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,

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

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,

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

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

The source area of the side is,

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

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

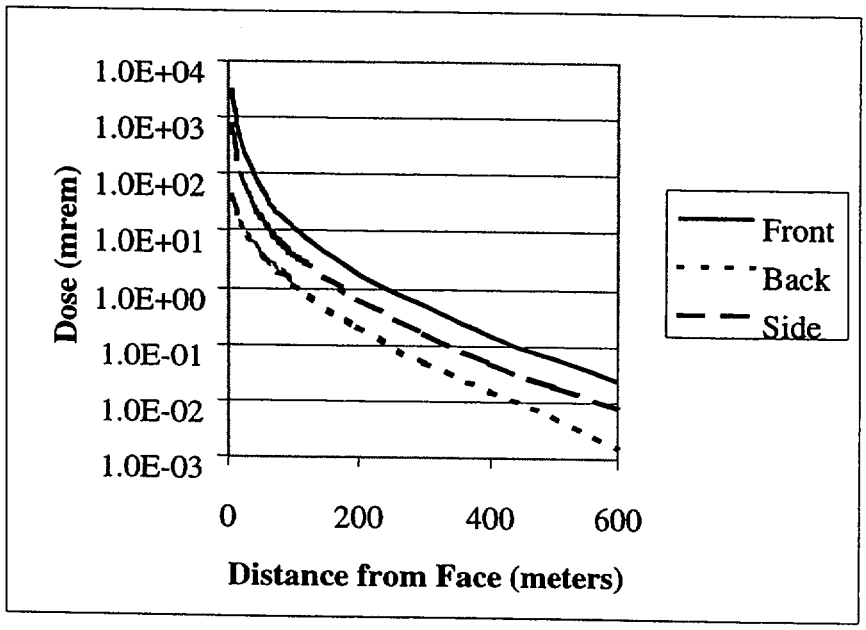


Figure 10.2-1
Total Annual Exposure from a Single AHSM as a Function of Distance

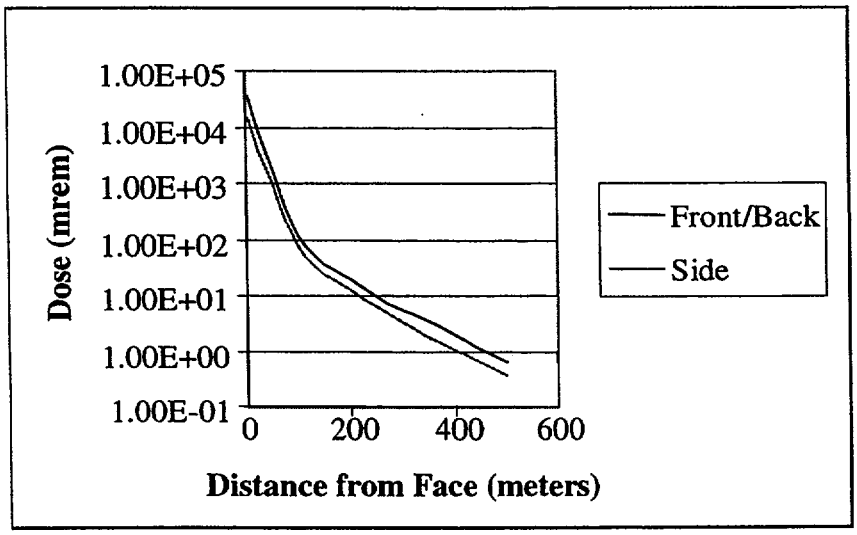


Figure 10.2-2
Total Annual Exposure from a 2x10 AHSM Array as a Function of Distance

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. 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 24PT1-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.

11.2.1.3 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]:

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

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

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]:

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

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

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]:

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

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

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:

$$v_R^{1.8} = v_I^{1.8} - v_P^{1.8}$$

where v_R = Residual missile velocity

v_I = Missile impact velocity ($=v_o$)

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

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

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

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

Therefore, $v_R^{1.8} = 205^{1.8} - 198.6^{1.8}$

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

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-sec^2/ft

v_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 travelling 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-sec^2/in = mass of the missile

M = (318.74+77.0+2*163.0)*1000/386.4 = 1868 lb-sec^2 /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)

$I_A = 5.17 \times 10^7$ lb-sec^2-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.

Overturning:

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:

$$M_{st} = 239.3x50.5 + 95.2x(101+18)$$

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

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:

$$M_{ot} = 8.07 \text{ K/ft. x } 19.583 \text{ ft. x } (18.5x12/2)$$

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

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 24PT1-DSC Flooding Analyses

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 Thermal Evaluation of Flood Accident

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

11.2.3.3 Accident Dose Calculations

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 Corrective Action

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 24PTl-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 Fire/Explosion

11.2.4.1 Cause of the Accident

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 11.1.1).

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

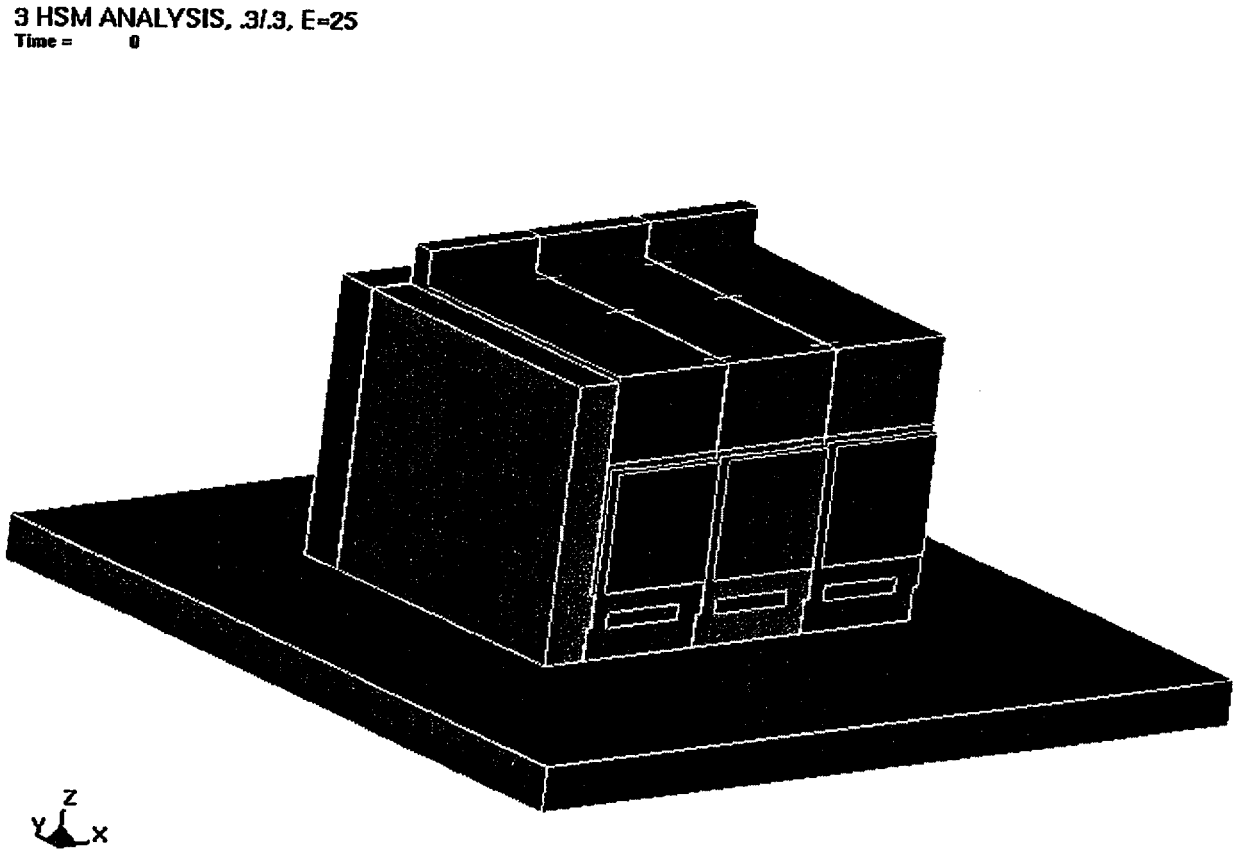


Figure 11.2-1
LS-DYNA AHSM Seismic Stability Model

[redacted]

[redacted]

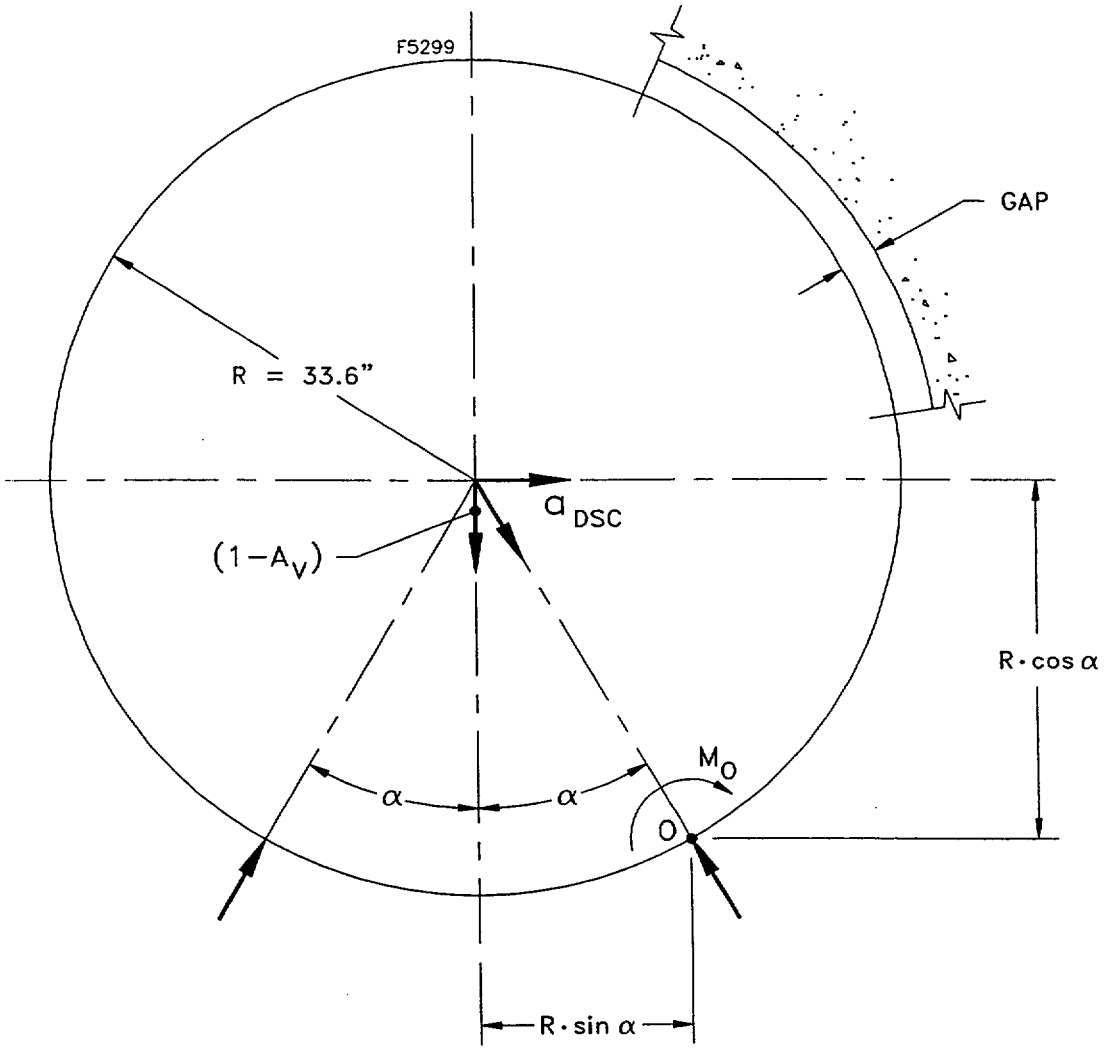
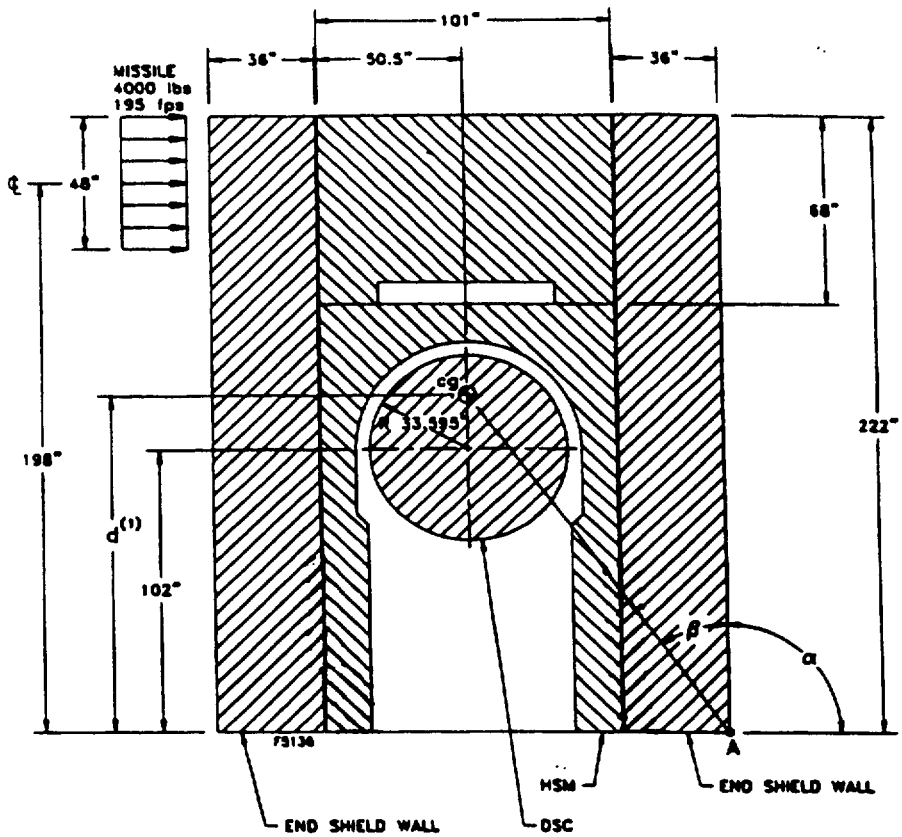


Figure 11.2-14
24PT1-DSC Stability Evaluation



Note:

(1) d = 121.42" with minimum DSC weight of 77 kips. (α is the angle between the side of the rotated AHSM and the horizontal)

Figure 11.2-15
AHSM Dimension for Missile Impact Stability Analysis

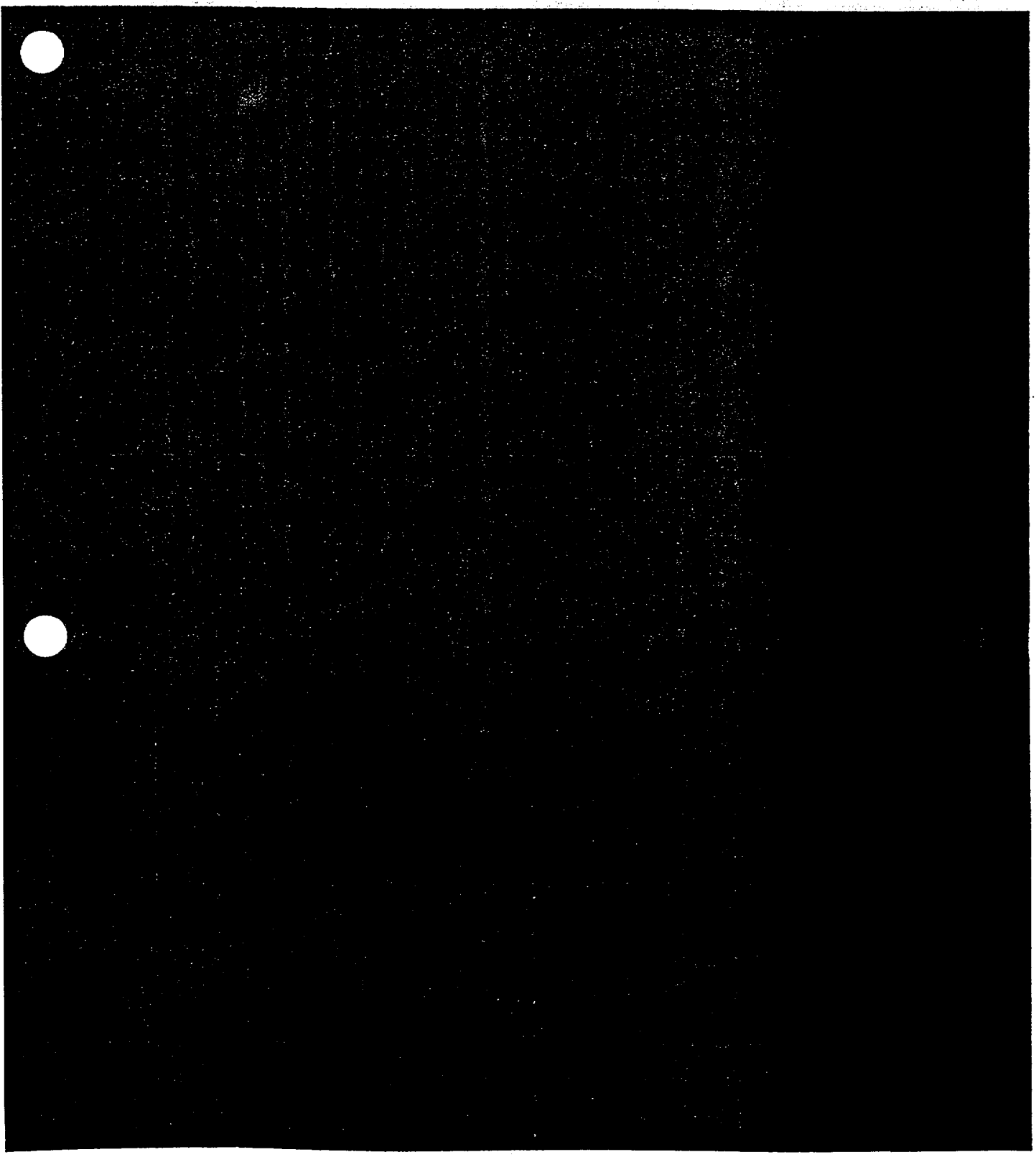


Figure 11.2-16
Analysis Case 2 (TH1-2): AHSM Sliding Response X-Direction

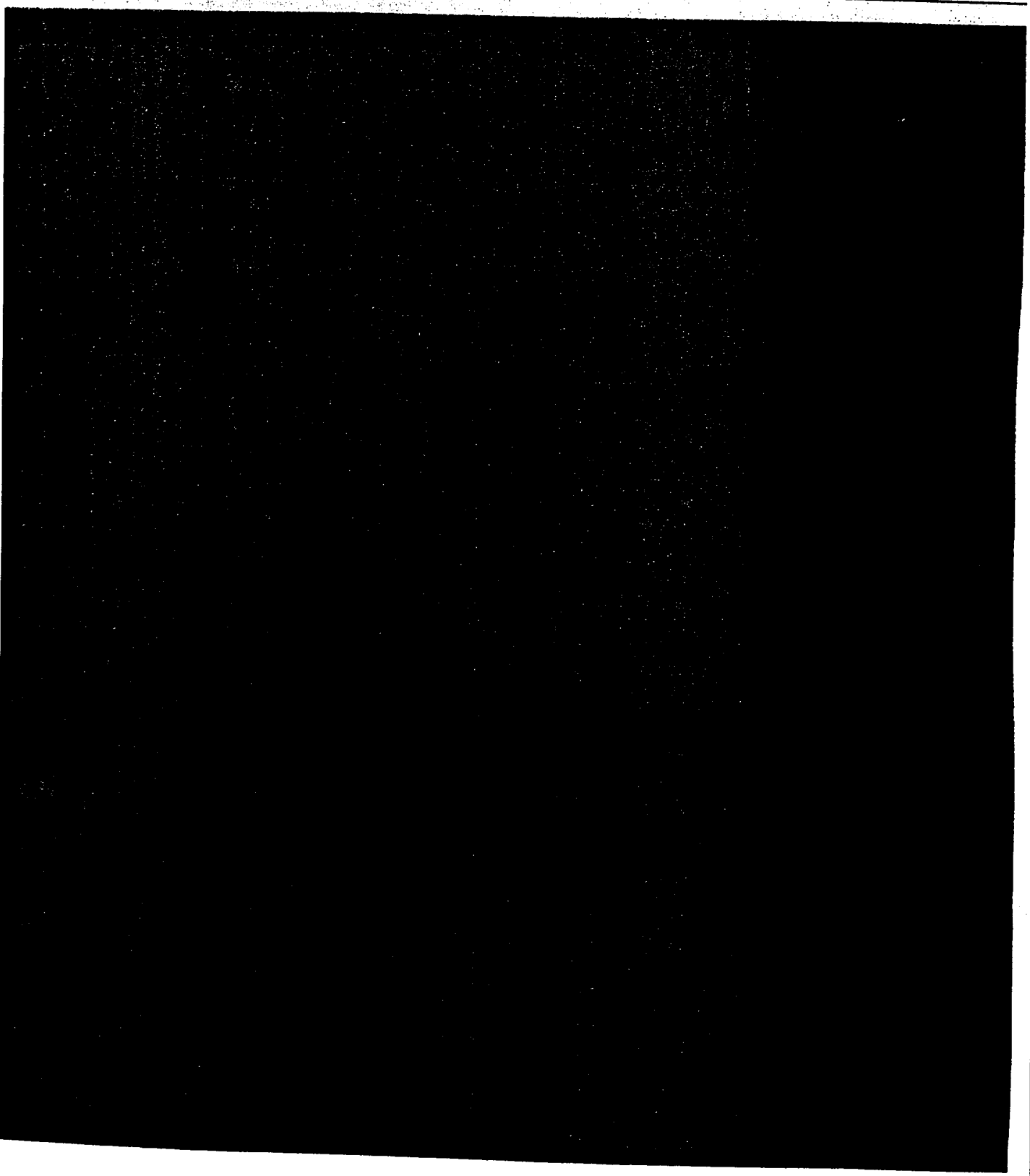


Figure 11.2-17
Analysis Case 3 (TH2-1): AHSM Sliding Response X-Direction

[redacted]

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, ASME Boiler and Pressure Vessel Code, 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] Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 5, August 2000, USNRC Docket No. 72-1004.

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

TECHNICAL SPECIFICATIONS FOR THE ADVANCED NUHOMS® SYSTEM

TABLE OF CONTENTS

12. OPERATING CONTROLS AND LIMITS

12.1.0 Use and Application

12.1.1 Definitions

----------NOTE ----------

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ADVANCED HORIZONTAL STORAGE MODULE (AHSM)	The AHSM is a reinforced concrete structure for storage of a loaded 24PT1-DSC at a spent fuel storage cask.
DAMAGED FUEL ASSEMBLY	A DAMAGED FUEL ASSEMBLY is a fuel assembly with known or suspected cladding defects greater than pinhole leaks or hairline cracks.
DRY SHIELDED CANISTER (24PT1-DSC)	A 24PT1-DSC is a welded pressure vessel that provides confinement of INTACT or DAMAGED FUEL ASSEMBLIES in an inert atmosphere.
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within a perimeter fence licensed for storage of spent fuel within AHSMs.
INTACT FUEL ASSEMBLY	Spent Nuclear Fuel Assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means.
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a 24PT1-DSC while it is being loaded with INTACT or DAMAGED FUEL ASSEMBLIES, and on a TRANSFER CASK while it is being loaded with a 24PT1-DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. LOADING OPERATIONS begin when the first INTACT or DAMAGED FUEL ASSEMBLY is placed in the

	24PT1-DSC and end when the TRANSFER CASK is ready for TRANSFER OPERATIONS.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while a 24PT1-DSC containing INTACT or DAMAGED FUEL ASSEMBLIES is located in an AHSM on the storage pad within the ISFSI perimeter.
TRANSFER CASK (TC)	The TRANSFER CASK will consist of a licensed NUHOMS® OS197 onsite transfer cask. The TRANSFER CASK will be placed on a transfer trailer for movement of a 24PT1-DSC to the AHSM.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involving the movement of a TRANSFER CASK loaded with a 24PT1-DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. TRANSFER OPERATIONS begin when the TRANSFER CASK is placed on the transfer trailer following LOADING OPERATIONS and end when the 24PT1-DSC is located in an AHSM on the storage pad within the ISFSI perimeter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a 24PT1-DSC to unload INTACT or DAMAGED FUEL ASSEMBLIES. UNLOADING OPERATIONS begin when the 24PT1-DSC is removed from the AHSM and end when the last INTACT or DAMAGED FUEL ASSEMBLY has been removed from the 24PT1-DSC.

12.1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors:

EXAMPLE 1.2-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

12.1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

12.1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO are not met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

12.1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions:

EXAMPLE 1.3-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2	12 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

12.1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	12 hours 36 hours

When a system is determined to not meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

EXAMPLES (continued)

EXAMPLE 1.3-3:

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

12.1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "Specified Frequency" is referred to throughout this section and each of the Specifications of Section 12.3, Surveillance Requirement (SR) Applicability. The "Specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 12.3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With a SR satisfied, SR 12.3.0.4 imposes no restriction.

12.1.4 Frequency

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified:

EXAMPLE 1.4-1:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify Pressure within limit.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 12.3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 12.3.0.1 (such as when the equipment is determined to not meet the LCO, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 12.3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 12.3.0.1.

If the interval as specified by SR 12.3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 12.3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 12.3.0.4.

12.1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one-time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 12.3.0.2.

"Thereafter" indicates future performances must be established per SR 12.3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

12.1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------NOTE------- Not required to be met until 96 hours after verifying the helium leak rate is within limit.	Once after verifying the helium leak rate is within limit.
Verify 24PT1-DSC vacuum drying pressure is within limit.	

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the vacuum drying pressure not be met immediately following verification of the helium leak rate while in LOADING OPERATIONS, this Note allows 96 hours to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency."

Once the helium leak rate has been verified to be acceptable, 96 hours, plus the extension allowed by SR 12.3.0.2, would be allowed for completing the Surveillance for the vacuum drying pressure. If the Surveillance was not performed within this 96 hour interval, there would then be a failure to perform the Surveillance within the specified Frequency, and the provisions of SR 12.3.0.3 would apply.

12.2.0 Functional and Operating Limits

12.2.1 Fuel To Be Stored In The 24PT1-DSC

The spent nuclear fuel to be stored in each 24PT1-DSC/AHSM at the ISFSI shall meet the following requirements:

a. Fuel shall be INTACT FUEL ASSEMBLIES or DAMAGED FUEL ASSEMBLIES. DAMAGED FUEL ASSEMBLIES shall be placed in screened confinement cans (failed fuel cans) inside the 24PT1-DSC guidesleeves. Damaged fuel assemblies shall be stored in outermost guidesleeves located at the 45, 135, 225 and 315 degree azimuth locations.

b. Fuel types shall be limited to the following:

   UO_2 Westinghouse 14x14 (WE 14x14) Assemblies (with or without IFBA fuel rods), as specified in Table 12.2-1.

   WE 14x14 Mixed Oxide (MOX) Assemblies, as specified in Table 12.2-1

   Fuel burnup and cooling time is to be consistent with the limitations specified in Table 12.2-4 for UO2 fuel.

   Control Components stored integral to WE 14x14 Assemblies in a 24PT1-DSC, shall be limited to Rod Cluster Control Assemblies (RCCAs), Thimble Plug Assemblies (TPAs), and Neutron Source Assemblies (NSAs). Location of control components within a 24PT1-DSC shall be selected based on criteria which does not change the radial center of gravity by more than 0.1 inches.

c. The maximum heat load for a single fuel assembly, *including control components,* is 0.583 kW. The maximum heat load per 24PT1-DSC, including any integral Control Components, shall not exceed 14 kW.

d. *Fuel can be stored in the 24PT1-DSC in any of the following configurations:*

   1) A maximum of 24 INTACT WE 14x14 MOX or SC fuel assemblies; or

   2) Up to four WE 14x14 SC DAMAGED FUEL ASSEMBLIES, with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES; or

   3) One MOX DAMAGED FUEL ASSEMBLY with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES.

   A 24PT1-DSC containing less than 24 fuel assemblies may contain dummy fuel assemblies in fuel assembly slots. The requirements of Section 12.2.1.b are also applicable.

No more than two empty fuel assembly slots are allowed in each DSC. They must be located at symmetrical locations about the 0-180° and 90-270° axes.

No more than 14 fuel pins in each assembly may exhibit damage. A visual inspection of assemblies will be performed prior to placement of the fuel in the 24PT1-DSC, which may then be placed in storage or transported anytime thereafter without further fuel inspection.

e. Fuel dimensions and weights are provided in Table 12.2-2.

f. The maximum neutron and gamma source terms are provided in Table 12.2-3.

12.2.2 Functional and Operating Limits Violations

If any Functional and Operating Limit of 12.2.1 is violated, the following actions shall be completed:

12.2.2.1 The affected fuel assemblies shall be placed in a safe condition.

12.2.2.2 Within 24 hours, notify the NRC Operations Center.

12.2.2.3 Within 30 days, submit a special report which describes the cause of the violation and the actions taken to restore compliance and prevent recurrence.

Table 12.2-1 Fuel Specifications

Fuel Type	Maximum Initial Enrichment	Cladding Material	Minimum Cooling Time	Minimum Initial Enrichment	Maximum Burnup
UO_2 WE 14x14 (with or without IFBA fuel rods)	4.05 weight % U-235	Type 304 Stainless Steel	10 years	*See Table 12.2-4 for Enrichment, Burnup, and Cooling Time Limits.*	
WE 14x14 MOX	*2.84 weight % Fissile Pu - 64 rods* *3.10 weight % Fissile Pu - 92 rods* *3.31 weight % Fissile Pu - 24 rods*	Zircalloy-4	20 years	*2.74 weight % Fissile Pu –64 rods* *3.01 weight % Fissile Pu –92 rods* *3.21 weight % Fissile Pu – 24 rods*	25,000 MWd/MTU
Integral Control Components	N/A	N/A	10 years	N/A	N/A

Table 12.2-2 Fuel Dimension and Weights

Parameter	WE 14x14 SC[1]	WE 14x14 MOX[1]
Number of Rods	180	180
Number of Guide Tubes/Instrument Tubes	*16*	*16*
Cross Section (in)	7.763	7.763
Unirradiated Length (in)	138.5	138.5
Fuel Rod Pitch (in)	0.556	0.556
Fuel Rod O.D. (in)	0.422	0.422
Clad Material	Type 304 SS	Zircaloy-4
Clad Thickness (in)	0.0165	0.0243
Pellet O.D. (in)	0.3835	0.3659
Max. initial ^{235}U Enrichment (%wt)	4.05	Note 2
Theoretical Density (%)	93-95	91
Active Fuel Length (in)	120	119.4
Max. U Content (kg)	375	Note 3
Ave. U Content (kg)	366.3	Note 3
Assembly Weight (lbs)	1210	1150
Max. Assembly Weight incl. NFAH[4] (lbs)	1320	1320

(1) Nominal values shown unless stated otherwise
(2) Mixed-Oxide assemblies with 0.71 weight % U-235 and *maximum* fissile Pu weight of 2.84 weight % (64 rods), 3.10 weight % (92 rods), and 3.31 weight % (24 rods)
(3) Total weight of Pu is 11.24 kg and the total weight of U is 311.225 kg
(4) Weights of TPAs and NSAs are enveloped by RCCAs

Table 12.2-3 Maximum Neutron and Gamma Source Terms

Parameter	WE 14x14 SC	WE 14x14 MOX
Gamma Source (γ/sec/assy)	3.43E+15	9.57E+14
Neutron Source (n/sec/assy)	2.84E+08	4.90E+07

Parameter	RCCAs	TPAs	NSAs
Gamma Source (γ/sec/assy)	7.60E+12	5.04E+12	1.20E+13
Decay heat (Watts)	1.90	1.2	1.66

Table 12.2-4 Fuel Qualification Table

(Minimum required years of cooling time after reactor core discharge)

Burnup GWd/MTU	Initial Enrichment (weight % U-235)			
	3.12	3.36	3.76	3.96
45.0	Not Analyzed		15.2	15.2*
43.3			15.2	11.5
40.0		10.9	10.9**	10.9**
36.8		10.9	10.0***	10.0***
35.0 or less	10.0***	10.0***	10.0***	10.0***

Notes

* *Cooling time based on 3.76 weight % enrichment is conservatively used.*

** *Cooling time based on 3.36 weight % enrichment is conservatively used.*

*** *Cooling time based on shielding analysis source term.*

General Notes:

- *Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Example: An assembly with an initial enrichment of 3.90 w/o U-235 and a burnup of 37 GWd/MTU is acceptable for storage after a 10.9 year cooling time as defined at the intersection of 3.76 weight % U-235 (rounding down) and 40 GWd/MTU (rounding up) on the qualification table.*

12.3.0 Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) Applicability

LCO 12.3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 12.3.0.2.
LCO 12.3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 12.3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 12.3.0.3	Not applicable to a spent fuel storage cask.
LCO 12.3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS, or that are related to the unloading of a 24PT1-DSC. Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered allow operation in the specified condition in the Applicability only for a limited period of time.
LCO 12.3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 12.3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate that the LCO is met.
LCO 12.3.0.6	Not applicable to a spent fuel storage cask.
LCO 12.3.0.7	Not applicable to a spent fuel storage cask.

SR 12.3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 12.3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 12.3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 12.3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 12.3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of a 24PT1-DSC.

12.3.1 24PT1-DSC Integrity

12.3.1.1 24PT1-DSC Vacuum Drying Time (Duration) and Pressure

LCO 12.3.1.2 Duration: Vacuum Drying of the 24PT1-DSC shall be achieved within the following time durations after the start of bulk water removal (blowdown):

Heat Load (kW)	Time Limit
kW ≤ 12	No limit
12< kW ≤ 13	71 Hours
13< kW ≤ 14	54 Hours

Pressure: The 24PT1-DSC vacuum drying pressure shall be sustained at or below 3 Torr (3 mm Hg) absolute for a period of at least 30 minutes following stepped evacuation.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------- NOTE ----------
This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 24PT1-DSC vacuum drying pressure limit not met within 47 hours for a DSC with heat load greater than 12 kW and ≤ 13 kW or within 30 hours for a DSC with heat load greater than 13 kW and ≤ 14 kW.	A.1 Establish helium pressure of at least 1 atm and no greater than 20 psig in the 24PT1-DSC.	24 hours
	OR	
	A.2 Flood the DSC with water submerging all fuel assemblies.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 12.3.1.1.1 Verify that the 24PT1-DSC vacuum pressure is less than, or equal to, 3 Torr (3 mm Hg) absolute for at least 30 minutes, within the specified total time duration based on heat load.	Once per 24PT1-DSC, after an acceptable NDE of the inner top cover plate weld.

12.3.1.2 24PT1-DSC Helium Backfill Pressure

LCO 12.3.1.3 24PT1-DSC helium backfill pressure shall be 1.5 ± 1.5 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------- NOTE ----------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 12.3.1.2.1 is performed.		
A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT1-DSC helium backfill pressure to within the limit.	24 hours
	OR	
	A.2 Flood the DSC with water submerging all fuel assemblies.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR12.3.1.2.1 Verify that the 24PT1-DSC helium backfill pressure is 1.5 ± 1.5 psig.	Once per 24PT1-DSC, after the completion of TS 12.3.1.1 actions.

12.3.1.3 24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent/Siphon Port Cover Welds

LCO 12.3.1.3 The 24PT1-DSC helium leak rate of the inner top cover plate and vent/siphon port cover welds shall be less than or equal to 10^{-7} std-cc/sec.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------- NOTE ----------
This specification is applicable to the inner top cover plate weld and vent/siphon port cover welds of all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 12.3.1.3.1 is performed.		
A. Helium leak rate not met.	A.1 Establish the 24PT1-DSC leak rate to within the limit.	7 days
B. Required Action A.1 and associated Completion Time not met.	B.1 Determine and complete corrective actions necessary to return the 24PT1-DSC to an analyzed condition.	30 days
	OR	
	B.2 Unload the 24PT1-DSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR12.3.1.3.1 Verify that the helium leak rate is within limit.	Once per 24PT1-DSC, after the welding of the root pass(es) for the outer top cover plate.

12.4.0 Design Features

The specifications in this section include the design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the Advanced NUHOMS® System design. The principal objective of this section is to describe the design envelope that may constrain any physical changes to essential equipment. Included in this section are the site environmental parameters that provide the bases for design, but are not inherently suited for description as LCOs.

12.4.1 Site

12.4.1.1 Site Location

Because this SAR is prepared for a general license, a discussion of a site-specific ISFSI location is not applicable.

12.4.2 Storage *System* Features

12.4.2.1 Storage Capacity

The total storage capacity of the ISFSI is governed by the plant-specific license conditions.

12.4.2.2 Storage Pad

For sites for which soil-structure interaction is considered important, the licensee is to perform site-specific analysis considering the effects of soil-structure interaction. Amplified seismic spectra at the location of the AHSM center of gravity (CG) is to be developed based on the SSI responses. The AHSM center of gravity is shown in Table 3.2-1. The site-specific spectra at the AHSM CG must be bounded by the spectra presented in Chapter 2.

The storage pad location shall have no potential for liquefaction at the site-specific SSE level earthquake.

Additional requirements for the pad configuration are provided in Section 12.4.4.2.

12.4.2.3 Canister Neutron Poison

Neutron poison in the configuration shown in the canister drawing provided in Section 1.5.2 with a minimum ^{10}B loading of 0.025 grams/square centimeter is provided for criticality.

12.4.2.4 Canister Flux Trap Configuration

The canister flux trap configuration is defined by the spacer disc ligament width dimensions. These dimensions are shown in the canister drawing provided in Section 1.5.2.

12.4.3 Codes and Standards

12.4.3.1 Advanced Horizontal Storage Module (AHSM)

The reinforced concrete AHSM is designed to meet the requirements of ACI 349-97. Load combinations specified in ANSI 57.9-1984, Section 6.17.3.1 are used for combining normal operating, off-normal, and accident loads for the AHSM.

12.4.3.2 Dry Shielded Canister (24PT1-DSC)

The 24PT1-DSC is designed fabricated and inspected to the maximum practical extent in accordance with ASME Boiler and Pressure Vessel Code Section III, Division 1, 1992 Edition with Addenda through 1994, including exceptions allowed by Code Case N-595-1, Subsections NB, NF, and NG for Class 1 components and supports. Code exceptions are discussed in 12.4.3.4.

12.4.3.3 Transfer Cask

The Transfer Cask shall meet the codes and standards that are applicable to its design under Certificate of Compliance C of C *72-1004, OS-197 Transfer Cask.*

A solar shield is required for cask transfer operations at temperatures exceeding 100°F.

12.4.3.4 Exceptions to Codes and Standards

ASME Code exceptions for the 24PT1-DSC are listed below:

DSC Shell Assembly ASME Code Exceptions, Subsection NB

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The 24PT1-DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability & certification are maintained in accordance with TNW's NRC approved QA program

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NB-6111	All completed pressure retaining systems shall be pressure tested	The shield plug support ring and vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the siphon block weld is helium leak tested after fuel is loaded and the inner top closure plate installed.
NB-7000	Overpressure Protection	No overpressure protection is provided for the 24PT1-DSC. The function of the 24PT1-DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The 24PT1-DSC is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The 24PT1-DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the DSC.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT1-DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the 24PT1-DSC. QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TNW's approved QA program.

Basket ASME Code Exceptions, Subsection NG/NF

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NG/NF-1100	Requirements for Code Stamping of Components	The 24PT1-DSC baskets are designed & fabricated in accordance with the ASME Code, Section III, Subsection NG/NF to the maximum extent practical as described in the SAR, but Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME N or NPT stamp or be ASME Certified.
NG/NF-2130 NG/NF-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG/NF-2130 is not possible. Material traceability & certification are maintained in accordance with TNW's NRC approved QA program.
Table NG-3352-1	Permissible Joint Efficiency Factors	Joint efficiency (quality) factor of 1 is assumed for the guidesleeve longitudinal weld. Table NG-3352-1 permits a quality factor of 0.5 for full penetration weld with visual inspection. Inspection of both faces provides n= (2*0.5)=1. This is justified by this gauge of material (0.12 inch) with visual examination of both surfaces which ensures that any significant deficiencies would be observed and corrected.

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NG/NF-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT1-DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the 24PT1-DSC. QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TNW's approved QA program.

Proposed alternatives to the ASME code, including the aforementioned ASME Code exceptions may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards, or designee. The applicant should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2. Compliance with the specified requirements of ASME Code, Section III, 1992 Edition with Addenda through 1994 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for exceptions in accordance with this section should be submitted in accordance with 10CFR 72.4.

12.4.3.5 24PT1-DSC Helium Leak Test

Helium gas used in leak testing per LCO 12.3.1.4 is to be performed using helium gas ≥99% purity.

12.4.4 Storage Location Design Features

The following storage location design features and parameters shall be verified by the system user to assure technical agreement with this SAR.

12.4.4.1 Storage Configuration

AHSMs are to be tied together as specified in the AHSM drawing provided in Section 1.5.2 in single rows or back to back arrays with not less than 3 modules tied together (side by side). Each group of modules not tied together must be separated from other groups by a minimum of 20 feet to accommodate possible sliding during a seismic event. The distance between any module and the edge of the ISFSI pad shall be no less than 10 feet.

12.4.4.2 Pad Properties to Limit 24PT1-DSC Gravitational Loadings Due to Postulated Drops

The TC/24PT1-DSC has been evaluated for drops of up to 80 inches onto a reinforced concrete pad. The evaluations are based on the concrete parameters specified in EPRI Report NP-4830, "The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks," October 1986.

12.4.4.3 Site Specific Parameters and Analyses

The following parameters and analyses shall be verified by the system user for applicability at their specific site. Other natural phenomena events, such as lightning, tsunamis, hurricanes, and seiches, are site specific and their effects are generally bounded by other events, but they should be evaluated by the user.

1. Tornado maximum wind speeds: 290 mph rotational
   70 mph translational

2. Flood levels up to 50 ft. and water velocity of 15 fps.

3. One-hundred year roof snow load of 110 psf.

4. Normal ambient temperatures of 0°F to 104°F.

5. Off-normal ambient temperature range of –40°F without solar insolation to 117°F with full solar insolation.

6. The potential for fires and explosions shall be addressed, based on site-specific considerations.

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

8. Seismic restraints shall be provided to prevent overturning of a loaded TC during a seismic event if a certificate holder determines that the horizontal acceleration is 0.40 g or greater. The determination of horizontal acceleration acting at the center of gravity (CG) of the loaded TC must be based on a peak horizontal ground acceleration at the site.

12.5.0 Administrative Controls

12.5.1 Procedures

Each user of the Advanced NUHOMS® System will prepare, review, and approve written procedures for all normal operations, maintenance, and testing at the ISFSI prior to its operation. Written procedures shall be established, implemented, and maintained covering the following activities that are important to safety:

- Organization and management
- Routine ISFSI operations
- Alarms and annunciators
- Emergency operations
- Design control and facility change/modification
- Control of surveillances and tests
- Control of special processes
- Maintenance
- Health physics, including ALARA practices
- Special nuclear material accountability
- Quality assurance, inspection, and audits
- Physical security and safeguards
- Records management
- Reporting
- All programs specified in Section 12.5.2

12.5.2 Programs

Each user of the Advanced NUHOMS® System will implement the following programs to ensure the safe operation and maintenance of the ISFSI:

- Safety Review Program
- Training Program
- Radiological Environmental Monitoring Program
- Radiation Protection Program
- AHSM Thermal Monitoring Program

12.5.2.1 Safety Review Program

Users shall conduct safety reviews in accordance with 10CFR 72.48 to determine whether proposed changes, tests, and experiments require NRC approval before implementation. Changes to the Technical Specification Bases and other licensing basis documents will be conducted in accordance with approved administrative procedures. Changes may be made to Technical Specification Bases and other licensing basis documents without prior NRC approval, provided the changes meet the criteria of 10CFR 72.48.

The safety review process will contain provisions to ensure that the Technical Specification Bases and other licensing basis documents are maintained consistent with the SAR.

Proposed changes that do not meet the criteria above will be reviewed and approved by the NRC before implementation. Changes to the Technical Specification Bases implemented without prior NRC approval will be provided to the NRC in accordance with 10CFR 72.48.

12.5.2.2 Training Program

Training modules shall be developed as required by 10CFR 72. Training modules shall require a comprehensive program for the operation and maintenance of the Advanced NUHOMS® System and the independent spent fuel storage installation (ISFSI). The training modules shall include the following elements, at a minimum:

- Advanced NUHOMS® System design (overview)
- ISFSI Facility design (overview)
- Systems, Structures, and Components Important to Safety (overview)
- Advanced NUHOMS® System Safety Analysis Report (overview)
- NRC Safety Evaluation Report (overview)
- Certificate of Compliance conditions
- Advanced NUHOMS® System Technical Specifications
- Applicable Regulatory Requirements (e.g.,10CFR 72, Subpart K, 10CFR 20, 10 CFR Part 73)
- Required Instrumentation and Use
- Operating Experience Reviews
- Advanced NUHOMS® System and Maintenance procedures, including:
  - Fuel qualification and loading,
  - Rigging and handling,
  - Loading Operations as described in Chapter 8 *and Section 9.2* of the SAR,
  - Unloading Operations including reflooding,
  - Auxiliary equipment operations and maintenance (i.e., welding operations, vacuum drying, helium backfilling and leak testing, reflooding),
  - Transfer operations including loading and unloading of the Transfer Vehicle,
  - ISFSI Surveillance operations,
  - Radiation Protection,
  - Maintenance,
  - Security,
  - Off-normal and accident conditions, responses and corrective actions.

12.5.2.3 Radiological Environmental Monitoring Program

a) A radiological environmental monitoring program will be implemented to ensure that the annual dose equivalent to an individual located outside the ISFSI controlled area does not exceed the annual dose limits specified in 10CFR 72.104(a).

b) Operation of the ISFSI will not create any radioactive materials or result in any credible liquid or gaseous effluent release.

c) In accordance with 10CFR 72.212(b)(2), a periodic report will be submitted by the licensee that specifies the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents during the previous calendar year of operation.

12.5.2.4 Radiation Protection Program

The Radiation Protection Program will establish administrative controls to limit personnel exposure to As Low As Reasonably Achievable (ALARA) levels in accordance with 10CFR Part 20 and Part 72.

a. As part of its evaluation pursuant to 10CFR 72.212, the licensee shall perform an analysis to confirm that the limits of 10CFR 20 and 10CFR 72.104 will be satisfied under the actual site conditions and configurations considering the planned number of 24PT1-DSCs to be used and the planned fuel loading conditions.

b. A monitoring program to ensure the annual dose equivalent to any real individual located outside the ISFSI controlled area does not exceed regulatory limits is incorporated as part of the environmental monitoring program in the Radiological Environmental Monitoring Program of Section 12.5.2.3.

c. Following placement of each loaded Transfer Cask into the cask decontamination area and prior to transfer to the ISFSI, the 24PT1-DSC smearable surface contamination levels on the outer surface of the 24PT1-DSC shall be less than 2,200 dpm/100 cm^2 from beta and gamma emitting sources, and less than 220 dpm/100 cm^2 from alpha emitting sources.

The contamination limits specified above are based on the allowed removable external radioactive contamination specified in 49 CFR 173.443 (as referenced in 10 CFR 71.87(i)) the system provides significant additional protection for the 24PT1-DSC surface than the transportation configuration. The AHSM will protect the 24PT1-DSC from direct exposure to the elements and will therefore limit potential releases of removable contamination. The probability of any removable contamination being entrapped in the AHSM air flow path released outside the AHSM is considered extremely small.

12.5.2.5 AHSM Thermal Monitoring Program

This program provides guidance for temperature measurements that are used to monitor the thermal performance of each AHSM. The intent of the program is to prevent conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

a) AHSM Concrete Temperature

The temperature measurement will be a direct measurement of the AHSM concrete temperature, or other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. A temperature measurement of the thermal performance for each AHSM will be taken on a daily basis.

If the temperature of the AHSM at the monitored location rises by more than 80°F, based on a daily surveillance, then it is possible that some type of an inlet and or outlet vent blockage has occurred and appropriate corrective actions will be taken to avoid exceeding the concrete and cladding temperature limits. *The 80°F value is obtained from a review of a transient thermal analysis of the AHSM with a 24 kW heat load to ensure that the rapid heatup is detected in time to initiate corrective action prior to exceeding concrete temperature limits.*
(Calculation SCE-01.0402, Figure 10-1)

In addition, if the temperature of the AHSM at the monitored location is greater than 225°F, then it is possible that some type of an inlet and or outlet vent blockage has occurred and appropriate corrective actions need to be taken to avoid exceeding the concrete and cladding temperature limits. *The 225°F temperature limit is chosen based on the expected concrete temperature for the 14 kW blocked vent scenario to ensure that the associated fuel clad temperature is not exceeded.*

The AHSM Thermal Monitoring Program provides a positive means to identify conditions that could approach the temperature criteria for proper AHSM operation and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

b) AHSM Air Temperature Difference

Following initial 24PT1-DSC transfer to the AHSM, the air temperature difference between ambient temperature and the roof vent temperature will be measured 24 hours after DSC insertion into the HSM and again 7 days after insertion into the AHSM. If the air temperature differential is greater than 100°F, the air inlets and exits should be checked for blockage. If after removing any blockage found, the temperature is still greater than that ≤100°F, corrective actions and analysis of existing conditions will be performed in accordance with the site corrective action program to confirm that conditions adversely affecting the concrete or fuel cladding do not exist.

The specified air temperature rise ensures the fuel clad and concrete temperatures are maintained at or below acceptable long-term storage limits. If the temperature rise is within the ≤100°F, then the AHSM and 24PT1-DSC are performing as designed and no further temperature measurements are required.

c) AHSM Air Vents

Since the AHSMs are located outdoors, there is a possibility that the AHSM air inlet and outlet openings could become blocked by debris. Although the ISFSI security fence and AHSM bird screens reduce the probability of AHSM air vent blockage, the ISFSI SAR postulates and analyzes the effects of air vent blockage.

The AHSM design and accident analyses demonstrate the ability of the ISFSI to function safely if obstructions in the air inlets or outlets impair airflow through the AHSM for extended periods. This specification ensures that blockage will not exist for periods longer than assumed in the analyses.

Site personnel will conduct a daily visual inspection of the air vents to ensure that AHSM air vents are not blocked for more than 40 hours and that blockage will not exist for periods longer than assumed in the safety analysis.

12.5.3 Lifting Controls

12.5.3.1 Cask Lifting Heights

The lifting height of a loaded cask/24PT1-DSC, is limited as a function of location and temperature, as follows:

a) The maximum lift height of the cask/24PT1-DSC inside the Fuel Handling Building shall be 80 inches if the ambient temperature is below 0°F but higher than -80°F.

b) No lift height restriction (no lift height restriction as a function of temperature, other 10CFR 50 administrative lift height restrictions may apply) is imposed on the cask/24PT1-DSC during LOADING OPERATIONS, if the ambient temperature is higher than 0°F.

c) The maximum lift height and handling height for all TRANSFER OPERATIONS shall be 80 inches if the ambient temperature is greater than 0°F.

These restrictions ensure that any 24PT1-DSC drop as a function of location or low temperature is within the bounds of the accident analysis. If the ambient temperature is outside of the specification limits, LOADING and TRANSFER OPERATIONS will be terminated.

12.5.3.2 Cask Drop

Inspection Requirement

The 24PT1-DSC will be inspected for damage after any transfer cask drop of fifteen inches or greater.

Background

TC/24PT1-DSC handling and loading activities are controlled under the 10CFR 50 license until a loaded TC/24PT1-DSC is placed on the transporter, at which time fuel handling activities are controlled under the 10CFR 72 license. Although the probability of dropping a loaded TC/24PT1-DSC while en route from the Fuel Handling Building to the ISFSI is small, the potential exists to drop the cask 15 inches or more.

Safety Analysis

The analysis of bounding drop scenarios shows that the transfer cask will maintain the structural integrity of the 24PT1-DSC confinement boundary from an analyzed drop height of 80 inches. The 80-inch drop height envelopes the maximum vertical height of the transfer cask when secured to the transfer trailer while en route to the ISFSI.

Although analyses performed for cask drop accidents at various orientations indicate much greater resistance to damage, requiring the inspection of the DSC after a drop of 15 inches or greater ensures that:

1. The DSC will continue to provide confinement

2. The transfer cask can continue to perform its design function regarding DSC transfer and shielding.

ADVANCED NUHOMS® SYSTEM TECHNICAL SPECIFICATION BASES

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B 12.2.0 FUNCTIONAL AND OPERATING LIMITS

BASES

BACKGROUND

The 24PT1-DSC design requires certain limits on spent fuel parameters, including fuel type, maximum allowable enrichment prior to irradiation, maximum burnup, minimum acceptable cooling time prior to storage in the 24PT1-DSC, and physical condition of the spent fuel (i.e., intact or damaged fuel assemblies). Other important limitations are the radiological source terms from associated Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs), and Thimble Plug Assemblies (TPAs). These limitations are included in the thermal, structural, radiological, and criticality evaluations performed for the canister.

APPLICABLE SAFETY ANALYSIS

Various analyses have been performed that use these fuel parameters as assumptions. These assumptions are included in the thermal, criticality, structural, shielding and confinement analyses.

Technical Specification Tables 12.2-1, 12.2-2 and 12.2-3 provide the key fuel parameters that require confirmation prior to 24PT1-DSC loading.

FUNCTIONAL AND OPERATING LIMITS VIOLATIONS

If Functional and Operating Limits are violated, the limitations on the fuel assemblies in the canister have not been met. Actions must be taken to place the affected fuel assemblies in a safe condition. This safe condition may be established by returning the affected fuel assemblies to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to remain in the canister if that is determined to be a safe condition.

Notification of the violation of a Functional and Operating Limit to the NRC is required within 24 hours. Written reporting of the violation must be accomplished within 30 days. This notification and written report are independent of any reports and notification that may be required by 10CFR 72.75.

REFERENCES

SAR Chapter 2

B 12.3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 12.3.0.1, 12.3.0.2, 12.3.0.4 and 12.3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 12.3.0.1

LCO 12.3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the canister is in the specified conditions of the Applicability statement of each Specification).

LCO 12.3.0.2

LCO 12.3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, the canister may have to be placed in the spent fuel pool and unloaded. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for

intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires if the equipment remains removed from service or bypassed.

When a change in specified condition is required to comply with Required Actions, the equipment may enter a specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 12.3.0.3 This specification is not applicable to the Advanced NUHOMS® System. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 12.3.0.4 LCO 12.3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the Advanced NUHOMS® System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the equipment being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the equipment for an unlimited period of time in specified condition provides an acceptable level of safety for continued operation. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 12.3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In

addition, the provisions of LCO 12.3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of a canister.

Exceptions to LCO 12.3.0.4 are stated in the individual Specifications.

Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Surveillances do not have to be performed on the associated equipment out of service (or on variables outside the specified limits), as permitted by SR 12.3.0.1. Therefore, changing specified conditions while in an ACTIONS Condition, either in compliance with LCO 12.3.0.4 or where an exception to LCO 12.3.0.4 is stated, is not a violation of SR 12.3.0.1 or SR 12.3.0.4 for those Surveillances that do not have to be performed due to the associated out of service equipment.

LCO 12.3.0.5

LCO 12.3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or not in service in compliance with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 12.3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The equipment being returned to service meets the LCO; or

b. Other equipment meets the applicable LCOs.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed required testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

LCO 12.3.0.6

This specification is not applicable to an Advanced NUHOMS® System. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 12.3.0.7

This specification is not applicable to an Advanced NUHOMS® System. The placeholder is retained for consistency with the power reactor technical specifications.

B 12.3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 12.3.0.1 through SR 12.3.0.4 establish the general requirements applicable to all Specifications in Sections 12.3.1 and 12.3.2 and apply at all times, unless otherwise stated.

SR 12.3.0.1

SR 12.3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 12.3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the equipment is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 12.3.0.2, prior to returning equipment to service.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment within its LCO. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 12.3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary equipment parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not

otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

SR 12.3.0.2

SR 12.3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 12.3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 12.3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR includes a Note in the Frequency stating, "SR 12.3.0.2 is not applicable".

As stated in SR 12.3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the equipment in an alternative manner.

The provisions of SR 12.3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 12.3.0.3

SR 12.3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency,

whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 12.3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance. The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 12.3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 12.3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 12.3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered not in service or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is not in service, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 12.3.0.1.

SR 12.3.0.4

SR 12.3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to an appropriate status before entering an associated specified condition in the Applicability. However, in certain circumstances, failing to meet an SR will not result in SR 12.3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 12.3.0.1, which states that Surveillances do not have to be performed on such equipment. When equipment does not meet the LCO, SR 12.3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 12.3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 12.3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 12.3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 12.3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of a AHSM or 24PT1-DSC.

The precise requirements for performance of SRs are specified such that exceptions to SR 12.3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternatively, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SR annotation is found in Section 12.1.4, operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

B 12.3.1 24PT1-DSC INTEGRITY

B 12.3.1.1 24PT1-DSC Vacuum Drying Time (Duration) and Pressure

BASES

BACKGROUND

A 24PT1-DSC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield plug is then placed on the 24PT1-DSC. Subsequent operations involve moving the 24PT1-DSC to the decontamination area and removing water from the 24PT1-DSC. After the 24PT1-DSC inner top cover plate is secured, vacuum drying of the 24PT1-DSC is performed, and the 24PT1-DSC is backfilled with helium. During normal storage conditions, the fuel assemblies are stored in the 24PT1-DSC with an inert helium atmosphere, which is a better conductor than air or vacuum, which results in lower fuel clad temperatures and provides an inert atmosphere during storage conditions.

24PT1-DSC vacuum drying is utilized to remove residual moisture from the cavity after the 24PT1-DSC has been drained of water. Any water which was not drained from the 24PT1-DSC evaporates from fuel or basket surfaces due to the vacuum. This vacuum drying operation is aided by the temperature increase due to the heat generation of the fuel.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in a 24PT1-DSC is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the 24PT1-DSC in which the fuel assemblies are stored. Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. This protective environment is accomplished by removing water from the 24PT1-DSC and backfilling the 24PT1-DSC with an inert gas. The removal of water is necessary to prevent phase change–related pressure increase upon heatup. Time limits on vacuum drying >12 kW heat load are required for keeping the 24PT1-DSC materials under the ASME maximum temperature limits. This SAR evaluates and documents that the 24PT1-DSC confinement boundary is not compromised due to any normal, abnormal or accident condition postulated (SAR Chapter 3 and 11 structural analyses).

LCO

A stable vacuum pressure of < 3 torr further ensures that all liquid water has evaporated in the 24PT1-DSC cavity, and that the resulting inventory of oxidizing gases in the 24PT1-DSC is well below 0.25 volume % (two vacuum drying cycles from atmospheric pressure, 760 torr, to 3 torr

will result in approximately $[3/760]^2 \times 100 < 0.002$ volume % of atmospheric, potentially oxidizing, gas).

APPLICABILITY

This is applicable to all 24PT1-DSCs.

ACTIONS

The actions specified require establishment of a helium pressure of at least 1 atmosphere within the time limits specified in the LCO for heat loads between 12 and 14 kW. The timeframe specified applies to the two vacuum drying operations and the helium backfill operations. If the required vacuum can not be established within the timeframe specified in the Condition column of the Actions table, a helium atmosphere (with a pressure of at least one atmosphere) is to be established within 24 hours or perform an assessment and implementation of corrective actions to return the 24PT1-DSC to an analyzed condition or reflood the DSC submerging all fuel assemblies. The 20 psig limit in the action section is based on the maximum allowed blowdown pressure.

SURVEILLANCE REQUIREMENTS

Ensure a minimum oxidizing gas content.

REFERENCES

SAR Chapter 3 and 4

B 12.3.1 24PT1-DSC INTEGRITY

B 12.3.1.2 24PT1-DSC Helium Backfill Pressure

BASES

BACKGROUND

A 24PT1-DSC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield plug is then placed on the 24PT1-DSC. Subsequent operations involve moving the 24PT1-DSC to the decontamination area and removing water from the 24PT1-DSC. After the 24PT1-DSC inner top cover plate is welded, vacuum drying of the 24PT1-DSC is performed, and the 24PT1-DSC is backfilled with helium. During normal storage conditions, the 24PT1-DSC is backfilled with helium, which is a better conductor than air or vacuum, which results in lower fuel clad temperatures. The inert helium environment protects the fuel from potential oxidizing environments.

APPLICABLE SAFETY ANALYSIS

Long-term integrity of the fuel cladding depends on storage in an inert atmosphere. SAR section 3.5 evaluates the effect of long term storage and short term temperature transients on fuel cladding integrity. Credit for the helium backfill pressure is taken to limit the potential for corrosion of the fuel cladding. SAR Chapter 4 evaluates the 24PT1-DSC maximum pressure under normal, off-normal, and accident conditions

LCO

24PT1-DSC backpressure is maintained within a range of pressure that will ensure maintenance of the helium backfill pressure over time and will not result in excessive 24PT1-DSC pressure in normal, off-normal and accident conditions.

APPLICABILITY

This specification is applicable to all 24PT1-DSCs.

ACTIONS

The actions required and associated completion times are associated with the time limits established in specification 12.3.1.2. The total time for vacuum drying and helium backfill is specified in specification 12.3.1.2 as a function of 24PT1-DSC heat load. These time limits are imposed to ensure that the 24PT1-DSC basket components will not exceed material temperatures for which ASME has specified maximum allowable stresses.

SURVEILLANCE REQUIREMENTS

To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat.

REFERENCES

SAR Chapters 3 and 4

B 12.3.1 24PT1-DSC INTEGRITY

B 12.3.1.3 24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent /Siphon Port Cover Welds

BASES

BACKGROUND

A 24PT1-DSC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield plug is then placed on the 24PT1-DSC. Subsequent operations involve moving the 24PT1-DSC to the decontamination area and removing water from the 24PT1-DSC. After the 24PT1-DSC inner top cover plate is secured, vacuum drying of the 24PT1-DSC is performed, and the 24PT1-DSC is backfilled with helium. Prior to completion of the 24PT1-DSC outer top cover plate welding, the helium leak rate is determined to ensure that the confinement boundary is leaktight, as required in the confinement analysis of SAR Chapter 7. The 24PT1-DSC shell and inner bottom cover plate confinement boundaries are confirmed to be leaktight during 24PT1-DSC fabrication.

APPLICABLE SAFETY ANALYSIS

The 24PT1-DSC confinement boundary is helium leak tested to confirm that it is leaktight in accordance with SAR Chapter 7 to preclude 24PT1-DSC leakage resulting in exposure to the public.

LCO

The confinement boundary is maintained leaktight to eliminate releases of radioactivity from the 24PT1-DSC to the environment during storage.

APPLICABILITY

This specification is applicable to the inner top cover plate and vent/siphon port cover welds of all 24PT1-DSCs.

ACTIONS

Should the helium leak rate not meet the requirements of this specification, the 24PT1-DSC must be returned to an analyzed condition or unloaded.

SURVEILLANCE REQUIREMENTS

To ensure that the 24PT1-DSC confinement boundary is leaktight and to retain helium cover gases within the 24PT1-DSC and prevent oxygen from entering the 24PT1-DSC.

REFERENCES

SAR Chapter 7

13. QUALITY ASSURANCE

TN West'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 SAR.

13.1 Introduction

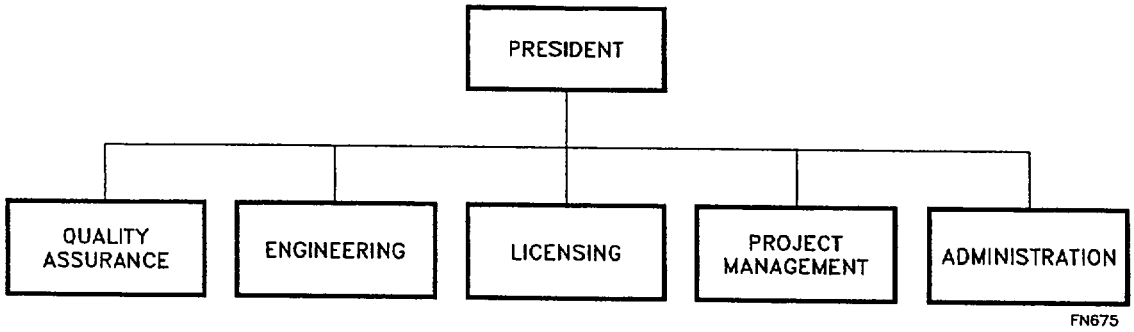
The complete description and specific commitments of the TN West QA Program are contained in the TN West QA Manual [13.2]. This manual has been approved by the Nuclear Regulatory Commission (NRC) for performing 10CFR 72 related activities. Changes to the TN West QA program shall be submitted to the NRC for approval within thirty (30) days of implementation. Changes to the TN West 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 West QA Manual and TN West 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 West 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 West 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 West 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 West's program, then the utility QA program is considered an acceptable substitute for their scope of responsibility.

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

13.3.1 Project Organization

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

QA duties are performed by the TN West 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 West's organizational structure, functional responsibilities, levels of authority, and lines of internal and external (client and supplier) communication may be found in the TN West 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 West QA Manual.

13.3.2 QA Program

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

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

13.3.3 Design Control

"Important to safety" and "safety related" Advanced NUHOMS® System design activities shall be implemented in accordance with the TN West 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 West 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 West or TN West's client. The time of submittal shall also be established. When TN West 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 West 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 West QA Manual. Controlled documents include, but are not limited to, the TN West 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 West 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 West 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 West 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 West 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 West 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 West QA Manual.

13.3.10 Inspection

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

13.3.11 Test Control

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

13.3.12 Control of Measuring and Test Equipment

The TN West 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 West 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 West QA Manual.

13.3.15 Control of Nonconforming Items

The TN West 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 West 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 West 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 West 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 West QA Manual.

Unannounced QA surveillances may be performed on activities affecting quality by the TN West 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 West 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 West Quality Assurance Manual," current revision.

[13.3] "Transnuclear West 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 Part 72 storage only 24PT1-DSC be upgraded to current Part 71 regulations. The 24PT1-DSC design is very similar to the FO-DSC licensed under Part 71 by TN West, C of C 71-9255 [14.2]. An amendment to C of C 71-9255 will be initiated 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}	5.3×10^{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] TN West, Safety Analysis Report for the NUHOMS® MP187 Multi-Purpose Cask, NUH-005, Revision 9, September 1998, USNRC 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.

Cross Reference Matrix

Section 14.1 reference to previous activation analyses for TN-32 and NUHOMS® site license are based on TN-32 SAR (for the 24PT1-DSC/transfer cask) and BGE calculation BGE001-0612.