upon to demonstrate proper fabrication and effectiveness of the neutron absorber sheets.

During fabrication, the neutron absorber sheets used in the 24PT1-DSC are verified to have their minimum total B^{10} per unit area (areal density) of the sandwiched material as specified on the drawings in Chapter 1.

Samples from each sheet of the neutron absorber are retained for testing and record purposes. The minimum B^{10} content per unit area and the uniformity of dispersion within a sheet are verified by chemical analysis and/or neutron attenuation testing. All material certifications, lot control records, and test records are maintained to assure material traceability and are part of the maintenance records discussed in Section 9.2.1.4.

Chemical (destructive) testing is the preferred method because the B^{10} areal density, which is the primary requirement for the material, can be directly measured. This is done by taking two *1 cm* square samples from each end of the rolled product. The size of the *1 cm* sample is small compared to the finished sheet size, 1300 in^2, so that non-uniformities in B^{10} areal density would be readily apparent between the samples. The material on the ends of the rolled product is known to be thinner than the rest of the sheet because of the rolling process. This characteristic of the finished product is confirmed by examination of thickness in four locations on 100% of the sheets. The thinnest of these samples is used for destructive chemical determination of the amount of boron in the sample using written, approved procedures and standard laboratory processes and equipment. Using the area of the sample, and the mass of boron in the sample, the areal boron density may be calculated. Once the areal boron density is known, the areal B^{10} density is calculated using the isotopic assay of the B_4C powder used to manufacture the product. The sample frequency is determined using standard statistical procedures to assure that minimum B^{10} areal densities are achieved with a 95/95 confidence level. This assures that the minimum safety requirements for the neutron absorber sheets are met. The neutron absorber sheets are capable of performing their function throughout the service life of the 24PT1-DSC as discussed in Chapter 6.

Neutron attenuation testing may be performed to augment or replace chemical (destructive) testing as required to demonstrate acceptable minimum areal B^{10} loading of the neutron absorber sheets. Neutron transmission measurements may be performed on samples, or completed sheets, using a neutron diffractometer. If used, the neutron transmission measurements shall be performed using written, approved procedures in accordance with applicable portions of ASTM E94 [9.6], "Recommended Practice for Radiographic Testing", ASTM E142 [9.7], "Controlling Quality of Radiographic Testing", and ASTM E545 [9.8], "Standard Method for Determining Image Quality in Thermal Radiographic Testing". Since this type of testing does not measure directly the B^{10} areal density of the sheets, the results of neutron attenuation tests shall be demonstrated by calculation to show that the minimum specified B^{10} areal densities are achieved with a 95/95 confidence level.

The 95/95 confidence level may be determined by a pre-defined sampling technique and associated criteria for increased sampling as a result of rejection of material in the initial sample. An initial sampling of 100% of the coupons may be employed with a reduced sampling of 50% of the coupons introduced if all coupons in the first 25% of the lot are acceptable. A rejection during reduced inspection will require a return to 100% inspection of the lot.

Alternatively, the 95/95 confidence level may be obtained by a review of test data. A one-sided tolerance limit factor (F) may be applied to the test data (average + F [standard deviation]) to establish the minimum acceptable value required to ensure test results meet the 95/95 confidence level [9.12], [9.13], [9.14] *and* [9.15].

9.2 Pre-Operational Testing and Maintenance Program

The Advanced NUHOMS® System is designed to be totally passive with minimal maintenance requirements. The 24PT1-DSC does not require any maintenance once it is loaded into the AHSM.

The transfer cask is designed to require minimal maintenance. OS197 Transfer cask maintenance is limited to periodic inspection of critical components and replacement of damaged or nonfunctioning components. A discussion of these requirements is provided in the associated cask SAR (C of C 72-1004 [9.9]).

9.2.1 Subsystems Maintenance

9.2.1.1 Inspection (Transfer Cask Only)

The following sections discuss typical subsystem maintenance requirements that are applicable to the OS197 and MP187 cask systems. Detailed requirements for these systems can be found in their associated SARs.

9.2.1.1.1 Routine Inspection

The following inspections should be performed prior to each use of the transfer cask and lifting hardware:

A. Visual inspection of the transfer cask exterior for cracks, dents, gouges, tears, or damaged bearing surfaces. Particular attention should be paid to the transfer cask trunnions and lifting yoke.

B. Visually inspect all threaded parts and bolts for burrs, chafing, distortion or other damage.

C. Check all quick-connect fittings to ensure their proper operation.

D. Visually inspect the interior surface of the cask for any indications of excessive wear.

E. Visually inspect the neutron shield jacket for indications of damage.

F. Visually inspect the Transfer Cask/24PT1-DSC annulus seal for indications of damage.

G. Visually inspect the seal (o-ring) for indications of damage.

9.2.1.1.2 Annual Inspection

The following inspections and tests shall be performed on an annual basis:

A. Test the transfer cask cavity quick-connect fittings and ram penetration seal for leaktightness.

B. Examine the transfer cask trunnions and cask lifting yoke.

Any parts which fail these tests shall be repaired or replaced as appropriate.

The transfer cask is designed to minimize maintenance and repair requirements. Any indications of damage, failure to operate, or excessive wear are evaluated to ensure that the safe operation of the transfer cask is not impaired. Damage which impairs the ability of the transfer cask to properly function shall be repaired or replaced. This work may be performed on site, depending upon the capabilities of the site resources, or at an approved vendor's facility. Repairs will be performed in accordance with the manufacturer's or cask designer's recommendations.

9.2.1.2 Pre-Operational Tests

The testing program conducted by the licensee utilizes a 24PT1-DSC loaded with mock-up fuel (the licensee shall determine the quantity of mock-up fuel assemblies required to demonstrate that the loading and unloading processes are sound for the dry run, and that operations personnel are adequately trained), the transfer cask and associated transfer equipment, and an AHSM. The tests will simulate, as nearly as possible, the actual operations involved in preparing a 24PT1-DSC for storage to ensure that they can be performed safely during actual loading of spent fuel in the ISFSI. Verification of ALARA practices, not achievable during dry runs, will take place during the initial fuel loadings. Guidelines for such tests are provided in the following paragraphs.

A. An actual 24PT1-DSC loaded with mock-up fuel is typically utilized for pre-operational testing. The parameters of the mock-up fuel are provided in Chapter 1. The 24PT1-DSC will be loaded into the transfer cask to verify fit and adequacy of the transfer cask/24PT1-DSC annulus seal. Additionally, the 24PT1-DSC may be used in operational testing of the transfer equipment and AHSM.

B. Functional testing is performed with the transfer cask and lifting yoke. These tests are to ensure that the transfer cask can be safely lifted from the plant's cask receiving area to the cask washdown area. The cask is also partially lowered into the spent fuel pool and positioned in the cask loading area to verify clearances and travel path.

C. The transfer cask is placed on the transfer trailer, which is moved to the ISFSI along the predetermined route and aligned with an AHSM. Compatibility of the transfer trailer with the transfer cask, verification of the transfer route to the ISFSI, and maneuverability within the confines of the ISFSI are verified.

D. The transfer trailer is aligned and docked with the AHSM. The hydraulic ram is used to insert the 24PT1-DSC loaded with mock-up fuel assemblies into the AHSM and then to retrieve it. Transfer of the 24PT1-DSC to the AHSM will verify that the support skid positioning system and the hydraulic ram system operate safely for both insertion and retrieval.

E. The automated welding system will be tested on mock-up cover plates to verify its functionality in welding and opening/cutting the welded canister consistent with applicable code requirements.

9.2.1.2.1 Pre-Operational Test Discussion

Implementation of the test program is discussed in the paragraphs which follow.

A. The purpose of the pre-operational tests is to ensure that a 24PT1-DSC can be properly and safely placed in the spent fuel pool, loaded with spent fuel, transferred to the ISFSI, inserted in the AHSM, and retrieved from the AHSM. Operation of the 24PT1-DSC, transfer cask, and transfer equipment, as well as the associated auxiliary equipment (e.g., automated welding equipment and vacuum drying system), provides such assurance.

B. Detailed procedures are developed and implemented by each licensee.

C. The expected results of the pre-operational tests are the successful completion of the following: placement of a 24PT1-DSC loaded with mock-up fuel assemblies into the transfer cask, placement of the transfer cask into and out of the spent fuel pool, moving the transfer cask loaded with a 24PT1-DSC and mock-up fuel assemblies to the ISFSI, and transfer to/from the AHSM. The tests are deemed successful if the expected results are achieved safely and without damage to any of the components or associated equipment.

D. Should any equipment or components require modification in order to achieve the expected results, it will be retested, as necessary, to confirm that the modification is adequate. Should any pre-operational procedures change in order to achieve the expected results, the changes will be incorporated into the appropriate operating procedures prior to use for fuel transfer.

E. Plant operation is not affected by testing of the ISFSI. Testing operations in the plant's fuel building can generally be conducted concurrently with plant operation except during refueling operations. Testing to be conducted within the plant's fuel building is scheduled so that there is no conflict with refueling. All normal prerequisites for safe handling of components in, or near, the spent fuel pool shall be satisfied, and normal safety and radiological practices employed.

9.2.1.3 Repair, Replacement, and Maintenance

Repair, replacement and maintenance activities may be required as a result of unforeseen events causing damage to the 24PT1-DSC, AHSM or transfer cask. Damage shall be evaluated with respect to the system design requirements. Repair, rework, and/or retest will be performed as determined by the evaluation to restore the system to its original requirements.

9.2.1.4 Maintenance of Records

The licensee shall maintain the maintenance records for the Advanced NUHOMS® System components provided by *TN*.

The Advanced NUHOMS® System component labeling is described in Chapter 1. The empty weight will be marked on the 24PT1-DSC by the fabricator.

9.2.1.5 Maintenance of Thermal Monitoring System

During fuel storage, the overall system requires daily visual inspection of the ventilation inlet and outlet openings of the AHSM to ensure that no vent blockage has occurred. If the inspection shows blockage of the vents, they shall be cleared. The AHSM concrete temperature is monitored to ensure that no vent blockage has occurred, as required by Chapter 12.

The licensee is responsible for maintenance and calibration of the Thermal Monitoring System. The licensee is also responsible for system data collection.

9.2.2 Valves, Rupture Discs, and Gaskets on Confinement Vessel

The 24PT1-DSC is the confinement vessel for the SFAs. There are no valves, rupture discs, or gaskets on the 24PT1-DSC.

9.3 Training Program

All personnel working at the ISFSI receive training and indoctrination aimed at providing and maintaining a well-qualified work force for safe and efficient operation of the ISFSI. The licensee may utilize the existing plant training program to provide this training and indoctrination. Additional sections to the program are added to include information specific to the ISFSI.

9.3.1 Program Description

9.3.1.1 Training for Operations Personnel

Generalized training should be provided to plant operations personnel in the applicable regulations and standards and the engineering principles of passive cooling, radiological shielding, and structural characteristics of the ISFSI. Detailed operator training will be provided for 24PT1-DSC preparation and handling, fuel loading, transfer cask preparation and handling, and transfer trailer loading.

9.3.1.2 Training for Maintenance Personnel

Generalized training should be provided to plant maintenance personnel on the applicable regulations and standards and in the engineering principles of passive cooling, radiological shielding, and structural characteristics of the ISFSI. Specific training is provided for use of the vacuum drying system; the automated welding equipment; operation of the transfer trailer; alignment of the transfer cask skid with the AHSM; assembly of the hydraulic ram system; and normal and off-normal operation of the hydraulic ram. Specific training is also provided for cleaning the AHSM air inlet and outlet vents.

9.3.1.3 Training for Health Physics Personnel

Generalized training should be provided to plant health physics personnel on the applicable regulations and standards and in the engineering principles of passive cooling, radiological shielding, and structural characteristics of the ISFSI. Specific training should be provided in the radiological shielding design of the system, particularly the 24PT1-DSC top shield plug, the transfer cask and the AHSM.

9.3.1.4 Training for Security Personnel

Details of the training program for security personnel are provided in the Security Plan to be maintained by the licensee, which is to be withheld from public disclosure in accordance with 10CFR 2.790(d) and 10CFR 73.21.

9.3.2 Retraining Program

Retraining is generally consistent with the retraining requirements in effect at the plant for personnel involved in fuel handling operations.

9.3.3 Administration and Records

The licensee's plant training organization is the organization responsible for training programs and for maintaining up-to-date records on the status of personnel training.

9.4 Supplemental Information

9.4.1 References

[9.1] Electric Power Research Institute, "NUHOMS® Modular Spent-Fuel Storage System: Performance Testing," EPRI Report NP-6941, September 1990.

[9.2] ASME Boiler and Pressure Vessel Code, Section III, 1992 Edition with 1994 Addenda as modified by Code Case N-595-1.

[9.3] SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1984.

[9.4] ANSI N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," New York, 1996.

[9.5] ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials", February 1998.

[9.6] ASTM E94, "Guide for Radiographic Testing", 1993.

[9.7] ASTM E142, "Method for Controlling Quality of Radiographic Testing", 1992.

[9.8] ASTM E545, "Method for Determining Image Quality in Direct Thermal Neutron Radiographic Examination", 1991.

[9.9] Nuclear Regulatory Commission, Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, December 1994, USNRC Docket Number 72-1004, File NUH003.0103.02.

[9.10] Rancho Seco Nuclear Plant ISFSI *FSAR, Revision 0, November 2000, USNRC Docket Number 72-11.*

[9.11] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, ACI 349-97 and ACI 349R-97, American Concrete Institute, Detroit, Michigan.

[9.12] Hahn, G. J., Statistical Intervals for a Normal Population, Part I," Journal of Quality Technology, Vol. 2, No. 3, July 1970.

[9.13] Hahn, G. J., Statistical Intervals for a Normal Population, Part II," Journal of Quality Technology, Vol. 2, No. 4, October 1970.

[9.14] Owen, D. B., "A Survey of Properties and Applications of the Noncentral t-Distribution," Technometrics, Vol. 10, No. 3, August 1968.

[9.15] "SCR607, Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," U.S. Department of Energy, Sandia Corporation, March 1963.

10. RADIATION PROTECTION

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

To ensure that occupational radiation exposures are ALARA, two primary factors were considered: (1) minimizing occupational exposure during canister loading and (2) minimizing storage dose rates. Occupational exposure is minimized by shielding incorporated into the canister, transfer cask, and handling equipment, remote operation of welding and handling equipment, and proven procedures for performing fuel loading. Storage dose rates are minimized by five feet of concrete shielding present in the module roof, use of self shielding by placing modules directly adjacent to one another, and by facing the lowest dose rate side of the module arrays toward the limiting boundary of the facility, where possible.

10.1.1 Policy Considerations

The 24PT1-DSC, transfer cask and AHSM design incorporates various methods of shielding and design features to minimize occupational radiation exposures. The licensee's existing radiation safety and ALARA policies for the plant should be applied to the ISFSI. The ALARA program should follow the general guidelines of Regulatory Guides 1.8 [10.4], 8.8 [10.1], 8.10 [10.3] and 10CFR 20 [10.6]. ISFSI personnel should be trained in the proper operation of the Advanced NUHOMS® System and updated on ALARA practices and dose reduction techniques. This training includes operations, inspections, repair and maintenance. Proper training of personnel helps to minimize exposure to radiation such that the total individual and collective exposure to personnel in all phases of operation and maintenance are kept ALARA. Implementation of ISFSI systems and equipment procedures should be reviewed by the licensee to ensure ALARA exposure during all phases of operations, maintenance and surveillance.

10.1.2 Design Considerations

The Advanced NUHOMS® System takes into account radiation protection considerations, which ensure that occupational radiation exposures are ALARA. The fuel will be stored dry inside the sealed and heavily shielded 24PT1-DSC, and AHSM. Shield plugs at the ends of the 24PT1-DSC provide shielding for welding operations and during onsite 24PT1-DSC transfer (See 24PT1-DSC drawing in Section 1.5.2). Cask lead shielding and neutron shielding provide required shielding during transfer activities (See Chapter 5, Figure 5.1-4). The AHSM walls, roof and shield walls provide shielding during storage (See AHSM drawing in Section 1.5.2). The 24PT1-DSC will not be opened nor fuel removed while at the ISFSI, unless the ISFSI is specifically licensed for these purposes. Storage of the fuel in the dry, leaktight 24PT1-DSC eliminates the possibility of leakage of contaminated liquids, particulate materials, or radioactive gases. The exterior of the transfer cask is decontaminated prior to transfer to the ISFSI, thereby minimizing exposure of personnel to surface contamination. The 24PT1-DSC outside surface is also contamination free (clean surface) due to the use of inflatable seals in the annulus between the cask and 24PT1-DSC during loading operations. The Advanced NUHOMS® System contains no active components which require periodic maintenance or surveillance thereby minimizing potential personnel dose due to maintenance activities. This method of spent fuel storage minimizes radiation exposure and eliminates the potential for personnel contamination. The NUHOMS® design configuration has been demonstrated to provide appropriate design features for reduction of doses and for facilitating decontamination in over 100 similar systems loaded as of September 2000.

Regulatory Position 2 of Regulatory Guide 8.8 [10.1], is incorporated into the design considerations, as described below:

- Regulatory Position 2a on access control is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.
- Regulatory Position 2b on radiation shielding is met by the heavy shielding attributes of the Advanced NUHOMS® System which minimizes personnel exposures.
- Regulatory Position 2c on process instrumentation and controls is met by designing the instrumentation for a long service life and locating readouts in a low dose rate location. The use of thermocouples for temperature measurements located in embedded thermowells provides reliable, easily maintainable instrumentation for this monitoring function.
- Regulatory Position 2d on control of airborne contaminants may be applicable for vacuum drying operations of DSCs containing damaged fuel. Monitoring of the vacuum drying system discharge and diversion to the gaseous radwaste system or other appropriate filtration systems will be implemented. No significant surface contamination is expected because the exterior of the transfer cask is decontaminated prior to transfer to the ISFSI and the exterior of the DSC is also contamination free.
- Regulatory Position 2e on crud control is not applicable to the ISFSI because there are no systems at the ISFSI that could transport crud. The leaktight DSC design ensures that spent fuel crud will not be released or transferred from the DSC.
- Regulatory Position 2f on decontamination is met because the transfer cask is decontaminated prior to transfer to the ISFSI. The transfer cask accessible surfaces are designed to facilitate decontamination.
- Regulatory Position 2g on radiation monitoring does not apply since no leakage of radioactive material is possible. There is no need for airborne radioactivity monitoring because the 24PT1-DSCs are sealed and leaktight. Airborne radioactivity due to damaged fuel is discussed under Regulatory Position 2d above. Area radiation monitors are not required because the ISFSI will not be occupied on a regular basis.
- Regulatory Position 2h on resin and sludge treatment systems is not applicable to the ISFSI because there are no radioactive systems containing resins or sludge associated with the ISFSI.
- Regulatory Position 2i concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the ISFSI.

10.1.3 Operational Considerations

The operational requirements are incorporated into the radiation protection design features described in Section 10.2 since the Advanced NUHOMS® System is heavily shielded to minimize occupational exposure.

The 24PT1-DSCs contain no radioactive liquids and, for intact fuel assemblies, are not expected to contain any radioactive gases. Damaged assemblies will be loaded into failed fuel cans before being loaded into the 24PT1-DSC; this provides another barrier to the escape of any small fuel particles into the DSC volume. Finally, the 24PT1-DSC is designed and tested to be leaktight.

The Advanced NUHOMS® System is designed to be essentially maintenance free. It is a passive system without any moving parts.

The only anticipated maintenance procedures are the visual inspection of the bird screens on the AHSM ventilation inlet and outlet openings, and periodic maintenance of the thermocouples. Maintenance operations on the transfer cask, transfer equipment and other auxiliary equipment is performed in a low dose environment during periods when fuel movement is not occurring. Maintenance activities that could involve significant radiation exposure of personnel should be carefully planned.

The ISFSI contains no systems that process liquids or gases or contain, collect, store, or transport radioactive liquids or solids other than payloads identified in Chapter 2. Therefore, the ISFSI meets ALARA requirements since there are no systems to be maintained or repaired other than those systems previously discussed.

10.2 Radiation Protection Design Features

10.2.1 Advanced NUHOMS® System Design Features

The Advanced NUHOMS® System has design features which ensure a high degree of integrity for the confinement of radioactive materials and reduction of direct radiation exposures to ALARA. These features are described below:

- The 24PT1-DSCs are loaded, sealed and leak tested prior to transfer to the ISFSI.
- The fuel will not be unloaded nor will the 24PT1-DSCs be opened at the ISFSI unless the ISFSI is specifically licensed for these purposes.
- The fuel is stored in a dry inert environment inside the 24PT1-DSCs so that no radioactive liquid is available for leakage.
- The 24PT1-DSCs are sealed and tested leaktight with a helium atmosphere to prevent oxidation of the fuel. The leaktight design features are described in Chapter 7.
- The 24PT1-DSCs are heavily shielded to reduce external dose rates. The shielding design features are discussed in Chapter 5.
- No radioactive material will be discharged during storage since the 24PT1-DSC is designed, fabricated and tested to be leaktight.

Geometric attenuation, enhanced by air and ground dispersion, provides additional shielding for distant locations at restricted area and site boundaries. However, the contribution of the sky shine dose rate must be considered for distant locations. The total dose rate estimation, including sky shine, is provided in the following section.

10.2.2 Radiation Dose Rates

Calculated dose rates in the immediate vicinity of the Advanced NUHOMS® System are presented in Chapter 5 which provides a detailed description of source term configuration, analysis model and expected dose rates. Dose rates for longer distances (off site doses) are presented in this section for the design basis fuel load with design basis control components.

The monte-carlo computer code MCNP [10.2] is used to calculate the dose rates at the required locations around the AHSM.

The assumptions used to generate the geometry of the AHSM and shield walls for the MCNP runs are summarized below.

- A single AHSM is modeled as a box enveloping the AHSM and 3 foot shield walls on the back and two sides. Source particles are then started on the surfaces of the box. A discussion of the source assumptions is provided below.

- The AHSM approach slab is modeled as a concrete slab, approximately 108 feet by 84 feet by 3 feet thick. The remaining volume below ground level is modeled as soil.
- AHSM interiors are filled with air. Most particles that enter the AHSM will therefore pass through unhindered.
- The "universe" is a sphere surrounding the AHSM. The radius of this sphere is more than 10 mean free paths (gamma) greater than that of the outermost detector.

The assumptions used to generate the AHSM surface sources for the MCNP runs are summarized below.

- The AHSM surface sources are generated using the AHSM surface dose rates calculated in Chapter 5.
- The AHSM is assumed to be filled with a canister containing 24 design basis fuel assemblies and 24 design basis thimble plugs.

The assumptions used for the MCNP computer runs are summarized below:

- Source particles are generated on the AHSM with initial directions following a cosine distribution. Radiation fluxes outside thick shields such as the AHSM walls and roof tend to have forward peaked angular distributions that are reasonably approximated by a cosine function. Vents through shielding regions such as the AHSM vents tend to collimate particles such that a semi-isotropic assumption is not appropriate.
- Point detectors are used for all of the dose rates on the four sides of the AHSM. All detectors have been placed three feet above ground level.

Source information required by MCNP includes gamma-ray and neutron spectra for the AHSM, total gamma-ray and neutron activities for each AHSM face and total gamma-ray and neutron activities for the entire AHSM. The neutron and gamma-ray spectra are determined using a 1-D ANISN run through the AHSM roof using the "in-core" design basis fuel source from Chapter 5. Table 10.2-9 provides the material and region thicknesses used in the ANISN model. The AHSM spectra as determined from ANISN are normalized to a one mrem/hour source using the flux-to-dose-factors from Chapter 5. These normalized spectra are then input in the MCNP ERG source variable.

The probability of a particle being born on a given surface is proportional to the total activity of that surface. The activity of each surface is determined by multiplying the sum of the normalized group fluxes, calculated above, by the average surface dose rate and by the area of the surface. This calculation is performed for the roof, sides and front of the AHSM. The sum of the surface activities is then input as the tally multiplier for each of the MCNP tallies to convert the tally results to fluxes (particles per second per square centimeter).

Gamma-ray spectrum calculations for the AHSM are shown in Table 10.2-6. The group fluxes on the AHSM roof are taken from the ANISN run. The dose rate contribution from each group is the product of the flux and the flux-to-dose factor. The "Input Flux" column in Table 10.2-6 is simply the roof flux in each group, divided by the total dose rate and represents the roof flux normalized to one mrem per hour. The total flux for a one mrem/hr average surface dose rate is then 1.09×10^3 $\gamma/cm^2 \cdot s$. Similar calculations for neutrons are shown in Table 10.2-7. The total neutron flux for a one mrem/hr average surface dose rate is 7.97×10^1 $n/cm^2 \cdot s$.

The AHSM modeled in MCNP is approximated by a box that envelops the individual AHSM and shield walls. The dimensions of the box also include the width of the AHSM end and back shield walls. As is discussed above, the total activity of each face of the box is calculated by multiplying the flux per mrem/hr by the average dose rate of the face and by the area of the face. The dimensions of an AHSM are:

AHSM Width..101 inches

Shield Wall Thickness..............................36 inches

AHSM Height..247 inches

Depth (Front-to-Back)..............................235 inches

The source area of the front and back is,

$$A_{front/Back} = [(247)(101+2(36))](2.54)^2 \\ = 275{,}683.3cm^2 \tag{1}$$

The total gamma activity for the front is 5.68×10^8 γ/s and the total neutron activity is 8.79×10^5 n/s. The total gamma activity for the back is 1.22×10^6 γ/s and the total neutron activity is 8.13×10^3 n/s.

The source area of the roof is,

$$A_{roof} = [(235+36)(101+2(36))](2.54)^2 \\ = 302{,}470.4cm^2 \tag{2}$$

The total gamma activity for the array roof is 9.89×10^6γ/s and the total neutron activity is 2.06×10^4 n/s.

The source area of the side is,

$$A_{side} = [(235+36)(247)](2.54)^2 \\ = 431{,}850.7cm^2 \tag{3}$$

The total gamma activity for each side is 1.22×10^8 γ/s and the total neutron activity is 3.44×10^5 n/s.

The AHSM surface activities are summarized in Table 10.2-8.

24PT1-DSC dose rates were calculated for distances of 6.1 meters (20 feet) to 500 meters from a single AHSM at the front, side and back of the AHSM. The results of the single AHSM analyses (with 3-foot shield walls on sides and back of module, as described in Chapter 1) are presented in Table 10.2-1, Table 10.2-2 and Table 10.2-3. The total annual dose (direct + sky shine) as a function of distance from the surfaces of the AHSM for a single AHSM is shown in Figure 10.2-1.

10.2.3 AHSM Dose Rates

A representative array of 2x10 modules (back to back), is also considered. Doses for this array are conservatively extrapolated from the single AHSM data as follows:

Front Dose Rate for 2x10 (either side) = [(Front Dose Rate for 1 AHSM) x 10] +

[(Back Dose Rate for 1 AHSM) x 10].

Side Dose Rate for 2x10 = [(Side Dose Rate for 1 AHSM) x 20].

Table 10.2-4, Table 10.2-5 and Figure 10.2-2 provide dose rates for the 2x10 array. The front dose estimate ignores the effect of attenuation due to the indirect scatter of radiation as measured from all but 9 of the AHSMs in the array (the 1 AHSM dose rate is taken in front of the AHSM vent, actual dose rate for a 2 x 10 array at any point in front of the array is similar to the one AHSM dose with the remaining 9 AHSM doses attenuated due to longer distance and indirect angle from the front of the other AHSMs to the dose point). The use of the back dose from the one AHSM case to estimate dose from AHSMs facing away from the dose point, is also conservative since it neglects additional attenuation due to distance and shielding provided by the front AHSM. The dose rate for the side of a 2 x 10 array conservatively assumes that the total of 20 AHSM side doses is appropriate ignoring attenuation due to distance and geometry.

For a single AHSM containing design basis fuel and NFAH, a minimum distance of approximately 80 meters is necessary to meet the 10CFR 72.104 [10.5] limits, assuming an exposure of 8,760 hours per year from the front of the AHSM (AHSM doses are highest at the front of the AHSM due to radiation through the air inlet opening). For a 2 x 10 array without any site specific shielding, a distance of approximately 200 meters from the front and back of the AHSMs is required to ensure doses are less than the 10CFR 72.104 limits. A distance of approximately 80 meters is required to meet these limits from the side of the single AHSM (extrapolated from Figure 10.2-1). A distance of approximately 200 meters is required to meet these limits from the side of a 2 x 10 array of AHSMs (extrapolated from Figure 10.2-2).

10.2.4 ISFSI Array

The dose rates from a typical ISFSI are evaluated by the licensee in a 10CFR 72.212 evaluation to address the site-specific ISFSI layout and its time phased installation.

Dose rates at the site boundary will depend on specific ISFSI parameters such as storage array configuration, number of stored 24PT1-DSCs, characteristics of stored fuel, fuel loading patterns, site geography, etc. Berms, walls, removable shields or preferential loading of "cooler" fuel in the outer cells of the 24PT1-DSC may be used as necessary to keep the site boundary dose rate within the 10CFR 72.104 limits. Shields located within ten feet of the perimeter of the ISFSI modules or attached to the modules must be analyzed to confirm that they do not adversely impact the design basis of the AHSM. Shields attached to the AHSM must be evaluated for their potential impact on all normal, off-normal and accident scenarios to ensure that they do not introduce an unreviewed safety question as part of the site analysis performed as required by 10CFR 72.104 and 10CFR 72.212.

Table 10.2-1
Dose Rates at Postulated Site Boundary from One AHSM
For 24 Design Basis Fuel Assemblies and Control Components

(based on distance from the front of a single AHSM)

Distance from Source*	Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)	Total (mrem/yr)
6 1 meters (20 feet)	9.99E-03	3.66E-01	3.76E-01	3.30E+03
100 meters	3.60E-05	1.23E-03	1.27E-03	1.11E+01
200 meters	5.02E-06	1.92E-04	1.97E-04	1.73E+00
300 meters	1.59E-06	5.39E-05	5.55E-05	4.86E-01
500 meters	2.00E-07	6.36E-06	6.57E-06	5.80E-02

* Distance from center of front face of AHSM

Table 10.2-2
Dose Rates at Postulated Site Boundary from One AHSM

For 24 Design Basis Fuel Assemblies and Control Components

(based on distance from the back of the single AHSM)

Distance from Source*	Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)	Total (mrem/yr)
6.1 meters (20 feet)	5.86E-04	4.10E-03	4.69E-03	4.10E+01
100 meters	1.09E-05	1.19E-04	1.30E-04	1.14E+00
200 meters	1.98E-06	2.21E-05	2.41E-05	2.11E-01
300 meters	6.55E-07	4.77E-06	5.43-06	4.76E-02
500 meters	9.39E-08	5.31E-07	6.30E-07	5.53E-03

* Distance from center of back face of AHSM

Table 10.2-3
Dose Rates at Postulated Site Boundary from a Single AHSM

For 24 Design Basis Fuel Assemblies and Control Components

(based on distance from the side of a single AHSM)

Distance from Source*	Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)	Total (mrem/yr)
6.1 meters (20 feet)	4.05E-03	7.77E-02	8.18E-02	7.16E+02
100 meters	2.41E-05	4.04E-04	4.28E-04	3.75E+00
200 meters	3 53E-06	6.93E-05	7.29E-05	6.38E+01
300 meters	1.05E-06	1.71E-05	1.82E-05	1.59E-01
500 meters	1.54E-07	2.04E-06	2.19E-06	1.90E-02

* Distance from side face of AHSM

Table 10.2-4
Dose Rates at Postulated Site Boundary from a
2x10 Array Of AHSMs

For 24 Design Basis Fuel Assemblies and Control Components

(based on distance from the front or back of a 2 X 10 array of AHSMs)

Distance from Source*	Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)	Total (mrem/yr)
6.1 meters (20 feet)	1.06E-01	3.72E-00	3.83E-00	3.30E+04
100 meters	4 69E-04	1.35E-02	1.40E-02	1.22E+02
200 meters	7.00E-05	2.14E-03	2.21E-03	1.94E+01
300 meters	2.25E-05	5.87E-04	6 09E-04	5.34E+00
500 meters	1.97E-06	6.89E-05	7.09E-05	6.21E-01

* Distance from center of front face of AHSMs

Table 10.2-5
Dose Rates at Postulated Site Boundary from a 2x10 Array Of AHSMs
For 24 Design Basis Fuel Assemblies and Control Components

(based on distance from the side of a 2 X 10 array of AHSMs)

Distance from Source*	Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)	Total (mrem/yr)
6.1 meters (20 feet)	8.10E-02	1.55E-00	1.63E-00	1.43E+04
100 meters	4.82E-04	8.08E-03	8.56E-03	7.50E+01
200 meters	7.06E-05	1.39E-03	1.45E-03	1.28E+01
300 meters	2.10E-05	3.42E-04	3.64E-05	3.18E-00
500 meters	3.08E-06	4.08E-05	4.38E-05	3 80E-01

* Distance from side face of AHSM

Table 10.2-6
AHSM Gamma-Ray Spectrum Calculation Results

Group Number	E_{upper} (MeV)	E_{mean} (MeV)	Flux-Dose ANSI/ANS-6 1.1-1977 (mR/hr)/(γ/cm²-sec)	Roof Flux (γ/cm²-sec)	Dose Rate (mR/hr)	Input Flux (γ/cm²-sec)
23	10	9	8.772E-03	1.43E-02	1.25E-04	2.72E-01
24	8	7.25	7.479E-03	7.15E-02	5 35E-04	1.37E+00
25	6.5	5.75	6 375E-03	1.07E-01	6 81E-04	2.04E+00
26	5	4.5	5 414E-03	1.08E-01	5 83E-04	2.06E+00
27	4	3.5	4.622E-03	1.37E-01	6 33E-04	2.62E+00
28	3	2.75	3 960E-03	8 26E-02	3 27E-04	1.58E+00
29	2.5	2.25	3 469E-03	1.51E-01	5 25E-04	2.89E+00
30	2	1.83	3 019E-03	1.26E-01	3 80E-04	2.40E+00
31	1.66	1.495	2 628E-03	1.58E+00	4.14E-03	3.01E+01
32	1.33	1 165	2.205E-03	3 30E+00	7.28E-03	6.31E+01
33	1	0.9	1.833E-03	2.91E+00	5.33E-03	5.56E+01
34	0 8	0 7	1.523E-03	4 12E+00	6 27E-03	7.87E+01
35	0.6	0.5	1.173E-03	6 41E+00	7.52E-03	1.23E+02
36	0 4	0.35	8.759E-04	4.70E+00	4.12E-03	8.98E+01
37	0 3	0.25	6.306E-04	7.45E+00	4.70E-03	1.42E+02
38	0 2	0.15	3 834E-04	1.90E+01	7.29E-03	3.63E+02
39	0.1	0.08	2.669E-04	6 93E+00	1.85E-03	1.32E+02
40	0 05	0.03	9 348E-04	3 85E-02	3 60E-05	7.36E-01
			Totals	5.73E+01	5 23E-02	1.09E+03

Table 10.2-7
AHSM Neutron Spectrum Calculations

Group Number	E_{upper} (MeV)	E_{mean} (MeV)	Flux-Dose ANSI/ANS-6.1.1-1977 (mR/hr)/(n/cm^2-sec)	Roof Flux (n/cm^2-sec)	Dose Rate (mR/hr)	Input Flux (n/cm^2-sec)
1	1.49E+01	1.36E+01	1.945E-01	6.54E-07	1.27E-07	3.74E-04
2	1.22E+01	1.11E+01	1.597E-01	3 14E-06	5.02E-07	1.79E-03
3	1.00E+01	9.09E+00	1.471E-01	1.48E-05	2.18E-06	8.46E-03
4	8.18E+00	7.27E+00	1.477E-01	1.30E-04	1.92E-05	7.44E-02
5	6.36E+00	5.66E+00	1.534E-01	3.27E-04	5 02E-05	1.87E-01
6	4.96E+00	4.51E+00	1.506E-01	2.90E-04	4.37E-05	1.66E-01
7	4 06E+00	3.54E+00	1.389E-01	3.47E-04	4.83E-05	1.98E-01
8	3.01E+00	2.74E+00	1.284E-01	6.66E-04	8.55E-05	3.80E-01
9	2.46E+00	2.41E+00	1.253E-01	6.42E-04	8 05E-05	3 67E-01
10	2.35E+00	2.09E+00	1.263E-01	1.16E-03	1.46E-04	6.60E-01
11	1.83E+00	1.47E+00	1.289E-01	1.95E-03	2.51E-04	1.11E+00
12	1.11E+00	8.30E-01	1.169E-01	2.28E-03	2.67E-04	1.30E+00
13	5.50E-01	3.31E-01	6.521E-02	3.72E-03	2.43E-04	2.13E+00
14	1.11E-01	5.72E-02	9.188E-03	5.37E-03	4.93E-05	3.07E+00
15	3.35E-03	1.97E-03	3.713E-03	2.57E-03	9.53E-06	1.47E+00
16	5 83E-04	3 42E-04	4.009E-03	3.07E-03	1.23E-05	1.75E+00
17	1.01E-04	6 50E-05	4.295E-03	2.54E-03	1.09E-05	1.45E+00
18	2.90E-05	1.96E-05	4 476E-03	1.80E-03	8.08E-06	1.03E+00
19	1.01E-05	6 58E-06	4.567E-03	2.43E-03	1.11E-05	1.39E+00
20	3 06E-06	2.09E-06	4.536E-03	2.14E-03	9.72E-06	1.22E+00
21	1.12E-06	7.67E-07	4.370E-03	2.24E-03	9.79E-06	1.28E+00
22	4.14E-07	2.12E-07	3.714E-03	1.06E-01	3 93E-04	6 05E+01
			Totals	1.40E-01	1.75E-03	7.97E+01

Table 10.2-8
Summary of AHSM Surface Activities

Source	Area (cm^2)	Gamma-Ray Activity (γ/sec)	Neutron Activity (neutrons/sec)
Roof	302,470.4	9 89E+06	2.06E+04
Front	275,683.3	5.68E+08	8.79E+05
Back	275,683.3	1.22E+06	8.13E+03
Side 1	431,850.7	1.22E+08	3.44E+05
Side 2	431,850.7	1.22E+08	3 44E+05
Total		8.24E+08	1.60E+06

Table 10.2-9
ANISN Model Details

Region	Material	Radius (cm)	Thickness (cm)
Fuel	In-Core Fuel	67.57	67.57
Gap	Air	83.74	16.17
Canister Wall	Stainless Steel	85.33	1.59
Gap	Air	156.58	71.25
Roof	Concrete	304.8	148.22

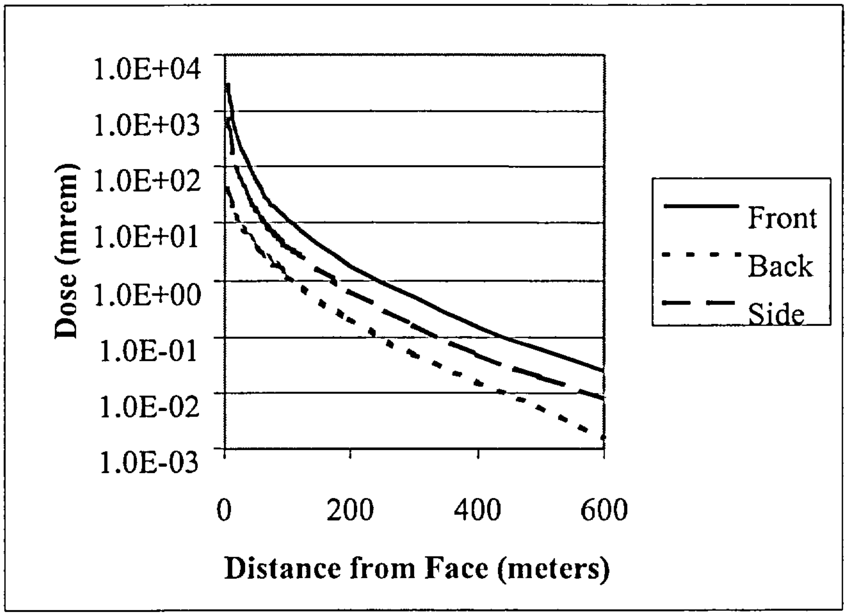


Figure 10.2-1
<u>Total Annual Exposure from a Single AHSM as a Function of Distance</u>

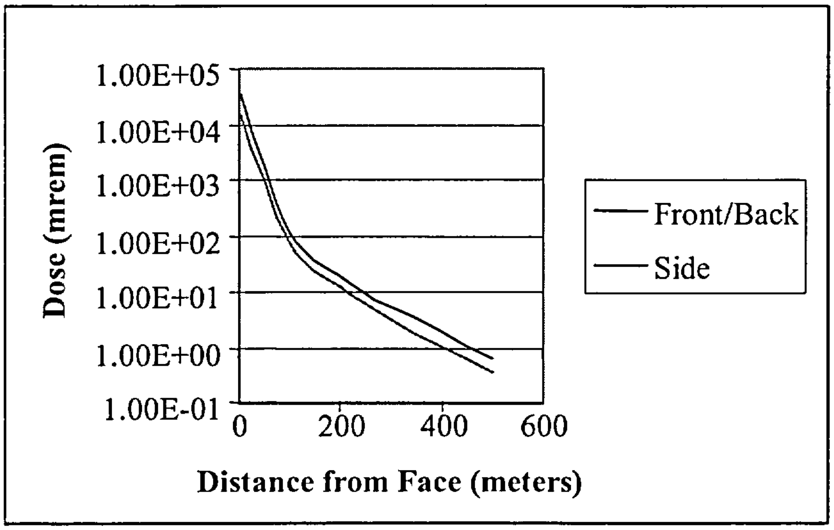


Figure 10.2-2
Total Annual Exposure from a 2x10 AHSM Array as a Function of Distance

10.3 Estimated Onsite Collective Dose Assessment

This section provides estimates of occupational and offsite doses for typical ISFSI configurations.

Assumed annual occupancy times, including the anticipated maximum total hours per year for any individual and total person-hours per year for all personnel for each radiation area during normal operation and anticipated operational occurrences will be evaluated by the licensee in a 10CFR 72.212 evaluation to address the site specific ISFSI layout, inspection, and maintenance requirements. In addition, the estimated annual collective person rem doses associated with loading operations will be addressed by the licensee in a 10CFR 72.212 evaluation.

10.3.1 Occupational Exposures

10.3.1.1 24PT1-DSC Loading, Transfer and Storage Operations

Table 10.3-1 shows the estimated occupational exposures to ISFSI personnel during loading, transfer, and storage of the 24PT1-DSC (time and manpower may vary depending on individual ISFSI practices). The task times, number of personnel required and total doses are listed in this table. These estimates are based on actual NUHOMS® system operating experience. Temporary shielding can be used by the licensee to maintain doses ALARA.

The average distance for a given operation takes into account that the operator may be in contact with the transfer cask, but this duration will be limited. For draining activities, vacuum drying, and leak testing, the attachment of fittings will take place closer to the cask than the operation of the pump and vacuum drying system. For decontamination activities, although operators could be near the cask for some activities, other parts of the operation could be performed from farther away. For this reason, 1 foot or 3 feet is an appropriate average distance for these operations.

The operator's hands may be in a high dose rate location momentarily, for example when connecting couplings or vacuum fittings at the ports. This does not translate into a whole-body dose, and therefore, these localized streaming effects are not considered here.

For operations near the top end of the 24PT1-DSC, most of the work will take place around the perimeter (top edge of DSC/Transfer cask) and a smaller portion will take place directly over the shield plug.

The areas of highest operational dose (potential streaming paths) are the front of a loaded AHSM at the air inlet vent, at the cask side surface with a dry DSC (outer cover plate welding, transfer operations) and at the cask/DSC annulus. Operating procedures and personnel training minimize personnel exposure in these areas.

The guidance of Reg. Guide 8.34 [10.7] is to be employed in defining the on-site occupational dose and monitoring requirements.

10.3.1.2 24PT1-DSC Retrieval Operations

Occupational exposures to ISFSI personnel during 24PT1-DSC retrieval are similar to those exposures calculated for 24PT1-DSC insertion. Dose rates for retrieval operations will be lower than those for insertion operations due to radioactive decay of the spent fuel inside the AHSM. Therefore, the dose rates for 24PT1-DSC retrieval are bounded by the dose rates calculated for insertion.

10.3.1.3 24PT1-DSC Fuel Unloading Operations

The process of unloading the 24PT1-DSC is similar to that used for loading the 24PT1-DSC. The identical ALARA procedures utilized for loading should also be applied to unloading.

Occupational exposures to plant personnel are bounded by those exposures calculated for 24PT1-DSC loading.

10.3.1.4 Maintenance Operations

The dose rate for surveillance activities is obtained from Table 10.2-1, Table 10.2-2 and Table 10.2-3 for AHSM 20 foot doses rates at the front of an AHSM. The 20 foot dose rate is a conservative estimate for surveillance activities. The AHSM surface dose rate provided in Chapter 5 is a conservative estimate for thermocouple maintenance activities including calibration and repair. The surface dose rate calculated in Chapter 5 also provides a conservative estimate of a dose rate at 3 ft. from the AHSM which may be encountered during operations associated with removal of debris from AHSM vents.

The ISFSI license applicant will evaluate the additional dose to station personnel from ISFSI operations, based on the particular storage configuration and site personnel requirements.

10.3.1.5 Doses During ISFSI Array Expansion

ISFSI expansion should be planned to eliminate the need for entry into a module adjacent to a loaded module. The reduction in shielding between the side of an array with an installed shield wall (4-feet of concrete, consisting of 1-foot side wall and 3- foot shield wall) versus shielding between the inside of an empty module and an adjacent loaded module (2-feet of concrete) is very significant. Pre-planning to limit entry into a module when it is not separated from a loaded module by at least one empty module (4-feet of concrete) is recommended. *Similarly, during array expansion, when the shield wall is removed, personnel access to the area should be controlled.* For a module separated from a loaded AHSM by an empty module, the resulting dose will be less than that specified for the side dose rate of an array with an installed shield wall. See Chapter 5 for estimated AHSM side dose rates for this operation.

10.3.2 Public Exposure

The only off-site dose to the public from the ISFSI is from direct and skyshine radiation at or beyond the controlled area of the ISFSI (as defined by 10CFR 72.106). Figure 10.2-1 and Table

10.2-1, Table 10.2-2 and Table 10.2-3 show the radiation dose rates in the vicinity of a single AHSM. Dose rates in the vicinity of a 2 x 10 array are provided in Figure 10.2-2 and Table 10.2-4 and Table 10.2-5. The collective off-site dose is a function of the number and arrangement of the AHSMs on the ISFSI, the proximity of the ISFSI to the site boundary and other plant considerations to be addressed by the licensee in accordance with 10CFR 72.212.

Each cask user or general licensee must perform a site-specific analysis as required by 10CFR 72.212(b) to demonstrate compliance with 10CFR 72.104(a) for normal operations and anticipated occurrences. The general licensee may consider site-specific conditions, such as actual distances to the nearest real person, topography, array configurations, characteristics of stored fuel, and use of engineered features, such as berms, walls or additional shield blocks, in their analysis of public doses. The site-specific analysis must also include the doses received from other fuel cycle activities (e.g., reactor operations) in the region.

Table 10.3-1
Advanced NUHOMS® System Operations Estimated Time for Occupational Dose Calculations

(for information only)

	Number of Workers	Completion Time (hours)	Dose (person-mrem)
LOCATION: AUXILIARY BUILDING AND FUEL POOL			
Prepare the 24PT1-DSC and Transfer Cask for Service	2	4.0	0
Place the 24PT1-DSC into the Transfer Cask	3	1.0	6
Fill the Cask/24PT1-DSC Annulus with Uncontaminated Water and Seal	2	2.0	7
Fill the 24PT1-DSC Cavity with borated Water	1	0.5	1
Place the Cask Containing the 24PT1-DSC in the Fuel Pool	5	1.0	10
Verify and Load the Candidate Fuel Assemblies into the 24PT1-DSC	3	8.0	48
Place the Top Shield Plug on the 24PT1-DSC and place the cask/DSC in the Decon Area	5	2 0	20
LOCATION: CASK DECON AREA			
Decontaminate the Outer Surface of the Cask	7	1.0	327
Drain Water Above Shield Plug	3	0 25	35
Set-Up Welding Machine	2	3.3	46
Weld the Inner Top Cover Plate to the Shell and Perform NDE (PT)	3	6 0	66
Prepare VDS for Removal of Water from the 24PT1-DSC Cavity	1	0.02	1
Operate the VDS and remove water	1	0.5	1
Vacuum Dry and Backfill the 24PT1-DSC with Helium	2	2.0	8
Seal Weld the Pre-fabricated Plugs to the Vent and Siphon Port and Perform NDE (PT)	2	1.5	932
Prelim. Helium Leak Test the inner top cover plate Weld	2	1.0	4
Fit-up the Outer Top Cover Plate	2	2.0	159
Weld the Outer Top Cover Plate to the Shell and Perform NDE (PT)	7	17.5	356

Table 10.3-1
Advanced NUHOMS® System Operations Estimated Time for Occupational Dose Calculations

(for information only)

(concluded)

	Number of Workers	Completion Time (hours)	Dose (person-mrem)
LOCATION: REACTOR /FUEL BUILDING BAY			
Helium Leak Test the inner top cover plate Weld	2	1.0	4
Weld helium leak test vent port plug and NDE weld	2	0.75	466
Drain the Cask/24PT1-DSC Annulus	2	0 25	7
Install the Transfer Cask Top Cover Plate	2	1 0	30
LOCATION: REACTOR /FUEL BUILDING BAY			
Place the Cask Onto the Skid and Trailer	2	0 5	117
Secure the Cask to the Skid	2	0.25	123
LOCATION: ISFSI SITE			
Remove the Cask Top Cover Plate	2	0.5	7
Align and Dock the Cask with the AHSM	4	0.5	57
Insert DSC into AHSM	4	0.5	38
Lift the Ram Back onto the Trailer and Un-Dock the Cask from the AHSM	2	0.25	23
Install the AHSM Door	2	0.5	2
Adjust DSC seismic restraint	2	1.1	216
Total	**N/A**	**60.7**	**3118**

10.4 Supplemental Information

10.4.1 References

[10.1] U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, Information Relevant to Ensuring That Occupational Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable, Revision 3, June 1978.

[10.2] MCNP4B2, "Monte Carlo N-Particle Transport Code System." Los Alamos National Laboratory, CCC-660, RSIC.

[10.3] U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures as low as is reasonably Achievable, Revision 1-R, May 1977.

[10.4] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.8, Qualification and Training of Personnel for Nuclear Power Plants, Revision 2, April 1987.

[10.5] Title 10 Code of Federal Regulations Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[10.6] Title 10 Code of Federal Regulations Part 20, Standards for Protection Against Radiation.

[10.7] U.S. Nuclear Regulatory Commission, Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, July 1992.

11. ACCIDENT ANALYSES

This Chapter describes the postulated off-normal and accident events that might occur during storage of the 24PT1-DSC in an AHSM at an ISFSI. In addition, this chapter also addresses the potential causes of these events, their detection and consequences, and the corrective course of action to be taken by ISFSI personnel. Accident analyses demonstrate that the functional integrity of the system is maintained by:

1. Maintaining sub-criticality within margins defined in Chapter 6.
2. Maintaining confinement boundary integrity
3. Ensuring fuel retrievability and
4. Maintaining doses within 10CFR 72.106 limits (<5 rem).

The Accident Dose Calculations sections report the expected doses resulting from the postulated event in terms of whole body doses only. The leaktight canister design and the maintenance of confinement boundary integrity under all credible off-normal and accident scenarios ensures no radiation leakage from the 24PTl-DSC, thereby limiting dose consequences to direct and scattered radiation doses without any associated inhalation or ingestion doses.

11.1 Off-Normal Operations

Off-normal operations are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9 [11.1]. Design Event II conditions consist of that set of events that, although not occurring regularly, can be expected to occur with moderate frequency, or on the order of once during a calendar year of ISFSI operation.

For the Advanced NUHOMS® System, off-normal events could occur during fuel loading, trailer towing, 24PT1-DSC transfer and other operational events. The two off-normal events, which bound the range of off-normal conditions, are:

1. A "jammed" 24PT1-DSC during loading or unloading from the AHSM

2. The extreme ambient temperatures of -40 °F (winter) and +117 °F (summer)

These two events envelope the range of expected off-normal structural loads and temperatures acting on the Advanced NUHOMS® System.

11.1.1 Off-Normal Transfer Loads

Although unlikely, the postulated off-normal handling event assumes that the leading edge of the 24PT1-DSC becomes jammed against some element of the support structure during transfer between the transfer cask and the AHSM.

11.1.1.1 Postulated Cause of the Event

It is postulated that if the transfer cask is not accurately aligned with respect to the AHSM, the 24PT1-DSC could bind or jam during transfer operations.

The interiors of the transfer cask and the AHSM are inspected prior to transfer operations to ensure there are no obstacles, and the 24PT1-DSC has beveled lead-ins on each end, designed to avoid binding or sticking on small (less than 1/4 inch) obstacles. The transfer cask and the 24PT1-DSC support rails inside the AHSM are also designed with lead-ins to minimize binding or obstruction during 24PT1-DSC transfer. The postulated off-normal handling load event assumes that the leading edge of the 24PT1-DSC becomes jammed against some element of the support structure because of gross misalignment of the transfer cask.

The interfacing dimensions of the top end of the transfer cask and the AHSM access opening sleeve are specified such that docking of the transfer cask with the AHSM is not possible should gross misalignments between the transfer cask and AHSM exist.

11.1.1.2 Detection of the Event

If the 24PT1-DSC were to jam or bind during transfer, the hydraulic pressure in the ram would increase. The maximum ram push/pull forces are limited by design features to a maximum load equal to 80 kips. Override controls are available to the operator to increase the ram force up to its maximum design load, equal to 80 kips, or to interrupt the transfer operation at any time.

During the transfer operation, the force exerted on the 24PT1-DSC by the hydraulic ram is that required to first overcome the static frictional resisting force between the transfer cask rails and the 24PT1-DSC. Once the 24PT1-DSC begins to slide, the resisting force is a function of the sliding friction coefficient between the 24PT1-DSC and the transfer cask rails and/or between the 24PT1-DSC and the AHSM support rails. If motion is prevented, the hydraulic pressure increases, thereby increasing the force on the 24PT1-DSC until the hydraulic ram system pressure limit is reached. This limit is controlled so that adequate force is available to overcome variations in surface finish, etc., but is sufficiently low to ensure that component damage does not occur.

To overcome potentially higher resistance loads due to sticking of the 24PT1-DSC in either the transfer cask or the AHSM, the maximum ram force is designed to be equal to the weight of the loaded DSC. This force corresponds to a coefficient of friction equal to 1.0, and is the design basis for the hydraulic ram system.

11.1.1.3 Analysis of Effects and Consequences

The 24PT1-DSC and the AHSM are designed and analyzed for off-normal transfer loads of 80 kips (maximum force that the ram is able to develop), during insertion (loading) and 60 kips during retrieval (unloading) operations. These analyses are discussed in Chapter 3.

For either loading or unloading of the 24PT1-DSC under off-normal conditions, the stresses on the shell assembly components are demonstrated to be within the ASME Service Level B allowable stress limits. Therefore, permanent deformation of the 24PT1-DSC shell components does not occur. In addition, the loads are applied to the outer bottom cover plate, which is not part of the confinement boundary. The internal basket assembly components are unaffected by these loads based or clearances provided between support rods and 24PT1-DSC internal envelope.

There is no breach of the confinement pressure boundary and, therefore, no potential for release of radioactive material exists.

11.1.1.4 Corrective Actions

The required corrective action is to reverse the direction of the force being applied to the 24PT1-DSC by the ram, and return the 24PT1-DSC to its previous position. Since no permanent deformation of the 24PT1-DSC occurs, the sliding transfer of the 24PT1-DSC to its previous position is unimpeded. The transfer cask alignment is then rechecked, and the transfer cask repositioned as necessary before attempts at transfer are renewed.

11.1.2 Extreme Ambient Temperatures

The Advanced NUHOMS® System is designed for use at reactor sites within the continental United States. Therefore, conservatively, off-normal ambient temperatures of –40°F (extreme winter) and 117°F (extreme summer) are chosen. Each licensee must verify that this range of ambient temperatures envelops the design basis ambient temperatures for their ISFSI site.

The Advanced NUHOMS® System components affected by the postulated extreme ambient temperatures are the 24PT1-DSC during transfer from the plant's fuel building to the ISFSI site and during storage in the AHSM, and the AHSM.

11.1.2.1 Postulated Cause of the Event

Off-normal ambient temperatures are natural phenomena.

11.1.2.2 Detection of Event

Off-normal ambient temperature conditions will be confirmed by the licensee to be bounding for their site.

11.1.2.3 Analysis of Effects and Consequences

Thermal analysis of the Advanced NUHOMS® System for extreme ambient conditions is presented in Chapter 4. The effects of extreme ambient temperatures on the Advanced NUHOMS® System are discussed in Chapter 3.

11.1.2.4 Corrective Actions

Install transfer cask solar shield if the ambient temperature exceeds 100°F as required in Chapter 12. As shown in the analyses described in Chapters 3 and 4, the extreme ambient temperatures analyzed do not adversely impact operation of the Advanced NUHOMS® System.

11.1.3 Radiological Impact from Off-Normal Operations

For loading and unloading operations under off-normal conditions, the stresses on the 24PT1-DSC shell assembly components are demonstrated to be within the ASME Code Service Level B stress limits. Therefore, there is no permanent deformation of the shell. There is no potential for breach of the confinement pressure boundary and therefore, no potential for release of radioactive material.

The 24PT1-DSC shell assembly stresses due to extreme ambient temperature conditions are demonstrated to be less than the ASME Code Service Level B stress limits as shown in Chapter 3. The AHSM design considers stresses due to extreme ambient temperature conditions and meets the provisions of the ACI Code. Therefore, no damage will occur in the shell assembly or the AHSM. There is no potential for breach of the confinement pressure boundary and therefore, no potential for release of radioactive material.

11.2 Postulated Accidents

The design basis accident events specified by ANSI/ANS 57.9-1984 [11.1], and other postulated accidents that may affect the normal safe operation of the Advanced NUHOMS® System are addressed in this section. Analyses are provided for a range of hypothetical accidents, including those with the potential to result in an annual dose greater than 25 mrem outside the controlled area in accordance with 10CFR 72. The accidents postulated, and the Advanced NUHOMS® System components affected by each accident, are shown in Table 11.2-1.

The following sections provide descriptions of the analyses performed for each accident condition. The analyses demonstrate that the requirements of 10CFR 72.122 are met and that adequate safety margins exist for the Advanced NUHOMS® System design. The resulting accident condition stresses in the Advanced NUHOMS® System components are evaluated and compared with the applicable code limits set forth in Section 3.1.2. Where appropriate, these accident condition stresses are combined with those of normal operating loads in accordance with the load combination definitions in Tables 3.1-5, 3.1-10, and 3.1-11. Load combination results for the Advanced NUHOMS® System are presented in Sections 3.6.1 and 3.6.2. Material properties are provided in Section 3.3.

Radiological calculations are performed to confirm that on-site and off-site dose rates are within acceptable limits.

The postulated accident conditions addressed in this section include:

- Earthquake
- Tornado Wind Pressure and Tornado Generated Missiles
- Flood
- Fire/Explosion
- Cask Drop
- Lightning
- Blockage of AHSM Air Inlet and Outlet Openings
- Accidental Pressurization
- Burial
- Inadvertent loading of a newly discharged fuel assembly

11.2.1 Earthquake

11.2.1.1 Cause of Accident

Earthquake events are natural phenomena. For this high-seismic design application, the earthquake is postulated to be a large magnitude event, with spectral accelerations that bound, with significant margin, those of most operating nuclear power plant sites in the United States.

The horizontal design response spectrum of NRC Regulatory Guide 1.60 [11.2] is selected for the seismic analyses of the Advanced NUHOMS® System components. The R.G. 1.60 spectral shape is anchored at 1.5g ZPA for the horizontal direction. The vertical direction spectral accelerations are 2/3 of the horizontal. The design response spectrum is applied to the top of the ISFSI basemat in the seismic analyses discussed below. Any effects due to soil-structure interaction shall be demonstrated to fall within the design response spectra, conservatively including amplification to the AHSM cg, for each ISFSI site by the licensee.

11.2.1.2 Accident Analysis

Both linear and non-linear analyses are performed to determine the seismic response of the Advanced NUHOMS® System. Linear elastic analyses are used in the structural evaluation of the 24PT1-DSC and AHSM to determine stresses and/or forces and moments within these components. Non-linear analyses are used for the seismic stability analyses to determine the maximum sliding and rocking response of the AHSM array.

The stress analyses results due to seismic loads for the 24PT1-DSC and the AHSM are summarized in Section 3.6. The non-linear stability analyses are discussed in Section 11.2.1.2.1.

Maximum (enveloping of all analyses) sliding displacements are on the order of 44 inches (3.67 ft) in the X-direction and 34 inches (2.83 ft) in the Y-direction. These maximum sliding displacements are well within the criteria of 10 ft. in each horizontal direction. Maximum tipping/uplift is 0.06 inch for the nominal design case (AHSM in contact with each other), and 0.6 inch for the worst-case sensitivity analysis (AHSMs not in contact to one another).

The LS-DYNA analyses demonstrate that the response of the AHSM assembly is dominated by sliding of the AHSMs and that the rocking response is negligibly small.

11.2.1.2.3 24PT1-DSC and AHSM Modal Frequencies

As described in Section 3.6, lower bound estimates of the natural frequencies of the 24PT1-DSC are determined using closed-form calculations, and are over 33 Hz for both the lateral and axial directions.

The natural frequencies of the loaded AHSM are determined by performing a frequency analysis using the ANSYS [11.6] finite element analytical model shown in Figure 3.6-7 and 3.6-8. First mode global frequencies of the loaded AHSM in each orthogonal direction are determined to be over 33 Hz.

Thus, both, the 24PT1-DSC and AHSM can be considered as rigid structures for purposes of seismic evaluation.

11.2.1.2.4 Determination of Maximum Accelerations for Seismic Analyses of the AHSM and 24PT1-DSC

11.2.1.2.5 24PT1-DSC Seismic Stress Analysis

The seismic analysis of the 24PT1-DSC inside the AHSM is discussed in Section 3.6. In the lateral and vertical directions, the 24PT1-DSC is conservatively assumed to behave as a simply supported beam. In the axial direction, the entire mass of the basket assembly is conservatively applied as a pressure load on the 24PT1-DSC cover plates to maximize the stresses in the cover plate to shell welds.

11.2.1.2.6 AHSM Seismic Analysis

An equivalent static analysis of the AHSM is performed using the ANSYS model described in Section 3.6.2.3.1 for 1.5g longitudinal, 1.5g transverse and 1.0g vertical accelerations in the base block. Analysis of the top shield block was conservatively performed for 2.25g (1.5g horizontal acceleration with 1.5 amplification) in the longitudinal and the transverse directions, and 1.0g in the vertical direction.

The responses for each orthogonal direction are combined using the SRSS method.

The seismic analysis results are incorporated in the loading combinations C4C (Table 3.6-12) and C4S (Table 3.6-13) for the concrete and the support structure components, respectively.

11.2.1.2.7 Transfer Cask Seismic Analysis

The OS197 transfer cask, when mounted on the transfer trailer during a 1.5g earthquake, is subjected to stresses which are bounded by the 80 inch cask drop analysis. The maximum height of the cask bottom above the grade level is 78 inches. This assumes that the trailer wheels and suspension have sufficient strength to allow for full rotation; if this event were to happen, the wheels, axles, and the trailer deck would likely collapse and the distance to the bottom of the cask would be significantly reduced. Additionally, the cask skid and the trailer will act as impact limiting devices due to the excessive deformations in these components that will occur during a roll-over event. The cask would, in all probability, break free of the top trunnion cask hold downs and would roll from the skid, further reducing the impact accelerations. Therefore, this condition is bounded by the cask drop analysis documented in Section 11.2.5.

Even assuming that the cask/trailer does not tip over, but the cask "rattles" and remains mounted on the trailer, these loads are bounded by the 75g accident drop evaluation documented in Section 11.2.5.

11.2.1.3 Accident Dose Calculations

The Advanced NUHOMS® System components are conservatively designed and analyzed to withstand the forces generated by a postulated design basis earthquake and do not fail. Hence, there are no dose consequences resulting from an earthquake.

11.2.1.4 Corrective Actions

Inspection of AHSMs subsequent to a significant earthquake is required to identify potential damage or change in AHSM configuration. Repair of damage to AHSM concrete components, including shield walls may be necessary. Movement of AHSMs as a result of the seismic event will require evaluation and possible repositioning of AHSMs and site specific shielding to pre-seismic event configuration.

11.2.2 Tornado Wind Pressure and Tornado Missiles

11.2.2.1 Cause of Accident

In accordance with ANSI-57.9 [11.1] and 10CFR 72.122 [11.7], the Advanced NUHOMS® System AHSM is designed for tornado effects including tornado wind loads. In addition, the AHSM is also designed for tornado missile effects. The Advanced NUHOMS® System is designed to be located anywhere within the United States; therefore, the most severe tornado wind and missile loadings specified by NUREG-0800 [11.4] and NRC Regulatory Guide 1.76 [11.9] are selected as a design basis for this postulated accident.

11.2.2.2 Accident Analysis

The applicable design parameters for the design basis tornado (DBT) are presented in Section 2.2.1.

Tornado pressure drop effects on the DSC are enveloped by internal design basis pressure analyses.

The determination of the tornado wind pressures and tornado missile loads acting on the AHSM are detailed in Section 3.6.2.2.

Stability and stress analyses are performed to determine the response of the AHSM to tornado wind pressure loads. The stability analyses are performed using closed-form calculation methods to determine sliding and overturning response of the AHSM array. A single AHSM with both the end and the rear shield walls is conservatively selected for the analyses. The stress analyses are performed using the ANSYS finite element model of a single AHSM to determine design forces and moments. These conservative generic analyses envelop the effects of wind pressures on the AHSM array. Thus, the requirements of 10CFR 72.122 are met.

In addition, the AHSM is evaluated for tornado missiles. The adequacy of the AHSM to resist tornado missile loads is addressed using empirical formulae [11.15].

11.2.2.2.1 Effect of DBT Wind Pressure Loads on AHSM

As described in Section 3.6.2.2, the AHSM is qualified for maximum DBT generated design wind loads of 397 lb/ft^2 and 196 lb/ft^2 on the windward and leeward AHSM walls, respectively and a pressure drop of 3 psi.

A single stand-alone AHSM is protected by shield walls on either side and at the rear. For an AHSM array, the critical module is on the windward end of the array. This module has an end shield wall to protect the module from tornado missile impacts. The shield wall is also subjected to the 397 lb/ft^2 windward pressure load. The leeward side of the same end module in the array has no appreciable suction load due to the proximity of the adjacent module. The 196 lb/ft^2 suction load is applicable to the end shield wall on the opposite end module in the array. A suction of 357 lb/ft^2 is also applied to the top shield block of each AHSM in the array.

For the stress analyses, the DBT wind pressures are applied to the AHSM as uniformly distributed loads. The rigidity of the AHSM in the transverse direction (frame and shear wall action of a single AHSM) is the primary load transfer mechanism assumed in the analysis. The bending moments and shear forces at critical locations in the AHSM concrete components are calculated by performing an analysis using the ANSYS analytical model of the AHSM. The resulting moments and shear forces are included in the AHSM load combination results reported in Section 3.6.2.2.

For conservatism, the design basis operating wind pressure loads are assumed to be equal to those calculated for the DBT in the formulation of AHSM load combination results.

A stability analysis is performed to evaluate the effects of overturning and sliding due to the postulated DBT. A single, freestanding AHSM with two end shield walls and rear shield wall is used for this analysis.

The pressure drop has no effect on the AHSM, since the AHSM is an open structure, due to the presence of the inlet and outlet vents.

11.2.2.2.1.1 AHSM Overturning Analysis

For the DBT wind overturning analysis, the overturning moment and the resulting stabilizing moments are calculated.

The stabilizing moment (M_{st}) for the windward module plus end shield walls is:

$$M_{st} = W(d + d_s) + W_s\,(d_s/2 + 2d + d_s) + W_s\,(d_s/2)$$

Where:

W = 396 K, Weight of AHSM + 24PT1-DSC with minimum weight

W_s = 163 K, Weight of end module shield wall

d = 50.5 in., Horizontal distance between center of gravity of AHSM to the outer edge of the module.

d_s = 36 in. , Thickness of the shield wall

Therefore: M_{st} = 62,400 K-in.

and the overturning moment (M_{to}) for the windward module plus shield wall due to DBT wind pressure is:

$$M_{to} = [(W1 + W_2)\,A_w h/2 + W_3 A_r (d + d_s)]12$$

Where:

W_1 = 0.397 K/ft.^{2}, Wind load, windward wall

W_2 = 0.196 K/ft^2, Wind load, leeward wall

h = 18.5 ft, Wall height

	W_3	=	0.357 K/ft.^{2}, Wind uplift on roof
	A_r	=	282.4 ft.^{2}, Top shield block area (including shield walls)
	A_w	=	362.3 ft.^{2}, Wall area
	d_s	=	3 ft. Thickness of the end shield wall
	d	=	4.21 ft., Half of the transverse dimension of the roof
Therefore:	M_{to}	=	32,600 K-in.

Because the overturning moment is smaller than the stabilizing moment, the freestanding AHSM will not overturn. The resulting factor of safety against overturning effects for the DBT wind loads is 1.9.

11.2.2.2.1.2 AHSM Sliding Analysis

To evaluate the potential for sliding of a single, free-standing AHSM, the sliding force generated by the postulated DBT wind pressure is compared to the sliding resistance provided by friction between the base of the AHSM and the ISFSI basemat.

The force (F_{sl}) required to slide the end module in an array is:

$$F_{sl} = [W+2W_s - W_3A_r]\mu$$

Where: μ = 0.6, coefficient of friction [11.10]

W, W_s, W_3 and A_r are defined above.

Substituting gives:

$$F_{sl} = 372.6 \text{ K}$$

The sliding force (F_{hw}) generated by DBT wind pressure for a single AHSM is:

$$F_{hw} = (W_1 + W_2)\, A_w$$

Where: W_1,W_2 and A_w are as defined above.

Substituting gives:

$$F_{hw} = 214.8 \text{ K}$$

Because the horizontal force generated by the postulated DBT is smaller than the force required to slide the end module in an AHSM array, the AHSM will not slide. The factor of safety against sliding of the AHSM due to DBT wind loads is 1.73.

11.2.2.2.2 AHSM Missile Impact Analysis

11.2.2.2.2.1 Local Damage Evaluation

Local missile impact effects consist of (a) missile penetration into the target, (b) missile perforation through the target and (c) spalling and scabbing of the target. This also includes punching shear in the region of the target. As per the ACI code [11.10] if the concrete thickness is at least 20% greater than that required to prevent perforation, the punching shear requirement of the code need not be checked. Several empirical formulas are available which are used to predict local damage effects.

The following enveloping missiles (based on the mass of the missile) are considered for local damage:

- Utility pole
- Armor piercing artillery shell
- Steel pipe

Large deformable missiles such as automobiles do not penetrate the structure. Therefore, the local effects from an automobile are evaluated using punching shear criteria of the ACI Code [11.10].

The following empirical formulas are used to determine the local damage effects:

Reinforced Concrete Target

(a) Modified NDRC formulas for penetration depth [11.15]:

$$x = [4KNWd^{-0.8}(v_o/1000d)^{1.8}]^{0.5} \qquad \text{for } x/d \leq 2.0$$

$$x = \{[KNW(v_o/1000d)^{1.8}] + d\} \qquad \text{for } x/d > 2.0$$

where, x = Missile penetration depth, inches

K = concrete penetrability factor = $180/\sqrt{fc'}$

N = projectile shape factor

= 0.72 flat nosed

= 0.84 blunt nosed

= 1.0 bullet nosed (spherical end)

= 1.14 very sharp nose

W = weight of missile, lbs

v_o = striking velocity of missile, fps

d = effective projectile diameter, inches.

for a solid cylinder, d = diameter of projectile and

for a non-solid cylinder, $d = (4A_c/\pi)^{1/2}$

A_c = projectile impact area, in^2

(b) Modified NDRC formula for perforation thickness [11.15]:

$$(e/d) = 3.19(x/d) - 0.718(x/d)^2 \qquad \text{for } x/d \le 1.35$$

$$(e/d) = 1.32 + 1.24\,(x/d) \qquad \text{for } 1.35 \le x/d \le 13.5$$

where e = perforation thickness, in.

In order to provide an adequate margin of safety the design thickness t_d = 1.2 e [11.10]

(c) Modified NDRC formula for scabbing thickness [11.15]:

$$(s/d) = 7.91(x/d) - 5.06(x/d)^2 \qquad \text{for } x/d \le 0.65$$

$$(s/d) = 2.12 + 1.36\,(x/d) \qquad \text{for } 0.65 \le x/d \le 11.75$$

where s = scabbing thickness, in.

In order to provide an adequate margin of safety the design thickness t_d = 1.2 s [11.10]

The concrete targets of the AHSM which may be subjected to local damage due to missile impact are:

- 60" thick top shield block
- 30" thick front block
- 36" thick end shield wall with 12" thick side wall
- 36" thick rear shield wall with 12" thick rear wall
- 36" thick rear shield wall at the rear of the top shield block (with vent opening)
- 24" thick composite shielding door (with 0.5" thick steel plate at rear)

Steel Targets

The steel barriers subjected to missile impact are designed to preclude perforation. The steel plate thickness for threshold of perforation is [11.17]:

$$T_p = (E_k)^{2/3} / 672D$$

Where: $E_k = M_m v_o^2/2$

T_p = steel plate thickness for threshold of perforation (in)

E_k = missile kinetic energy (ft-lbs)

M_m = mass of the missile (lb-sec^2 /ft)

v_o = missile striking velocity (fps)

D = missile diameter (in), for pipe missiles, D is the outside diameter of the pipe

The design thickness to prevent perforation is $t_p = 1.25\ T_p$ [11.17].

The steel target of the AHSM which may be subjected to local damage due to missile impact is the composite steel door (24" thick concrete + 0.5" thick steel plate at rear).

11.2.2.2.2.1.1 Local Missile Impact Effects of Utility Pole Missile

The wood missiles (utility pole missile) do not have sufficient strength to penetrate a concrete target and the scabbing thickness required for wood missiles is substantially less than that required for a steel missile with the same mass and velocity. Practical wooden pole missiles are not capable of causing local damage to walls 12 inches thick, or greater for the missile velocities considered. Because none of the concrete targets are less than 12 inch thick, the postulated wood missiles will not cause any local damage to the AHSM concrete structure. Steel targets are also resistant to penetration which implies that only nondeformable missiles can perforate a steel target.

11.2.2.2.2.1.2 Local Missile Impact Effects of Armor Piercing Artillery Shell

Concrete Wall Evaluation:

d = diameter of missile = 8"

W = 280 lbs (conservatively assumed)

V_o = 185 fps

f_c' = 5000 psi

K = $180/\sqrt{5000} = 2.55$

N = 0.84 blunt nosed

Penetration thickness = x $= 4.67$ in for $x/d = 0.584 \leq 2$

Perforation thickness e = 12.95"

Required Perforation thickness = 1.2*12.95 = 15.5"

Scabbing thickness s = 23.1" inches

Required scabbing thickness = 1.2*23.2 = 27.7"

Shielded Door Evaluation:

Required perforation thickness of concrete is 15.5" which is less than 24". Therefore, the missile will not perforate the concrete in the shielded door. The missile will not scab the concrete because the shielded door is provided with 0.5" thick rear steel plate.

11.2.2.2.2.1.3 Local Missile Impact Effects of 12 Inch Diameter Steel Pipe Missile

Concrete Wall Evaluation:

Diameter of missile = 12.75" (Outer diameter of 12" dia Sch 40 pipe)

Contact surface area = A_c = 15.7 in^2 (cross section metal area of 12" dia Sch 40 pipe)

Effective diameter = d = $(4*15.7/\pi)^{1/2}$ = 4.47 inches

W = 1500 lbs

v_o = 205 fps

f_c' = 5000 psi

K = $180/\sqrt{5000} = 2.55$

N = 0.72 flat nosed

Perforation thickness x = 15.2 in for x/d > 2

Perforation thickness e = 24.75 in

Required perforation thickness 1.2*24.75 = 29.7"

Scabbing thickness = s = 30.15 inches

Required scabbing thickness = 1.2*30.15 = 36.2 inches

The top shield block (60" thick), front block (30" thick) and the end shield walls (36" thick +12" thick wall = 48") will not be perforated. However, the missile may produce scabbing in the front block and rear shield wall above the bottom of the top shield block. Assuming some scabbed concrete from the front block and rear shield wall above the bottom of the top shield block, will fall into the vent openings, the possibility of causing a blocked vent scenario exists. This scenario is addressed in Section 11.2.7. In a worst case scenario where debris blocking the exhaust vent can not be removed, DSC unloading within 40 hours or prior to AHSM temperature exceeding allowable limits may be required.

Shielded Door Evaluation:

The required perforation thickness is 29.7". However, the thickness of concrete in the door is 24". Therefore, the missile will perforate the concrete in the door.

The exit velocity at perforation is calculated as follows:

$$\nu_R^{1.8} = \nu_I^{1.8} - \nu_P^{1.8}$$

where ν_R = Residual missile velocity

ν_I = Missile impact velocity ($=\nu_o$)

ν_P = Velocity to just perforate the target (i.e., no residual velocity)

In order to compute ν_P first determine x at e = 24"

$$x = (e/d - 1.32)d/1.24 = 14.6 \text{ in}$$

$$\nu_P = [(x-d)(1000d)^{1.8} / KNW)^{1/1.8} = 198.6 \text{ fps}$$

Therefore, $\nu_R^{1.8} = 205^{1.8} - 198.6^{1.8}$

$$\nu_R = 41.1 \text{ fps.}$$

Evaluate the local missile effect of 0.5" thick rear steel plate at the rear of the door subjected to the missile at 41.1 fps.

M_m = 1500/32.2 = 46.6 lb-sec²/ft

ν_s = 41.1 fps

E_k = 39359

D = 12.75 in

Tp = 0.14 inches

The required thickness = 1.25 Tp = 1.25*0.14 = 0.18 inches < 0.5"

Therefore, the steel door will not be perforated by this missile. Also the rear steel plate will prevent the concrete from scabbing.

11.2.2.2.2.2 Massive Missile Impact Analysis

The AHSM stability and potential damage due to impact of the postulated DBT massive missile consisting of a 4000 lb. automobile, 20 sq. ft. frontal area traveling at 195 ft./sec., is evaluated. The massive missile is assumed to impact the shield wall of an end module in an array. Using the principles of conservation of momentum with a coefficient of restitution of zero, the analysis presented below demonstrates that the end module remains stable and the missile energy is dissipated by sliding or slight tipping of the module.

Using conservation of momentum, the missile impact force equals the change in linear (sliding) or angular (overturning) momentum of the AHSM. The AHSM velocities immediately after impact are:

Sliding: $V = (m*v_i) / (M+m)$ (Eq. 11.2-5)

Overturning: $\omega_a = (m*d_m*v_i)/(m*d_m^2 + I_A)$ (Eq. 11.2-6)

Where, V = initial linear velocity of module after impact

v_i = 195 ft/sec = initial velocity of missile (conservative)

ω_A = initial rotational velocity about bottom right corner of the module and end shield walls (Figure 11.2-15)

d_m = Vertical distance of the CG of the missile from A (Figure 11.2-15) = 198 inches

m = 4000/386.4 = 10.35 lb-sec²/in = mass of the missile

M = (318.74+77.0+2*163.0)*1000/386.4 = 1868 lb-sec² /in = Mass of loaded AHSM + End Shield walls

d = (77.0*102+318.74*126.11)/(77.0+318.74) in = 121.42 (Elevation of the CG of the empty AHSM = 126.11 in)

I_A = Mass moment of inertia of loaded AHSM about point A (Figure 11.2-15)

I_A = 5.17×10^7 lb-sec²-in

Sliding:

From Eq. 11.2-5: V = 12.90 in/sec = 1.075 ft/sec

For an impact at the bottom of the AHSM wall, the kinetic energy imparted to the AHSM is absorbed by sliding friction between the concrete of the AHSM and the basemat. Coefficient of friction is 0.6 [11.10].

Assuming that the missile impact load results in sliding of the AHSM and equating the kinetic energy generated by the moving module to the work done by sliding friction force gives:

$$\mu * g * (M+m) * \Delta = (M+m)*V^2/2$$

$$\Delta = 0.0299 \text{ ft} = 0.36 \text{ inch}$$

Therefore, a massive missile impact on a single AHSM will slide the complete module approximately 0.36 inches sideways. The sliding distance is significantly reduced due to presence of more than one module side by side. Considering a three array module:

$$M = [3*(318.74+77.0) + 2*163]*1000/386.4$$

$$= 3916.2 \text{ lb-sec}^2/\text{in}$$

$$V = 0.52 \text{ ft/sec}$$

$$\Delta = 0.084 \text{ inches}$$

Therefore, the sliding displacement of the modules due to a massive missile impact is insignificant and will not cause any structural damage.

<u>Overturning:</u>

When the massive missile impacts at the top of the AHSM, the missile energy is absorbed by plastic deformation of the missile and in rotation of the AHSM. Therefore, equating the loss of kinetic energy to increase in the potential energy:

$$I_A\, \omega_A^2 /2 = M *g*d\, [\cos(\beta+\alpha-90)-\cos\beta] \quad \text{(Figure 11.2-15)}$$

From Eq. 11.2-6: $\omega_A = 0.092$ rad/sec

$$\beta = \tan^{-1} \{(50.5+36)/ 121.42)= 35.5°$$

$$M = 1868 \text{ lb-sec}^2/\text{in}$$

$$\cos(35.47+\alpha-90) - \cos (35.47) = 0.002507$$

$$\cos(35.47+\alpha- 90) = 0.002507 +0.81442 = 0.816926$$

$$90-\alpha = 35.47-35.22 = 0.25$$

Therefore, a loaded AHSM rotates a maximum of 0.25° from vertical. The loaded AHSM is stable against overturning as tip-over does not occur until the CG rotates past the edge point (point A Figure 11.2-15) to an angle of more than 35.5° [= $\tan^{-1}$(86.5/121.42)].

Displacement at top of AHSM =222*tan(0.25) =0.97". The maximum uplift at one edge = 173*tan(0.25) = 0.76". However, this tipping displacement is prevented by the seismic ties and keys, which connect the AHSM to the adjacent module.

11.2.2.3 Accident Dose Calculations

Each exposed component of the Advanced NUHOMS® System is specifically designed to withstand tornado-generated missiles as discussed in the preceding paragraphs. Loss of structural bending strength of the shield wall(s) due to tornado missile impact, should it occur, is acceptable and does not affect the safe operation of the AHSM. Recovery from this event can be performed in a planned and deliberate manner to replace the shield wall(s). This requires temporary shielding during removal and replacement of the wall(s), or removal of the AHSM from service. At no time is there a danger of a release of radioactive materials to the general public.

11.2.2.4 Corrective Actions

Evaluation of AHSM damage as a result of a Tornado is to be performed to assess the need for temporary shielding and AHSM repairs to return the AHSMs to pre-tornado design conditions.

11.2.3 Flood

11.2.3.1 Cause of Accident

Flooding conditions (such as tsunami and seiches) simulating a range of flood types, as specified in 10CFR 72.122(b) are considered. In addition, floods resulting from other sources, such as high water from a river or a broken dam, are postulated as the cause of the accident.

11.2.3.2 Accident Analysis

Because the source of flooding is site specific, the exact source, or quantity of flood water, should be established by the licensee. However, for this generic evaluation of the 24PT1-DSC and AHSM, flood conditions are specified that envelope those postulated for most plant sites. As described in Section 3.1.2.2 the design basis flood load is specified as a 50-foot static head of water and a maximum flow velocity of 15 feet per second. Each licensee should confirm that this represents a bounding design basis for their specific ISFSI site.

11.2.3.2.1 AHSM Flooding Analysis

Because the AHSM is open to the atmosphere, static differential pressure due to flooding is not a design load.

The maximum drag force, F, acting on the AHSM due to a 15-fps flood water velocity is calculated as follows [11.12]:

$$F = (v^2/2g)C_d A \rho_w$$

Where:

v = 15 fps, Flood water velocity

C_D = 2.0, Drag coefficient for flat plate

A = 18.5 ft., AHSM area per foot length

ρ_w = 62.4 lb./ft.3, Flood water density

F = Drag force (lb.)

g = 32.2 ft./s^2 = Acceleration due to gravity

The resulting flood induced pressure load of 8.07 K/ft. is applied normally to the end module shield wall of a stand-alone AHSM.

11.2.3.2.1.1 AHSM Overturning Analysis

The factor of safety against overturning of a single AHSM with shield walls, for the postulated flooding conditions, is calculated by summing moments about the bottom outside corner of a single, free-standing AHSM. A net weight of 239.3 kips for a loaded AHSM plus 95.2 kips for the upstream end shield wall, including buoyancy effects, is used to calculate the stabilizing moment resisting the overturning moment applied to the AHSM by the flood water drag force. The stabilizing moment is:

$$M_{st} = 239.3 \times 50.5 + 95.2 \times (101+18)$$

$$= 23{,}400 \text{ K-in.}$$

The maximum drag force due to the postulated water current velocity of 15 fps is calculated in Section 11.2.3.2.1 as 8.07 k/ft. acting over the entire height and width of an end shield wall of a single free-standing AHSM. Therefore, the overturning moment due to the postulated flood current is:

$$M_{ot} = 8.07 \text{ K/ft.} \times 19.583 \text{ ft.} \times (18.5 \times 12/2)$$

$$= 17{,}500 \text{ K-in.}$$

The factor of safety (F.S.) against overturning for a single, freestanding AHSM due to the postulated design basis flood water velocity is given by:

F.S. = 23,400 / 17,500 = 1.3

11.2.3.2.1.2 AHSM Sliding Analysis

The factor of safety against sliding of a freestanding single AHSM due to the maximum postulated flood water velocity of 15 fps is calculated using methods similar to those described above. The effective weight of the AHSM including the 24PT1-DSC and end shield wall acting vertically downward, less the effects of buoyancy acting vertically upward is 334.5 k. The friction force resisting sliding of the AHSM is equal to the product of the net weight of the AHSM and 24PT1-DSC and the coefficient of friction for concrete placed against another concrete surface such as that between the AHSM and basemat, which is 0.6 [11.10]. Therefore, the force resisting sliding of the AHSM is 0.6 x 334.5 or 200.7 kips. The drag force acting on a single AHSM is 8.07 kips/ft x19.583ft = 158.1 kips total acting on the side wall of a single AHSM, due to a flood velocity of 15 fps. The resulting factor of safety against sliding of a single free standing AHSM due to the design basis flood water velocity is 1.27.

11.2.3.2.2 <u>24PT1-DSC Flooding Analyses</u>

The 24PT1-DSC is evaluated for the design basis 50-foot hydrostatic head of water producing external pressure of 21.7 psi on the 24PT1-DSC shell and outer cover plates. A pressure of 22 psi is used for the structural evaluations.

The 24PT1-DSC shell stresses for the postulated flood condition are determined using the ANSYS analytical model shown in Figure 3.6-1. The 22-psig external pressure is applied to the model as a uniform pressure on the outer surfaces of the top cover plate, 24PT1-DSC shell and bottom cover plate. Flood induced stresses are combined in accordance the load combinations in Table 3.6-1. The resulting total stresses for the 24PT1-DSC are reported in Table 3.6-3.

11.2.3.2.3 <u>Thermal Evaluation of Flood Accident</u>

The thermal analyses and consequences of the flood accident are discussed in Chapter 4.

11.2.3.3 <u>Accident Dose Calculations</u>

The radiation dose due to flooding of the AHSM is negligible. The radioactive material inside the 24PT1-DSC will remain confined in the 24PT1-DSC and, therefore, will not contaminate the encroaching flood water. The minimal amount of contamination that may be on the outside surface of the 24PT1-DSC is not sufficient to be a radiological hazard if it were to be washed off the 24PT1-DSC outer surface.

11.2.3.4 <u>Corrective Action</u>

If flooding should occur, any silt deposits can be removed using a pump suction hose, or fire hose inserted through the inlet vent, to suck the silt out, or produce a high velocity water flow to flush the silt through the AHSM inlet vent. The corrosion inhibiting design features of the 24PT1-DSC are addressed in Section 3.4. The AHSM design allows temporary removal of a segment of the front vent concrete block to facilitate removal of silt deposits. Temporary shielding may be required during this removal process.

11.2.4 <u>Fire/Explosion</u>

11.2.4.1 <u>Cause of the Accident</u>

Combustible materials will not normally be stored at an ISFSI. Therefore, a credible fire would be very small and of short duration such as that due to a fire or explosion from a vehicle or portable crane.

However, a hypothetical fire accident is evaluated for the Advanced NUHOMS® System based on a fuel fire. The source of fuel is postulated to be from a ruptured fuel tank of the transfer cask transporter tow vehicle. The bounding capacity of the fuel tank is 300 gallons and the bounding hypothetical fire is an engulfing fire around the transfer cask. Direct engulfment of the AHSM is highly unlikely. Any fire within the ISFSI boundary while the DSC is in the AHSM would be bounded by the fire during transfer cask movement. The AHSM concrete acts as a significant insulating fire wall to protect the 24PT1-DSC from the high temperatures of the fire.

11.2.4.2 Accident Analysis

The evaluation of the hypothetical fire event is presented in Section 4.6.4 of the SAR. The fire thermal evaluation is performed primarily to demonstrate the confinement integrity and fuel retrievability of the 24PT1-DSC. This is assured by demonstrating that the DSC temperatures and internal pressures will not exceed those of the blocked vent condition (see Section 11.2.7) during the fire scenario. Peak temperatures for the Advanced NUHOMS® System components are summarized in Table 4.6-1.

As shown in Chapter 3, the 24PT1-DSC is designed for 22 psi external pressure due to flood and the AHSM is designed to withstand tornado wind pressures. These pressures are considered significantly higher than the pressures generated by a credible explosion in the general vicinity of the AHSM.

11.2.4.3 Accident Dose Calculations

The 24PT1-DSC confinement boundary will not be breached as a result of the postulated fire/explosion scenario. Accordingly, no 24PT1-DSC damage or release of radioactivity is postulated. Because no radioactivity is released, no resultant dose increase is associated with this event.

The fire scenario may result in the loss of cask neutron shielding should the fire occur while the 24PT1-DSC is in the cask. The effect of loss of the neutron shielding due to a fire is bounded by that resulting from a cask drop scenario. See Section 11.2.5.3 for evaluation of the dose consequences of a cask drop.

11.2.4.4 Corrective Actions

Evaluation of AHSM or cask neutron shield damage as a result of a fire is to be performed to assess the need for temporary shielding (for AHSM or cask, if fire occurs during transfer operations) and repairs to restore the transfer cask and AHSM to pre-fire design conditions.

11.2.5 Accidental Drop of the 24PT1-DSC Inside the Transfer Cask

11.2.5.1 Cause of Accident

This section addresses the structural integrity of the 24PT1-DSC shell and internal basket assemblies when subjected to postulated cask drop accident conditions. Drops are postulated for the 24PT1-DSC when positioned inside the transfer cask and can not occur once the 24PT1-DSC is transferred into the AHSM.

11.2.5.1.1 Cask Handling and Transfer Operation

As described in Chapter 8, handling operations involving hoisting and movement of the on-site transfer cask and 24PT1-DSC are typically performed inside the plant's fuel handling building. These include utilizing the crane for placement of the empty 24PT1-DSC into the transfer cask cavity, lifting the transfer cask/24PT1-DSC into and out of the plant's spent fuel pool, and placement of the transfer cask/24PT1-DSC onto the transport skid/trailer. An analysis of the plant's lifting devices used for these operations, including the crane and lifting yoke, is needed to address a postulated drop accident for the transfer cask and its contents. The postulated drop accident scenarios addressed in the plant's 10CFR 50 licensing basis are plant specific and should be addressed by the licensee.

Once the transfer cask is loaded onto the transport skid/trailer and secured, it is pulled to the AHSM site by a tractor vehicle. A predetermined route is chosen to minimize the potential hazards that could occur during transport. This movement is performed at very low speeds. System operating procedures and technical specification limits defining the safeguards to be provided ensure that the system design margins are not compromised. As a result, it is highly unlikely that any plausible incidents leading to a transfer cask drop accident could occur. Similarly, at the ISFSI site, the transport skid/trailer is backed-up to, and aligned with, the AHSM using hydraulic positioning equipment. The transfer cask is then docked with, and secured to, the AHSM access opening. The loaded 24PT1-DSC is transferred to or from the AHSM using a hydraulic ram system. The hold down mechanisms that secure the transfer cask to the transport skid/trailer remain in place at all times during the 24PT1-DSC transport. As a result, there is no reasonable way during these operations for a cask drop accident to occur.

11.2.5.1.2 24PT1-DSC Drop Accident Scenarios

In spite of the highly incredible nature of any scenario that could lead to a drop accident for the transfer cask, the following drop scenarios are conservatively selected for design of the 24PT1-DSC:

1. A 75g horizontal side drop.

2. A 25g oblique corner drop at an angle of 30° to the horizontal, onto the corner of the transfer cask.

A vertical end drop is not credible because the 24PT1-DSC is not handled in the vertical orientation once it is loaded onto the transfer trailer. However, for purposes of bounding the 25g corner drop, and as part of 10CFR 50 and 10CFR 71 evaluations, the 24PT1-DSC is also analyzed for a 60g end drop.

11.2.5.1.3 Transfer Cask Drop Surface Conditions

Because of the passive nature of the Advanced NUHOMS® System operations and the protective measures taken during transfer of the transfer cask to and from the AHSM, it is concluded that a postulated cask drop accident is much less plausible during transfer from the fuel handling building to the ISFSI than during transfer operations within the ISFSI. Site conditions away from the AHSM storage pad will typically be relatively thin concrete slabs (12 inch or less), asphalt road surfaces or compacted gravel. The target hardness numbers of these surfaces are typically small compared with the concrete parameters provided in Chapter 12. Therefore, the expected cask decelerations for a cask drop accident will be substantially less than the assumed 75g-side drop, and the 25g corner drop design basis loadings.

Furthermore, the impact of an object as massive and stiff as the transfer cask, will tend to punch through lightly reinforced concrete slabs because of the very high shear stresses induced over small areas. Punching shear failures would be expected to occur for deceleration values ranging from as low as 0.5gs for a corner drop, to 2.6gs for a side drop. For these reasons, the cask drop scenarios postulated and evaluated by site license applicants should focus on conditions that exist at the ISFSI site location.

11.2.5.2 Accident Analysis

The stress analyses of the 24PT1-DSC resulting from the two drop scenarios are summarized in Section 3.6.

11.2.5.3 Accident Dose Calculations

The accidental transfer cask drop scenarios do not impact the transfer cask/24PT1-DSC confinement boundary. The transfer cask lead shielding is not impacted by these drops. The transfer cask neutron shield, however, may be damaged in an accidental drop.

The loss of the neutron shield has been previously analyzed in C of C 72-1004 FSAR, Section 8.2.5.3 [11.16]. The transfer cask surface dose rate, with the neutron shield intact for the 24PT1-DSC in the transfer cask is calculated in Chapter 5 of this SAR as 419 mrem/hr gamma and 145 mrem/hr neutron. This is bounded by the 72-1004 surface dose rate of 428 mrem/hr gamma and 164 mrem/hr neutron ([11.16], Table 7.3-2).

Therefore, the dose rate at the transfer cask surface due to the loss of the neutron shield will be less than 2128 mrem/hr gamma and neutron ([11.16] , Section 8.2.5.3.2), which is bounding for the 24PT1-DSC.

The only potential off-site dose consequences would be additional direct and air scattered radiation if the accident were to occur sufficiently close to the site boundary. It is assumed that eight hours would be required to either recover the neutron shield or to add temporary shielding while arranging recovery operations. As a result, it is estimated that on-site workers at an average distance of fifteen feet would receive an additional dose of less than 2.5 rem (310 mrem/hr for 8 hours) ([11.16], Section 8.2.5.3.2).

Off-site individuals at a distance of 2000 feet would receive an additional dose of 0.04 mrem for the assumed eight hour exposure ([11.16], Section 8.2.5.3.2). Comparing to the annual dose for 600 meters from Figure 10.2-1 (front of AHSM), this dose is approximately twice the normal annual dose. Extrapolating from Figure 10.2-2, this dose represents approximately 22 mrem at 100 meters. This increase is well within the limits of 10CFR 72 for an accident condition. Also, this does not preclude handling operations for recovery of the transfer cask and its contents. Water bags or other neutron absorbing material could be wrapped around the transfer cask to reduce the surface dose rate to an acceptable level for recovery operations, thus minimizing exposure of personnel in the vicinity. The actual local and off-site dose rates, recovery time and operations needed to retrieve the cask, and the required actions to be performed following the event, depend upon the severity of the event, site characteristics and the resultant cask and trailer/skid damage.

11.2.5.4 Corrective Actions

The DSC will be inspected for damage, and the DSC opened and the fuel removed for inspection, as necessary. Removal of the transfer cask top cover plate may require cutting of the bolts in the event of a corner drop onto the top end. These operations will take place in the plant fuel building decontamination area and spent fuel pool after recovery of the transfer cask.

Following recovery of the transfer cask and unloading of the DSC, the transfer cask will be inspected, repaired and tested as appropriate prior to reuse.

For recovery of the cask and contents, it may be necessary to develop a special sling/lifting apparatus to move the transfer cask from the drop site to the fuel pool. This may require several weeks of planning to ensure all steps are correctly organized. During this time, lead blankets may be added to the transfer cask to minimize on-site exposure to site operations personnel. The transfer cask would be roped off to ensure the safety of the site personnel.

11.2.6 Lightning

11.2.6.1 Cause of Accident

Lightning striking the AHSM and causing an off-normal condition is not considered credible.

Lightning protection system requirements are site specific and depend upon the frequency of occurrence of lightning storms in the proposed ISFSI location and the degree of protection offered by other grounded structures in the proximity of the AHSMs. The addition of simple lightning protection equipment, if required by plant criteria, to AHSM structures (i.e., grounded handrails, ladders, etc.) is considered a miscellaneous attachment and is allowed by the AHSM drawing (Dwg. No. NUH-03-4011), Section 1.5.2.

11.2.6.2 Accident Analysis

Should lightning strike in the vicinity of the AHSM the normal storage operations of the AHSM will not be affected. The current discharged by the lightning will follow the low impedance path offered by the surrounding structures. Therefore, the AHSM will not be damaged by the heat or mechanical forces generated by current passing through the higher impedance concrete. Since the AHSM requires no electrical equipment for its continued operation, the resulting current surge from the lightning will not affect the normal operation of the AHSM.

11.2.6.3 Accident Dose Calculations

Since no off-normal condition will develop as the result of lightning striking in the vicinity of the AHSM, no radiological consequences are expected.

11.2.6.4 Corrective Actions

No corrective actions are required since no damage to the AHSM or 24PT1-DSC is expected.

11.2.7 Blockage of Air Inlet and Outlet Openings

11.2.7.1 Cause of Accident

This accident conservatively postulates the complete blockage of the AHSM ventilation air inlet and outlet openings.

Since the Advanced NUHOMS® System AHSMs are located outdoors, there is a remote probability that the ventilation air inlet and outlet openings could become blocked by debris from such unlikely events as floods and tornadoes. Thus, for this conservative generic analysis, such an accident is postulated to occur and is analyzed.

11.2.7.2 Accident Analysis

The structural consequences due to the weight of the debris blocking the air inlet and outlet openings are negligible and are bounded by the AHSM loads induced for a postulated tornado (Section 11.2.2) or earthquake (Section 11.2.1).

The thermal analysis of the blocked vent condition is presented in Chapter 4.

The thermal-induced stresses for the blocked vent case are calculated using the AHSM structural models discussed in Section 3.6.2.3. The resulting elastic forces and moments are modified to account for the concrete cracked section properties in accordance with ACI 349 Appendix A, and combined with the calculated forces and moments from other loads.

11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation where it is conservatively estimated that the on-site workers will receive an additional dose of no more than one man-rem during the eight hour period it is estimated may be required for removal of the debris from the air inlet and outlet openings in the AHSM.

11.2.7.4 Corrective Actions

Blockage of the AHSM vents is to be cleared within the 40 hour time frame analyzed to restore AHSM ventilation.

11.2.8 Accidental Pressurization of the 24PT1-DSC

11.2.8.1 Cause of Accident

The bounding internal pressurization of the 24PT1-DSC for this conservative generic evaluation is postulated to result from cladding failure of the spent fuel, and the consequent release of spent fuel rod fill gas and free fission gas.

11.2.8.2 Accident Analysis

Analysis of the accidental pressurization of the 24PT1-DSC is presented in Chapter 4.

11.2.8.3 Accident Dose Calculations

There are no dose consequences as the result of the accidental pressurization of the 24PT1-DSC since the DSC confinement boundary is not breached.

11.2.8.4 Corrective Actions

None required since the DSC is designed to maintain confinement under a very conservative postulated pressurization event.

11.2.9 Burial

11.2.9.1 Cause of Accident

The cause of this accident is postulated to be an earthquake or other natural phenomenon resulting in collapse of earthen material onto an AHSM.

11.2.9.2 Accident Analysis

An evaluation was made to determine the increase in 24PT1-DSC temperature with time assuming the AHSM was completely buried in a medium which does not provide the equivalent cooling of natural convection and unrestricted radiation to the environment. The scenario is bounded by the blocked vent scenario addressed in Section 11.2.7.

The results of this analysis show that, if the AHSM is uncovered within 40 hours, there will be no impact to the fuel or the confinement boundary, as discussed above.

11.2.9.3 Accident Dose Calculations

Provided that the AHSM is unburied within 40 hours, there will be no increase in dose rate due to burial. It is reasonable to assume that the AHSM can be unburied before temperatures are reached which would result in 24PT1-DSC confinement barrier failure.

11.2.9.4 Corrective Actions

Material blocking proper AHSM inlet and outlet ventilation is to be removed within 40 hours to ensure restoration of AHSM cooling requirements.

11.2.10 Inadvertent Loading of a Newly Discharged Fuel Assembly

11.2.10.1 Cause of Accident

The possibility of a spent fuel assembly, with a heat generation rate greater than 0.583 kW, being erroneously selected for storage in a 24PT1-DSC has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

11.2.10.2 Accident Analysis

The fuel assemblies require many years of storage in the spent fuel pool before the heat generation decays to a rate below 0.583 kW. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the DSC, with a heat generation rate in excess of the design basis specified in Chapter 2.

In order to preclude this accident from going undetected, and to ensure that appropriate corrective actions can take place prior to the sealing of the DSC, a final verification of the assemblies loaded into the 24PT1-DSC and a comparison with fuel management records is required to assure that the correct assemblies are loaded.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 8.

Appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of 0.583 kW, is not considered credible in view of the multiple administrative controls. There is no thermal or shielding analysis impact since the improperly loaded 24PT1-DSC will not be removed from the fuel pool due to independent review.

A review of Westinghouse 14x14 stainless steel clad UO_2 assemblies and 14x14 zircalloy clad mixed oxide fuel assemblies fabricated to date has confirmed, based on current 24PT1-DSC licensing and fabrication schedules, that the inventory of all fuel assemblies of this type will meet the fuel specification requirements of Section 12 of the SAR. Therefore, inadvertent loading of a fuel assembly exceeding these requirements with respect to enrichment, burnup, decay time and decay heat is not credible.

Although this event is not considered credible, the following evaluation is provided of the consequences of inadvertently loading spent fuel assemblies not allowed by SAR Chapter 12.

The highest burnup fuel assembly in inventory as of January 2001 has a decay heat of approximately 0.7 kW. The actual number of fuel assemblies greater than 0.581 kW in inventory is 13. Assuming these 13 assemblies are 0.7 kW each and the balance of the assemblies are 0.581 kW each and including 24 control components at 0.002 kW each, the maximum canister heat load is less than 16 kW. The analysis at 16 kW provided in Chapter 4 of the SAR indicates that vacuum drying is the only operating condition in which a specified material temperature limit is exceeded (the spacer disc temperature limit is exceeded in this case). Since this is a short term scenario and based on the conservative limits used, the misloading can not impact 24PT1-DSC confinement boundary integrity.

11.2.10.3 Accident Dose Calculations

The inadvertent loading of a fuel assembly not intended for storage in a 24PT1-DSC is not considered to be a credible occurrence. Therefore, no resultant doses would occur.

11.2.10.4 Corrective Actions

If it has been determined that a fuel assembly which is outside the bounds of the design basis has been loaded, it shall be removed from the 24PT1-DSC prior to removing the cask from the fuel pool.

Table 11.2-1
Postulated Accident Loading Identification

Accident Load Type	Section Reference	Advanced NUHOMS® System Component Potentially Affected			
		24PT1-DSC Shell Assembly	24PT1-DSC Internal Basket	24PT1-DSC Support Structure	AHSM
Earthquake	11 2 1	X	X	X	X
Extreme Wind and Tornado Missiles	11 2 2				X
Flood	11.2 3	X			X
Fire/Explosion	11.2.4	X	X		X
Accident Cask Drop	11.2 5	X	X		
Lightning	11.2.6				X
Blockage of Air Inlet and Outlet Openings	11 2.7	X	X	X	X
Accidental Pressurization of the 24PT1-DSC	11 2 8	X			
AHSM Burial	11.2.9	X			X
Inadvertent Loading of a Newly Discharged Fuel Assembly	11.2.10	X	X	X	X

Table 11.2-2
Summary of AHSM Sliding/Uplift Displacements

[redacted]

[redacted]

[redacted]

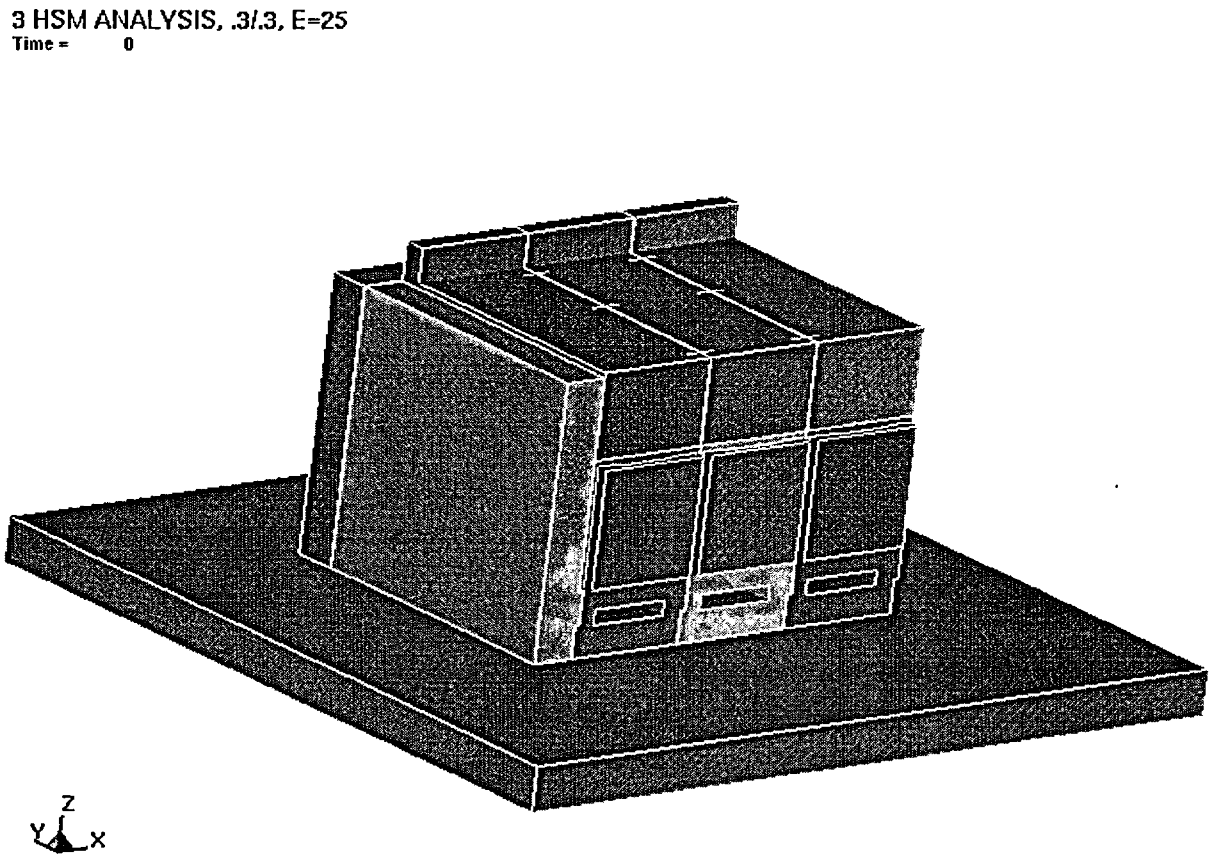


Figure 11.2-1
LS-DYNA AHSM Seismic Stability Model

Figure 11.2-2
Horizontal Time History H1, Set 1 (Taiwan, 1999)

Figure 11.2-3
Horizontal Time History H2, Set 1 (Taiwan, 1999)-

Figure 11.2-4
Vertical Time History V, Set 1 (Taiwan 1999)-

Figure 11.2-5
Horizontal Time History H1, Set 2 (Tabas 1978)-

Figure 11.2-6
Horizontal Time History H2, Set 2 (Tabas 1978)-

Figure 11.2-7
Horizontal Time History V, Set 2 (Tabas 1978)-

Figure 11.2-8
Horizontal Time History H1, Set 3 (Landers/Lucern 1992)-

Figure 11.2-9
Horizontal Time History H2, Set 3 (Landers/Lucern 1992)-

Figure 11.2-10
Vertical Time History V, Set 3 (Landers/Lucern 1992)-

Figure 11.2-11
AHSM Sliding Response X-Direction-

Figure 11.2-12
AHSM Sliding Response Y-Direction-

Figure 11.2-13
AHSM Uplift Response Z-Direction-

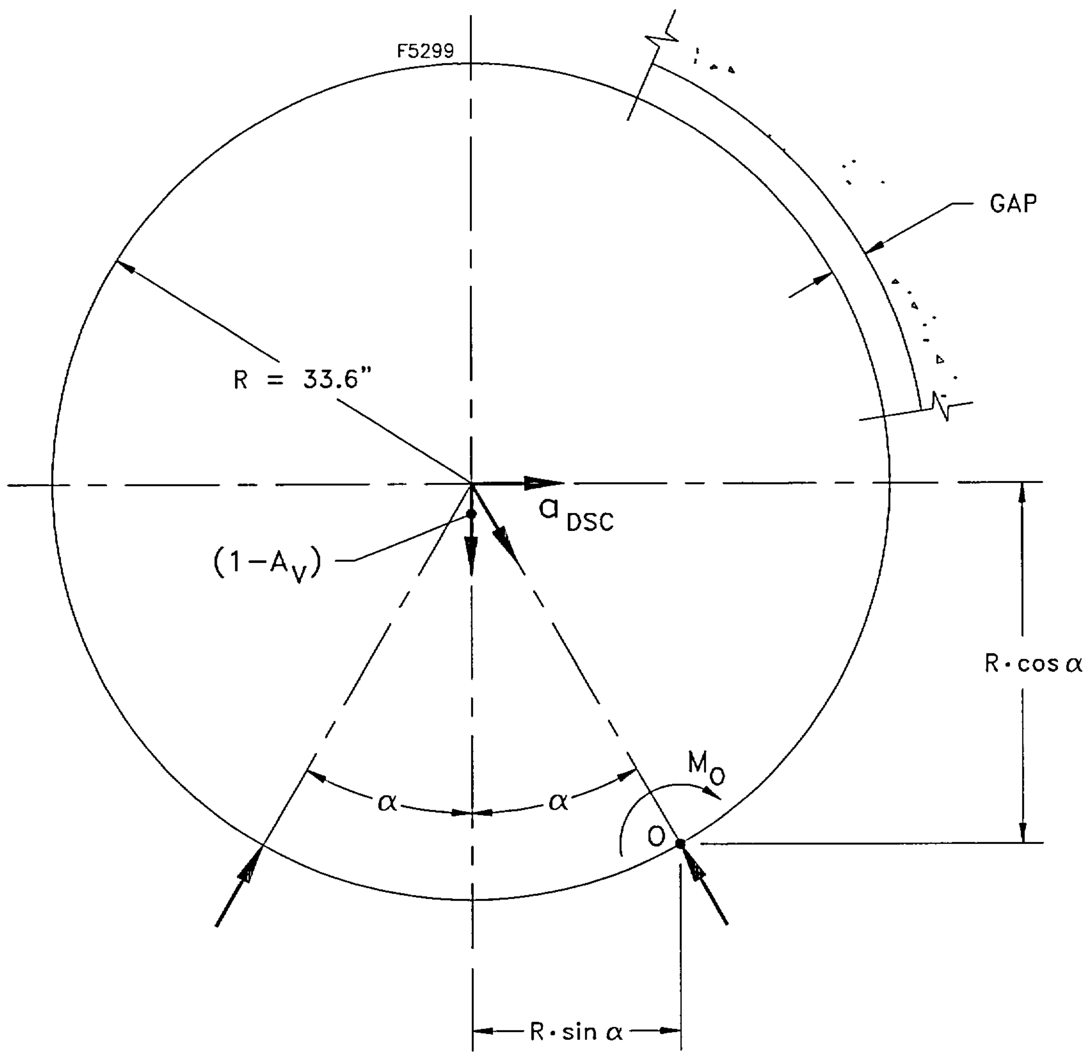


Figure 11.2-14
24PT1-DSC Stability Evaluation

Figure 11.2-15
AHSM Dimension for Missile Impact Stability Analysis-

Figure 11.2-16
Analysis Case 2 (TH1-2): AHSM Sliding Response X-Direction

Figure 11.2-17
Analysis Case 3 (TH2-1): AHSM Sliding Response X-Direction

Figure 11.2-18
Analysis Case 4 (TH2-2): AHSM Sliding Response X-Direction

11.3 Supplemental Information

11.3.1 References

[11.1] American National Standards Institute, American Nuclear Society, ANSI/ANS-57.9-1984, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1984.

[11.2] NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, 1973.

[11.3] LS-DYNA Version 950(C), User's Manual, May 1999, Livermore Software Technology Corporation.

[11.4] NRC NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, July 1981.

[11.5] NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973.

[11.6] Swanson Analysis Systems Inc., ANSYS Engineering Analysis System User's Manual, Version 5.3, Pittsburgh, PA.

[11.7] CFR Title 10, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[11.8] Not used.

[11.9] NRC Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, April 1974.

[11.10] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, ACI 349-97 and ACI 349R-97, American Concrete Institute, Detroit, MI.

[11.11] J. Roark and W. C. Young, Formulas for Stress and Strain, Sixth Edition, McGraw-Hill, New York, N.Y., (1989).

[11.12] "Fluid Mechanics," Raymond C. Binder, 4th Edition, Prentice-Hall, Inc.

[11.13] American Society of Mechanical Engineers, <u>ASME Boiler and Pressure Vessel Code</u>, Section III, 1992 Edition with 1994 Addenda, as amended by Code Case N-595-1.

[11.14] American Society of Civil Engineers, ASCE 7-95, Minimum Design Loads for Buildings and Other Structures, (formerly ANSI A58.1).

[11.15] American Society of Civil Engineers, ASCE Manual No. 58, Structural Analysis and Design of Nuclear Plant Facilities, 1980.

[11.16] TN West Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision *6, November 2001*, USNRC Docket No. 72-1004.

[11.17] "Design of Structures for Missile Impact", BC-TOP-9A, Revision 2, September 1974, Bechtel Power Corporation.

12. OPERATING CONTROLS AND LIMITS

The information previously presented in SAR Chapter 12, Operating Controls and Limits, and the associated bases are contained in Appendices A and B to C of C 1029. Hence, the contents of SAR Chapter 12 are deleted in their entirety to avoid maintenance of duplicate documents.

SAR Chapter 12 requirements are currently referenced in various FSAR chapters and TN documents. Table 12-1 *provides a cross-reference index of these requirements against the corresponding Technical Specification, as listed in Appendix A of C of C 1029.* Table 12-2 *provides a similar cross-reference index of these requirements against the associated bases listed in Appendix B of C of C 1029.*

Table 12-1
Cross-Reference Index of Technical Specifications vs. Historical SAR References

C of C Appendix A, Section No.	Title of Technical Specification	Historical SAR Reference
1.0	Use and Application	12.1.0
1.1	Definitions	12.1.1
1.2	Logical Connectors	12.1.2
1.3	Completion Times	12.1.3
1.4	Frequency	12.1.4
2.0	Functional and Operating Limits	12.2.0
2.1	Fuel To Be Stored In The 24PT1-DSC	12.2.1
2.2	Functional and Operating Limits Violations	12.2.2
3.0	Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) Applicability	12.3.0
3.31	24PT1-DSC and Fuel Cladding Integrity	12.3.1
3.1.1	24PT1-DSC Vacuum Drying Time (Duration) and Pressure	12.3.1.1
3.1.2	24PT1-DSC Helium Backfill Pressure	12.3.1.2
3.1.3	24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent/Siphon Port Cover Welds	12.3.1.3
4.0	Design Features	12.4.0
4.1	Site	12.4.1
4.1.1	Site Location	12.4.1.1
4.2	Storage System Features	12.4.2
4.2.1	Storage Capacity	12.4.2.1
4 2.2	Storage Pad	12.4.2.2
4.2.3	Canister Neutron Absorber	12.4.2.3
4.2.4	Canister Flux Trap Configuration	12.4.2.4
4.2.5	Fuel Spacers	12.4.2.5
4.3	Codes and Standards	12.4.3
4.3.1	Advanced Horizontal Storage Module (AHSM)	12.4.3.1
4.3.2	Dry Shielded Canister (24PT1-DSC)	12.4.3.2
4.3.3	Transfer Cask	12.4.3.3
4.3.4	Exceptions to Codes and Standards	12.4.3.4
4.4	Storage Location Design Features	12.4.4
4.4.1	Storage Configuration	12.4.4.1
4.4.2	Concrete Storage Pad Properties to Limit 24PT1-DSC Gravitational Loadings Due to Postulated Drops	12.4.4.2
4.4.3	Site Specific Parameters and Analyses	12.4.4.3
5.0	Administrative Controls	12.5.0

C of C Appendix A, Section No.	Title of Technical Specification	Historical SAR Reference
5.1	Procedures	12.5.1
5.2	Programs	12.5.2
5.2.1	Safety Review Program	12.5.2.1
5.2.2	Training Program	12.5.2.2
5.2.3	Radiological Environmental Monitoring Program	12.5.2.3
5.2.4	Radiation Protection Program	12.5.2.4
5.2.5	AHSM Thermal Monitoring Program	12.5.2.5
5.3	Lifting Controls	12.5.3
5.3.1	Cask Lifting Heights	12.5.3.1
5.3.2	Cask Drop	12.5.3.2

C of C Appendix A, Table No.	Title of Technical Specification Table	Historical SAR Reference
Table 2-1	Fuel Specifications	Table 12.2-1
Table 2-2	Fuel Dimension and Weights	Table 12.2-2
Table 2-3	Maximum Neutron and Gamma Source Terms	Table 12.2-3
Table 2-4	Fuel Qualification Table	Table 12.2-4

C of C Appendix A, Figure No.	Title of Technical Specification Figure	Historical SAR Reference
Figure 4-1	Minimum Spacer Disc Ligament Widths	Figure 12.4-1

Table 12-2
Cross-Reference Index of Bases for Technical Specifications vs. Historical SAR References

C of C Appendix B, Bases No.	Title of Bases for Technical Specification	Historical SAR Reference
B 2.0	Functional and Operating Limits	B 12.2.0
B 3 0	Limiting Condition for Operation (LCO) Applicability	B 12.3.0
B 3 0	Surveillance Requirement (SR) Applicability	B 12.3.0
B 3 1	24PT1-DSC and Fuel Cladding Integrity	B 12.3.1
B 3.1.1	24PT1-DSC Vacuum Drying Time (Duration) and Pressure	B 12.3.1.1
B 3.1.2	24PT1-DSC Helium Backfill Pressure	B 12.3.1.2
B 3.1.3	24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent / Siphon Port Cover Welds	B 12.3.1.3

13. QUALITY ASSURANCE

TN's Quality Assurance (QA) Program has been established in accordance with the requirements of 10CFR 72, Subpart G [13.1]. The QA Program applies to the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of the Advanced NUHOMS® System and components identified as "important to safety" and "safety related." These components and systems are defined in Chapter 2 of the *F*SAR.

13.1 Introduction

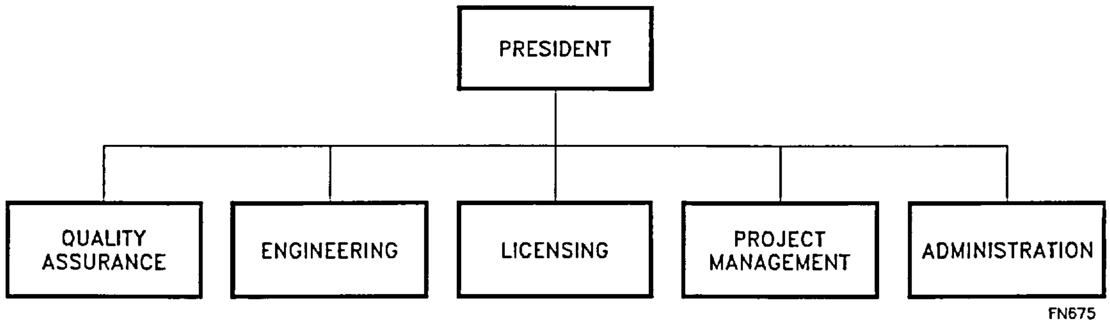
The complete description and specific commitments of the *TN* QA Program are contained in the *TN* QA Manual [13.2]. This manual has been approved by the Nuclear Regulatory Commission (NRC) for performing 10CFR 72 related activities. Changes to the *TN* QA program shall be submitted to the NRC for approval within thirty (30) days of implementation. Changes to the *TN* QA program which decrease or delete previously approved QA commitments shall be submitted to the NRC for approval prior to implementation.

The matrix in Table 13.1-1 shows the 10CFR 72, Subpart G criteria and the respective sections of the *TN* QA Manual and *TN* Quality Procedures Manual [13.3] that address the criteria.

Figure 13.1-1 shows the organization structure for the Advanced NUHOMS® System project.

Table 13.1-1
QA Manual and Quality Procedures Manual

10CFR 72, Subpart G	QA Manual	
.142	1.0	Organization
.144	2.0	QA Program
.146	3.0	Design Control
.148	4.0	Procurement Document Control
.150	5.0	Procedures, Instructions, and Drawings
.152	6.0	Document Control
.154	7.0	Control of Purchased Items and Services
.156	8.0	Identification and Control of Materials, Parts, and Components
.158	9.0	Control of Special Processes
.160	10.0	Inspection
.162	11.0	Test Control
.164	12.0	Control of Measuring and Test Equipment
.166	13.0	Handling, Storage, and Shipping
.168	14.0	Inspection and Test Status
.170	15.0	Control of Nonconforming Items
.172	16.0	Corrective Action
.174	17.0	Records
.176	18.0	Audits



Notes: 1. Licensing may report to Engineering.
2. Administration activities may report to the various other organizations

Figure 13.1-1
Project Organization Chart

13.2 "Important-to-Safety & "Safety Related" Advanced NUHOMS® System Components

TN will apply its QA Program to the Advanced NUHOMS® System components within its scope of responsibility which are defined as "important to safety" and " safety related" as delineated in Section 2.5. QA procedures are used to establish the quality category of components, subassemblies, and piece parts according to each item's importance to safety.

In Section 2.5, each component is identified as "important to safety," "not important to safety," or "safety related". During the design process, items that are considered "important to safety" are further categorized using a graded quality approach. When the graded quality approach is used, a list shall be developed for each "important to safety" item which includes an assigned quality category consistent with the item's importance to safety. Quality categories are determined based on the following and the guidance provided in NUREG/CR-6407 [13.4]:

Category A items are critical to safe operation. These items include structures, components and systems whose failure or malfunction could directly result in a condition adversely affecting public health and safety. This would include conditions such as loss of primary containment with subsequent release of radioactive material, loss of shielding or an unsafe geometry compromising criticality control.

Category B items have a major impact on safety. These items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. An unsafe operation could result only if a primary event occurs in conjunction with a secondary event or other failure or environmental occurrence.

Category C items have a minor impact on safety. These items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would be unlikely to create a condition adversely affecting public health and safety.

For "safety related" items the Quality Assurance Program is applied as described for Category A items. The Quality Assurance Program as described in Section 13.3 is applied to each "important to safety" graded category and is limited as follows.

Category A

A. The design is based on the most stringent industrial codes or standards. Design verification shall be accomplished by prototype testing or formal design review.

B. Vendors for items and services for this category may only be selected from the Approved Suppliers List.

C. *TN* suppliers and sub-tier suppliers must have a QA program based on applicable criteria in Subpart G to 10CFR 72, or equivalent.

D. Complete traceability of raw materials and the use of certified welders and processes is required.

E. All personnel performing Quality Assurance related inspections, tests, and examinations shall be qualified and certified in accordance with the requirements of the QA program.

F. Only qualified and certified auditors and lead auditors shall perform audits.

G. *TN* QA personnel shall be required to inspect and/or approve supplier fabricated components prior to authorizing shipment release.

H. Welding consumables shall be procured as a Category A item if the intended use is unknown. If purchased for a specific Category B or C application, the material must be identified and its use restricted to fabrication of the same level.

Category B

A. The design is based on the most stringent industrial codes and standards. But design verification may be accomplished by use of alternate calculations or computer codes.

B. The procurement of items may be from suppliers on the Approved Suppliers List or QA program requirements for the supplier may be based upon the inspection and test requirements of the procured item.

C. Traceability of materials is not required; however, specified welds require completion by qualified, certified welders.

D. Quality Assurance verification activities shall be performed by personnel qualified and certified in accordance with the requirements of the QA program.

E. Only lead auditor personnel require certification in accordance with the QA program.

Category C

A. Items may be purchased from a catalog or "off-the-shelf."

B. When received, the item shall be identified and checked for compliance with the purchase order and for damage.

Items not considered important-to-safety will be controlled in accordance with good industrial practices.

If a utility elects to perform construction, and has an NRC approved QA program (10CFR 50 [13.5]) that is equivalent to or exceeds *TN*'s program, then the utility QA program is considered an acceptable substitute for their scope of responsibility.

13.3 Description of *TN* 10CFR 72, Subpart G QA Program

13.3.1 Project Organization

The Advanced NUHOMS® System has been designed by a dedicated *TN* project organization.

QA duties are performed by the *TN* project organization, the QA Manager, and QA Engineers.

The organization structure for the Advanced NUHOMS® System project is presented in Figure 13.1-1. A description of *TN*'s organizational structure, functional responsibilities, levels of authority, and lines of internal and external (client and supplier) communication may be found in the *TN* QA Manual.

Project QA controls are determined by the Project Manager and approved by the QA Manager. All Project Plans, regardless of the indicated applicability of QA requirements, are reviewed by the QA Manager to assure that QA controls are commensurate with the specific activity, item complexity, importance to safety and client-imposed contractual requirements.

Project personnel are indoctrinated, trained, and qualified in accordance with the *TN* QA Manual.

13.3.2 QA Program

TN will apply the QA Program to components defined in Section 2.5 as "important to safety" and "safety related" in accordance with the *TN* QA Manual.

TN has established and implemented a QA program for the control of quality in the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of storage containers for nuclear products. Training and/or evaluation of personnel qualifications in accordance with written procedures are required for personnel performing activities affecting quality. The QA program assures that all quality requirements, engineering specifications and specific provisions of any package design approval are met. Those characteristics critical to safety are emphasized.

The *TN* QA Manager regularly evaluates the *TN* QA program for adherence to the 18 point criteria in scope, implementation and effectiveness. Further, the *TN* President requires that the QA Program, including the QA Manual Policies and Procedures, be implemented and enforced on all applicable projects at *TN*.

13.3.3 Design Control

"Important to safety" and "safety related" Advanced NUHOMS® System design activities shall be implemented in accordance with the *TN* QA Manual. Design verification will be performed by a competent individual with the appropriate skill level. However, this individual's skill level may not be the same as the originator but must be equivalent.

Errors and deficiencies in the design, including the design process, are documented in the form of Corrective Action Reports.

Industry standards and specifications are used for the selection of suitable materials, parts, equipment and processes for "important to safety" and "safety related" structures, systems, or components as defined in the various chapters and sections of this SAR.

13.3.4 Procurement Document Control

Procurement documents are prepared in accordance with the *TN* QA Manual which delineates the actions to be accomplished in the preparation, review, approval, and control of procurement documents. Review and approval of procurement documents by the QA Manager are documented on the procurement documents prior to release to assure the adequacy of quality requirements stated therein. This review determines that quality requirements are correctly stated, inspectable, and controllable; that there are adequate acceptance and rejection criteria; and that the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements. Refer to Section 13.2 for supplier selection requirements.

The procurement documents shall identify the documentation required to be submitted for information, review, or approval by *TN* or *TN*'s client. The time of submittal shall also be established. When *TN* requires the supplier to maintain specific QA records, the retention times and disposition requirements shall be prescribed.

13.3.5 Procedures, Instructions, and Drawings

As required by the *TN* QA Manual, activities affecting quality are prescribed in approved, written procedures, instructions, or drawings and these procedures, instructions, and drawings shall be followed.

13.3.6 Document Control

The issuance, distribution, and receipt of documents which prescribe activities affecting quality are controlled in accordance with the *TN* QA Manual. Controlled documents include, but are not limited to, the *TN* design specifications and criteria documents, drawings, instructions, and test procedures.

The individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto are identified in the "Responsibilities" sections of the *TN* QA Manual.

13.3.7 Control of Purchased Items and Services

The control of purchased items and services shall be implemented in accordance with the *TN* QA Manual.

Surveillance of subcontracted activities is planned and performed in accordance with written procedures to assure conformance to the purchase order. These procedures provide for instructions that specify the characteristics to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions.

TN suppliers shall furnish documentation that identifies any procurement requirements which have not been met, together with a description of those nonconformances dispositioned as "use-as-is" or "repair."

Documentation from *TN* suppliers which demonstrates compliance with procurement requirements (such as material test reports, NDE results, performance test results, etc.) is periodically evaluated by audits, independent inspections, or tests as necessary to assure its validity.

13.3.8 Identification and Control of Materials, Parts, and Components

Materials, parts, and components shall be identified and controlled in accordance with the *TN* QA Manual. Hardware identification requirements are determined during generation of design drawings and specifications such that the location and method of identification do not affect the form, fit, function, or quality of the item being identified.

13.3.9 Control of Special Processes

The control of special processes, such as nondestructive examination, chemical cleaning, welding, and heat treating shall be performed in accordance with the *TN* QA Manual.

13.3.10 Inspection

Receipt inspections, and in-process and final inspections of *TN*-fabricated, constructed, or erected items, systems, components, or structures shall be performed in accordance with the *TN* QA Manual.

13.3.11 Test Control

Test control shall be accomplished in accordance with the *TN* QA Manual.

13.3.12 Control of Measuring and Test Equipment

The *TN* QA Manual defines the requirements for calibration of measuring and test equipment. Calibration is against certified measurement standards which have known relationships to national standards, where such standards exist. Where such standards do not exist, the basis for calibration shall be documented.

13.3.13 Handling, Storage and Shipping

Handling, storage, and shipping shall be conducted in accordance with the *TN* QA Manual. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by qualified individuals in accordance with predetermined work and inspection instructions.

13.3.14 Inspection and Test Status

The use of inspection and test status tags shall be implemented in accordance with the *TN* QA Manual.

13.3.15 Control of Nonconforming Items

The *TN* QA Manual defines the requirements and assigns the responsibilities for the control, identification, segregation, documentation, and close-out of nonconforming items to prevent their inadvertent installation or use in fabrication, construction, or erection.

Nonconformance reports identify the item description and quantity, the disposition of the nonconformance, the inspection requirements, and signature approval of the disposition. They are periodically analyzed to show quality trends and help identify root causes of nonconformances. Significant results are reported to responsible management for review and assessment.

Nonconforming items are segregated from acceptable items and tagged to prevent inadvertent use until properly dispositioned and closed out.

13.3.16 Corrective Action

Corrective action for conditions adverse to quality shall be taken in accordance with the *TN* QA Manual. For significant conditions adverse to quality the cause is determined and action to preclude recurrence is taken and reported to the appropriate levels of management.

13.3.17 Records

The *TN* QA Manual defines the scope of the records program such that sufficient records are maintained to provide documentary evidence of the quality of items and activities affecting quality.

13.3.18 Audits and Surveillances

A comprehensive system of planned and documented audits, including audits of suppliers and site construction activities, verifies compliance with all aspects of the *TN* QA Program and determines the effectiveness of the program.

Audits are performed by certified lead auditors and are planned, performed, and documented in accordance with the *TN* QA Manual.

Unannounced QA surveillances may be performed on activities affecting quality by the *TN* QA Manager, or his designee, on an as-needed basis to further assure compliance with QA requirements.

13.4 Conditions of Approval Records

As required by 10CFR 72, Subpart L, *TN* will establish and maintain records for each storage component fabricated under a certificate of compliance as required by §72.234(d). The records will be available for inspection as required by §72.234(e). Written procedures and appropriate tests will be established prior to use of the storage components which will be provided to each Advanced NUHOMS® System user as required by §72.234(f).

13.5 Supplemental Information

13.5.1 References

[13.1] CFR Title 10, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[13.2] "Transnuclear Quality Assurance Manual," current revision.

[13.3] "Transnuclear Quality Assurance Procedures Manual," current revision.

[13.4] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[13.5] CFR Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities.

14. DECOMMISSIONING

14.1 Decommissioning Considerations

The Advanced NUHOMS® System design features inherent ease and simplicity for decommissioning by providing easily decontaminable surfaces and isolating the external surfaces of the 24PT1-DSC from contact with the fuel pool. At the end of its service life, the 24PT1-DSC decommissioning could be performed by one of the options listed below:

- Option 1, the 24PT1-DSC, including stored spent fuel, could be shipped to either a monitored retrievable storage system (MRS) or a geological repository for final disposal, or

- Option 2, the spent fuel could be removed from the 24PT1-DSC (either at the ISFSI site or at another off site location) and shipped in an NRC approved transportation cask.

The first option requires that the 24PT1-DSC be upgraded to current Part 71 regulations. An amendment to C of C 71-9255 [14.2] *has been approved by the NRC* to allow for transport of this 24PT1-DSC using the MP187 cask.

The first option does not require any decommissioning of the 24PT1-DSC. No residual contamination is expected to be left behind on the concrete AHSM. The AHSM, fence, and peripheral utility structures will require no decontamination or special handling after the last 24PT1-DSC is removed. The AHSM, fence, and peripheral utility structures could be demolished and recycled with normal construction techniques.

The second option would require decontamination of the 24PT1-DSC and transfer cask (if applicable). The sources of contamination in the interior of the 24PT1-DSC or transfer cask would be the primary contamination left from the spent fuel pool water; or crud, hot particles and fines from the spent fuel pins. This contamination could be removed with a high pressure water spray. If further surface decontamination of the 24PT1-DSC or transfer cask is necessary, electropolishing or chemical etching can be used to clean the contaminated surface. After decontamination, the 24PT1-DSC and/or transfer cask could be cut up for scrap, partially scrapped, or refurbished for reuse. Any activated metal would be shipped as low level radioactive waste to a near surface disposal facility.

A review of cask activation analyses previously performed for similar systems (TN-32 cask [14.4] and NUHOMS® site license storage system) indicates that the levels of activation of the 24PT1-DSC, AHSM and transfer cask would be orders of magnitude below the specific activity of the isotopes listed in Tables 1 and 2 of 10CFR 61.55 [14.3]. A detailed analysis is not considered necessary based on the significant margins determined from these analyses. A comparison of the source terms for this application to those referenced above including the activation analysis summary for the above applications is provided below:

Comparison of Source Terms for Activation Analyses

Source Term (including Control Components)	24PT1-DSC	TN-32 (Metal Cask)	NUHOMS® Site License HSM
γ (γ/sec/assy)	3.4×10^{15}	$5\ 3 \times 10^{15}$	1.53×10^{15}
n (n/sec/assy)	2.8×10^{8}	3.3×10^{8}	2.23×10^{8}

TN 32 and NUHOMS® Site License HSM Activation Analysis Results

Nuclide	Activity Ci/m^3			
	HSM Concrete	HSM Steel	TN-32	10CFR 61.55 Limit
H-3			8.3×10^{-11}	40
C-14			2.3×10^{-10}	8
Co-60	4.4×10^{-5}	8.1×10^{-2}	7.7×10^{-6}	700
Ni-59	1.4×10^{-10}	3.1×10^{-6}	2.5×10^{-6}	220
Ni-63	8.3×10^{-8}	3.2×10^{-4}	3.4×10^{-4}	3.5
Nb-94		3.9×10^{-8}		.2
<5 year half life	4.6×10^{-3}	2.0×10^{-1}	2.3×10^{-2}	700

Following surface decontamination, the radiation levels in the 24PT1-DSC or transfer cask due to activation will be below the acceptable limits of Regulatory Guide 1.86 [14.1]. The activation levels of the 24PT1-DSC or transfer cask materials will be far below the specific activity limits for both short and long lived nuclides for Class A waste. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal, should refurbishment not be elected.

The procedure for decommissioning a 24PT1-DSC or transfer cask not being returned to service is summarized below:

- Remove fuel in accordance with the unloading procedures of Chapter 8.

- Survey interior of 24PT1-DSC or transfer cask. Wash down the inside of the 24PT1-DSC or transfer cask. Pump out and filter contaminated water and cleaning agent. Survey interior of 24PT1-DSC or transfer cask again, decontaminate as required. It is expected that surface decontamination will be minimal. If so, dispose of the 24PT1-DSC or transfer cask body as scrap metal. If unable to decontaminate to acceptable levels, the 24PT1-DSC and/or transfer cask body can be disposed of as low level radioactive waste.

- Decontaminate the top inner and outer cover plates until able to dispose of as scrap metal. If unable to achieve acceptable levels, dispose of them as low level radioactive waste.

The fuel unloading and decontamination steps for 24PT1-DSC, AHSM, or cask refurbishment are as outlined for the scrap choices, discussed above. However, the only pieces discarded are components damaged by unloading or that are considered to be difficult to decontaminate. Following a comprehensive survey to confirm continued 24PT1-DSC, AHSM or transfer cask functionality within design basis, the components will be eligible for returning to spent fuel storage service.

The volume of waste material produced incidental to ISFSI decommissioning is expected to be limited to that necessary to accomplish surface decontamination of the 24PT1-DSCs, if the spent fuel elements are removed. No chemical or mixed waste is anticipated. The licensee is responsible for the disposal of any waste generated by decontamination. Furthermore, it is estimated that the 24PT1-DSC materials will be slightly activated as a result of their long term exposure to the relatively small neutron flux emanating from the spent fuel, and that the resultant activation level will be well below the allowable limits for general release as noncontrolled material. Therefore, it is anticipated that the 24PT1-DSCs may be decommissioned from nuclear service by surface decontamination alone. This activity could be performed at the utility, or other suitable facility.

A detailed decommissioning plan will be submitted prior to the commencement of decommissioning activities. The costs of decommissioning the ISFSI are expected to represent a small and negligible fraction of the cost of decommissioning a nuclear power station.

14.2 Supplemental Informational

14.2.1 References

[14.1] Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors."

[14.2] *NUHOMS® MP187 Multipurpose Cask Transportation Package, Certificate of Compliance No. 9255, Revision 8, Docket Number 71-9255.*

[14.3] U.S. Nuclear Regulatory Commission, Title 10 Code of Federal Regulations, Part 61, "Licensing Requirements for Land Disposal or Radioactive Waste".

[14.4] Safety Analysis Report for the TN-32 Cask, Docket 72-1021, Revision 0, January 2000.