N.10 Radiation Protection

Section 7.4.1 discusses the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, several of which involve potential exposure to personnel. This section of the Appendix provides occupational exposure and off-site dose rates from NUHOMS®-24PHB DSC to be stored in NUHOMS® HSM Model 102.

N.10.1 Occupational Exposure

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Table 7.4-1. The total exposure for the occupational dose due to placing a single NUHOMS®-24PHB DSC into storage is conservatively estimated to be 2.7 person-rem (24PHBS DSC) and 3.1 person-rem (24PHBL DSC). This is a very conservative estimate because the dose rates on and around the 24PHB DSC's used in these calculations are based on very conservative assumptions for the design-basis source terms and analyses models. The calculated exposures for both configurations are due mainly to the expected gamma dose rate during preparation for welding. The increased calculated exposure for the 24PHBL DSC is due to the additional design basis BPRA gamma source.

The NUHOMS®-24PHB System loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table N.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. This evaluation is performed for the storage of one design-basis NUHOMS®-24PHBS DSC and 24PHBL DSC in an HSM. The loading operations are identical for the 24PHBS and 24PHBL DSC. The dose rates applicable for each operation are based on the results presented in Section N.5.4 for loading operations. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Each Licensee must evaluate any such changes in accordance with its ALARA program.

The amount of time required to complete some operations as identified in Table N.10-1 may be greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The total exposure calculated for each task is therefore not necessarily equal to the number of workers multiplied by the time required, multiplied by a dose rate. The exposure estimation for each task accounts for cases such as vacuum drying correctly, and assumes that good ALARA practices are followed.

The results of the evaluations of the 24PHBS and 24PHBL are presented in Table N.10-1.

N.10.2 Off-Site Dose Calculations

Calculated dose rates in the immediate vicinity of the NUHOMS®-24PHB System are presented in Section N.5 which provides a detailed description of source term configuration, analysis models and bounding dose rates. Off-site dose rates and doses are presented in this section. This evaluation determines the neutron and gamma-ray off-site dose rates including skyshine in the vicinity of the two generic ISFSI layouts containing design-basis fuel in the NUHOMS®-24PHB DSCs. The first generic ISFSI evaluated is a 2x10 back-to-back array of HSMs loaded with design-basis fuel, including BPRAs, in NUHOMS®-24PHBL DSCs. The second generic layout evaluated is two 1x10 front-to-front arrays of HSMs loaded with design-basis fuel, including BPRAs, in NUHOMS®-24PHB DSCs. This evaluation provides results for distances ranging from 6.1 to 600 meters from each face of the two arrays of HSMs.

The total annual exposure for each ISFSI layout as a function of distance from each face is given in Table N.10-2 and plotted in Figure N.10-1. The total annual exposure assumes 100% occupancy for 365 days.

The Monte Carlo computer code MCNP [10.1] calculated the dose rates at the specified locations around the arrays of HSMs. The results of this calculation provide an example of how to demonstrate compliance with the relevant radiological requirements of 10CFR20 [10.2], 10CFR72 [10.3], and 40CFR190 [10.4] for a specific site. Each site must perform specific site calculations to account for the actual layout of the HSMs and fuel source.

The assumptions used to generate the geometry of the two ISFSIs for the MCNP analysis are summarized below.

- The 20 HSMs in the 2x10 back-to-back array are modeled as a box enveloping the 2x10 array of HSMs including the six inch gaps between modules and the 2-foot shield walls on the two sides of the array. MCNP starts the source particles on the surfaces of the box.

- The 20 HSMs in the two 1x10 front-to-front arrays are modeled as two boxes which envelope each 1x10 array of HSMs including the six inch gaps between modules and the 2-foot shield walls on the two sides and back of each array. MCNP starts the source particles on the surfaces of one of the boxes.

- The ISFSI approach slab is modeled as concrete. Because the ground composition has, at best, only a secondary impact on the dose rates at the detectors, any differences between this assumed layout and the actual layout would not have a significant affect on the site dose rates.

- For the 2x10 array, the interiors of the HSMs and shield walls are modeled as air. Most particles that enter the interiors of the HSMs and shield walls will therefore pass through unhindered.

- For the two 1x10 arrays, the interiors of one array of HSMs and shield walls are modeled as air. Most particles that enter the interiors of these HSMs and shield walls will therefore pass through unhindered. The other 1x10 array is modeled as concrete to simulate the shielding

provided by the second array of HSMs for the direct radiation from the front of the opposing 1x10 array.

- The "universe" is a sphere surrounding the ISFSI. To account for skyshine radius of this sphere (r=500,000 cm) is more than 10 mean free paths for neutrons and 50 mean free paths for gammas greater than that of the outermost surface, thus ensuring that the model is of a sufficient size to include all interactions, including skyshine, affecting the dose rate at the detectors.

The assumption used to generate the HSM surface sources for the MCNP analysis is summarized below.

- The HSM surface sources are bootstrapped (input to provide an equivalent boundary condition) using the HSM surface average dose rates calculated in Section N.5.4.

The assumptions used for the MCNP analysis are summarized below.

- MCNP starts the source particles on the ISFSI array surface with initial directions following a cosine distribution. Radiation fluxes outside thick shields such as the HSM walls and roof tend to have forward peaked angular distributions; therefore, a cosine function is a reasonable approximation for the starting direction distribution. Vents through shielding regions such as the HSM vents tend to collimate particles such that a semi-isotropic assumption would not be appropriate.

- Point detectors determine the dose rates on the four sides of the ISFSI as a function of distance from the ISFSI. All detectors represent the dose rate at three feet above ground level.

- Source information required by MCNP includes gamma-ray and neutron spectra for the HSM array surfaces, total gamma-ray and neutron activities for each HSM array face and total gamma-ray and neutron activities for the entire ISFSI. The neutron and gamma-ray spectra are determined using a 1-D ANISN [10.6] run through the HSM roof using the design-basis in-core neutron and gamma fuel sources. Use of the roof is conservative because it represents the thickest cross section of the HSM shield. The thicker shield increases the dose rate importance of the higher energy neutrons and gamma-rays from the fuel because the thicker shield filters out the lower energy particles. Therefore, use of the thickest part of the shield results in a harder spectrum for all of the other surfaces. The HSM spectra as determined from ANISN are normalized to a one mrem/hour source using the flux-to-dose-factors from Reference [10.5]. 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, back and front of the HSM. 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 HSM are shown in Table N.10-3. The group fluxes on the HSM 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 N.10-3 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. Similar calculations for neutrons are shown in Table N.10-4.

N.10.2.1 Activity Calculations

2x10 Back-to-Back Array

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the 2x10 back-to-back array of HSMs. The dimensions of the box also include the width of the HSM end shield walls. As 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.

Two 1x10 Front-to-Front Arrays

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the two 1x10 arrays of HSMs. The dimensions of the box also include the width of the HSM end and back shield walls. As 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 HSM surface activities are summarized in Table N.10-5.

N.10.2.2 Dose Rates

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs. The HSM is modeled in MCNP as a box, representing the HSM arrays.

Neutron and gamma-ray sources are placed on each HSM, with shield walls, surface using the spectra and activities determined above. The angular distribution of source particles is modeled as a cosine distribution. The contribution of capture gamma-rays has been neglected, as has the contribution of bremsstrahlung electrons. The inclusion of coherent scattering greatly increases the variance in a problem with point detector tallies without improving the accuracy of the calculation. Thus, coherent scattering of photons is ignored.

The MCNP models of the two ISFSI layouts are described herein. For the 2x10 back-to-back array of HSMs with end shield walls, the "box"dimensions are as follows. The total width is 1158 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm.

For the two 1x10 front-to-front arrays of HSMs with end and back shield walls, the "box" dimensions for each array are as follows. The total width is 640 cm. The length of the "box" is 3220 cm and the height of the "box" is 457 cm. The two 1x10 arrays are 1066 cm (35 feet) apart.

Point detectors are placed at the following locations as measured from each face of the "box": 6.095 m (20 feet), 10 m, 20 m, 30 m, 40 m, 50 m, 60 m, 70 m, 80 m, 90 m, 100 m, 200 m, 300 m, 400 m, 500 m, and 600 m. Each point detector is placed 91.4 cm (3 feet) above the ground.

The MCNP results for each detector from the front of 2x10 back-to-back array are summarized in Table N.10-6. The MCNP results as a function of distance from the back of the two 1x10 front-to-front arrays are summarized in Table N.10-7. The MCNP results as a function of distance from the side of the 2x10 back-to-back array and the two 1x10 front-to-front arrays are summarized in Table N.10-8. The results from Table N.10-6, Table N.10-7 and Table N.10-8 are plotted in Figure N.10-1.

The preceding analyses and the results provided in Figure N.10-1 are intended to provide typical dose rates for the generic ISFSI layouts described in Section N.10.2. They may not be applicable to an actual ISFSI. The written evaluations performed by a licensee for an actual ISFSI must consider the type and number of storage units, layout, characteristics of the irradiated fuel to be stored, site characteristics (e.g., berms, distance to the controlled area boundary, etc.), and reactor operations at the site in order to demonstrate compliance with 10CFR72.104.

N.10.3 References

10.1 "MCNP 4 - Monte Carlo Neutron and Photon Transport Code System," CCC-200A/B, Oak Ridge National Laboratory, RSIC Computer Code Collection, October 1991.

10.2 Title 10, "Energy," Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation."

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

10.4 Title 40, "Protection of Environment," Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

10.5 "American National Standard Neutron and Gamma-Ray Fluence-to-Dose Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, La Grange Park, Illinois, March 1977.

10.6 "ANISN-ORNL - One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering," CCC-254, Oak Ridge National Laboratory, RSIC Computer Code Collection, April 1991.

Table N.10-1
Occupational Exposure Summary, 24PHB System

Task	Number of Workers	Completion Time (hours)
Location: Auxiliary Building and Fuel Pool		
Ready the DSC and TC for Service(1)	2	4
Place the DSC into the TC(1)	3	1
Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2
Fill the DSC Cavity with borated Water(2)	1	6
Install Shield Plug and Connect VDS	2	0.5
Place the Cask Containing the DSC in the Fuel Pool	5	0.5
Verify and Load the Candidate Fuel Assemblies into the DSC	3	5
Place the Top Shield Plug on the DSC	2	1
Raise the Cask/DSC to the Fuel Pool Surface	5	0.5
Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	2	1.03
Location: Cask Decon Area		
Decontaminate the Outer Surface of the Cask (on the hook)(3)	2	2.75
Cask Decontamination (in the decon area)(3)	3	1.5
Drain Water Above DSC Shield Plug	1	0.08
Remove the Cask/DSC Annulus Seal and Set-up Welder(3)	3	3
Weld the Inner Top Cover to the DSC Shell and Perform NDE(3)	3	6.5
Drain DSC Cavity (3)	3	0.77
Vacuum Dry and Backfill the DSC with Helium(3)	3	0.77
Helium Leak Test the Inner Top Cover Plate Weld	2	1
Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE	1	0.5
Install DSC Outer Top Cover Plate(3)	2	1
Leak Tight Test	2	1
Weld the Outer Top Cover Plate to DSC Shell and Perform NDE(3)	5	17.25
Install the Cask Lid	2	1
Location: Reactor /Fuel Building Bay		
Ready the Cask Support Skid and Transport Trailer for Service(1)	2	2
Place the Cask Onto the Skid and Secure(2)	3	0.75
Location: ISFSI Site		
Ready the HSM and Hydraulic Ram System for Service(1)	2	2
Transport the Cask to the ISFSI(1)	6	1
Position the Cask in Close Proximity with the HSM(1)	3	1
Remove the Cask Lid	2	1
Align and Dock the Cask with the HSM	2	0.25
Position and Align Ram with Cask(3)	2	0.5
Remove the RAM Access Cover Plate	1	0.25
Transfer the DSC from the Cask to the HSM(1)	3	0.5
Un-Dock the Cask from the HSM	2	0.08
Install the HSM Access Door	2	0.5
Total estimated dose is 2.7 person-rem per 24PHBS canister load		
Total estimated dose is 3.1 person-rem per 24PHBL canister load		
Total estimated completion time is approximately 68 hrs.		

(1) Performed away from any significant radiation sources.
(2) Personnel are not present throughout this activity.
(3) Dose rates and locations vary during this task.

Table N.10-2
Total Annual Exposure, 24PHB System

2x10 Back-To-Back Array		
Distance (meters)	Front Total Dose (mrem)	Side Total Dose (mrem)
6.096	93082	100758
10	56142	49618
20	21982	15730
30	11596	7869
40	7051	4844
50	4698	3305
60	3418	2399
70	2538	1791
80	1984	1455
90	1550	1125
100	1240	904
200	226	181
300	63	52
400	23	16
500	7	6
600	3	3

Two 1x10 Front-To-Front Array		
Distance (meters)	Back Total Dose (mrem)	Side Total Dose (mrem)
6.096	13038	65274
10	9692	43287
20	5803	17571
30	3863	9145
40	2872	5624
50	2171	3797
60	1713	2731
70	1382	2148
80	1132	1583
90	921	1280
100	795	1016
200	171	197
300	53	62
400	17	18
500	6	7
600	2	3

Table N.10-3
HSM Gamma-Ray Spectrum Calculation Results

Group Number	E_{upper} (MeV)	E_{mean} (MeV)	Flux-Dose Conversion Factors (mR/hr)/(γ/cm^2-sec)	Roof Flux (γ/cm^2-sec)	Dose Rate (mR/hr)	Input Flux (γ/cm^2-sec per mrem/hr)
23	10	9	8.77E-03	2.16E+00	1.90E-02	2.14E-02
24	8	7.25	7.48E-03	1.83E+01	1.37E-01	1.81E-01
25	6.5	5.75	6.37E-03	2.31E+01	1.48E-01	2.29E-01
26	5	4.5	5.41E-03	3.05E+01	1.65E-01	3.02E-01
27	4	3.5	4.62E-03	5.31E+01	2.45E-01	5.26E-01
28	3	2.75	3.96E-03	6.92E+01	2.74E-01	6.86E-01
29	2.5	2.25	3.47E-03	6.21E+02	2.15E+00	6.15E+00
30	2	1.83	3.02E-03	5.28E+02	1.60E+00	5.24E+00
31	1.66	1.495	2.63E-03	2.79E+03	7.33E+00	2.76E+01
32	1.33	1.165	2.21E-03	6.19E+03	1.37E+01	6.14E+01
33	1	0.9	1.83E-03	5.82E+03	1.07E+01	5.77E+01
34	0.8	0.7	1.52E-03	9.26E+03	1.41E+01	9.17E+01
35	0.6	0.5	1.17E-03	1.49E+04	1.75E+01	1.48E+02
36	0.4	0.35	8.76E-04	1.03E+04	9.06E+00	1.02E+02
37	0.3	0.25	6.31E-04	1.42E+04	8.94E+00	1.40E+02
38	0.2	0.15	3.83E-04	3.26E+04	1.25E+01	3.23E+02
39	0.1	0.08	2.67E-04	9.19E+03	2.45E+00	9.10E+01
40	0.05	0.03	9.35E-04	2.58E+01	2.41E-02	2.55E-01
			Totals	**1.07E+05**	**1.01E+02**	**1.06E+03**

Table N.10-4
HSM Neutron Spectrum Calculation Results

Group Number	E_{upper} (MeV)	E_{mean} (MeV)	Flux-Dose Conversion Factors (mR/hr)/(n/cm^2-sec)	Roof Flux (n/cm^2-sec)	Dose Rate (mR/hr)	Input Flux (n/cm^2-sec per mrem/hr)
1	1.49E+01	1.36E+01	1.94E-01	6.53E-04	1.27E-04	5.19E-04
2	1.22E+01	1.11E+01	1.60E-01	3.32E-03	5.30E-04	2.64E-03
3	1.00E+01	9.09E+00	1.47E-01	1.45E-02	2.14E-03	1.16E-02
4	8.18E+00	7.27E+00	1.48E-01	1.01E-01	1.49E-02	8.01E-02
5	6.36E+00	5.66E+00	1.53E-01	2.65E-01	4.06E-02	2.11E-01
6	4.96E+00	4.51E+00	1.51E-01	2.46E-01	3.70E-02	1.96E-01
7	4.06E+00	3.54E+00	1.39E-01	2.83E-01	3.93E-02	2.25E-01
8	3.01E+00	2.74E+00	1.28E-01	6.44E-01	8.28E-02	5.13E-01
9	2.46E+00	2.41E+00	1.25E-01	5.70E-01	7.14E-02	4.53E-01
10	2.35E+00	2.09E+00	1.26E-01	9.24E-01	1.17E-01	7.35E-01
11	1.83E+00	1.47E+00	1.29E-01	1.43E+00	1.84E-01	1.13E+00
12	1.11E+00	8.30E-01	1.17E-01	1.49E+00	1.74E-01	1.19E+00
13	5.50E-01	3.31E-01	6.52E-02	2.13E+00	1.39E-01	1.69E+00
14	1.11E-01	5.72E-02	9.19E-03	3.49E+00	3.21E-02	2.78E+00
15	3.35E-03	1.97E-03	3.71E-03	1.66E+00	6.18E-03	1.32E+00
16	5.83E-04	3.42E-04	4.01E-03	1.98E+00	7.94E-03	1.58E+00
17	1.01E-04	6.50E-05	4.29E-03	1.65E+00	7.07E-03	1.31E+00
18	2.90E-05	1.96E-05	4.48E-03	1.18E+00	5.30E-03	9.42E-01
19	1.01E-05	6.58E-06	4.57E-03	1.60E+00	7.29E-03	1.27E+00
20	3.06E-06	2.09E-06	4.54E-03	1.43E+00	6.46E-03	1.13E+00
21	1.12E-06	7.67E-07	4.37E-03	1.51E+00	6.59E-03	1.20E+00
22	4.14E-07	2.12E-07	3.71E-03	7.42E+01	2.76E-01	5.90E+01
			Totals	**9.68E+01**	**1.26E+00**	**7.70E+01**

Table N.10-5
Summary of ISFSI Surface Activities, 24PHB System

2x10 Back-To-Back Array

Source	Area (cm^2)	Gamma-Ray Activity (γ/sec)	Neutron Activity (neutrons/sec)
Roof	3.73×10^{6}	1.90×10^{11}	2.66×10^{8}
Front 1	1.47×10^{6}	2.74×10^{10}	2.49×10^{8}
Front 2	1.47×10^{6}	2.74×10^{10}	2.49×10^{8}
Side 1	5.30×10^{5}	1.72×10^{10}	2.00×10^{7}
Side 2	5.30×10^{5}	1.72×10^{10}	2.00×10^{7}
Total		2.80×10^{11}	8.05×10^{8}

Two 1x10 Front-To-Front Arrays

Source	Area (cm^2)	Gamma-Ray Activity (γ/sec)	Neutron Activity (neutrons/sec)
Roof	2.06×10^{6}	1.05×10^{11}	1.47×10^{8}
Front	1.47×10^{6}	2.74×10^{10}	2.49×10^{8}
Back	1.47×10^{6}	2.20×10^{9}	7.94×10^{6}
Side 1	2.93×10^{5}	9.49×10^{9}	1.11×10^{7}
Side 2	2.93×10^{5}	9.49×10^{9}	1.11×10^{7}
Total		1.54×10^{11}	4.27×10^{8}

Table N.10-6
MCNP Front Detector Dose Rates for 2x10 Array, 24PHB System

Distance (meters)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
6.10E+00	9.28E+00	1.35E+00	1.06E+01
1.00E+01	5.62E+00	7.93E-01	6.41E+00
2.00E+01	2.21E+00	2.96E-01	2.51E+00
3.00E+01	1.17E+00	1.51E-01	1.32E+00
4.00E+01	7.16E-01	8.92E-02	8.05E-01
5.00E+01	4.80E-01	5.61E-02	5.36E-01
6.00E+01	3.51E-01	3.87E-02	3.90E-01
7.00E+01	2.61E-01	2.91E-02	2.90E-01
8.00E+01	2.06E-01	2.08E-02	2.27E-01
9.00E+01	1.61E-01	1.62E-02	1.77E-01
1.00E+02	1.29E-01	1.25E-02	1.42E-01
2.00E+02	2.38E-02	2.04E-03	2.58E-02
3.00E+02	6.44E-03	7.30E-04	7.17E-03
4.00E+02	2.29E-03	2.93E-04	2.58E-03
5.00E+02	7.41E-04	1.01E-04	8.43E-04
6.00E+02	2.72E-04	5.21E-05	3.24E-04

Table N.10-7
MCNP Back Detector Dose Rates for the Two 1x10 Arrays, 24PHB System

Distance (meters)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
6.10E+00	1.31E+00	1.76E-01	1.49E+00
1.00E+01	9.76E-01	1.31E-01	1.11E+00
2.00E+01	5.83E-01	7.98E-02	6.62E-01
3.00E+01	3.91E-01	4.99E-02	4.41E-01
4.00E+01	2.91E-01	3.64E-02	3.28E-01
5.00E+01	2.22E-01	2.57E-02	2.48E-01
6.00E+01	1.78E-01	1.79E-02	1.96E-01
7.00E+01	1.43E-01	1.49E-02	1.58E-01
8.00E+01	1.18E-01	1.14E-02	1.29E-01
9.00E+01	9.65E-02	8.65E-03	1.05E-01
1.00E+02	8.41E-02	6.65E-03	9.07E-02
2.00E+02	1.79E-02	1.59E-03	1.95E-02
3.00E+02	5.40E-03	5.99E-04	6.00E-03
4.00E+02	1.70E-03	2.04E-04	1.90E-03
5.00E+02	5.76E-04	8.45E-05	6.61E-04
6.00E+02	2.25E-04	4.62E-05	2.71E-04

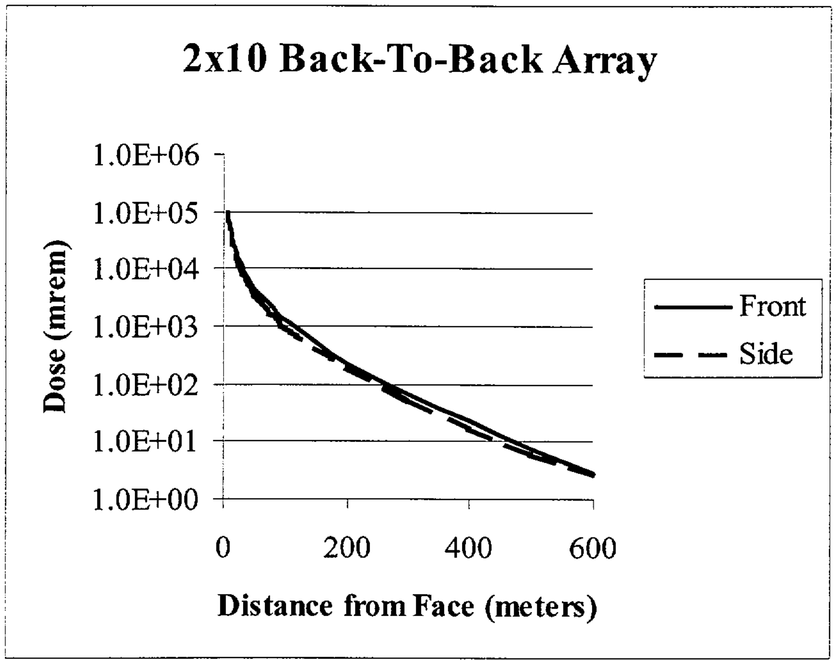
Table N.10-8
MCNP Side Detector Dose Rates, 24PHB System

2x10 Back-to-Back Array

Distance (meters)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
6.10E+00	1.12E+01	3.27E-01	1.15E+01
1.00E+01	5.47E+00	1.94E-01	5.66E+00
2.00E+01	1.71E+00	8.41E-02	1.80E+00
3.00E+01	8.48E-01	5.02E-02	8.98E-01
4.00E+01	5.20E-01	3.27E-02	5.53E-01
5.00E+01	3.52E-01	2.52E-02	3.77E-01
6.00E+01	2.57E-01	1.70E-02	2.74E-01
7.00E+01	1.91E-01	1.32E-02	2.04E-01
8.00E+01	1.55E-01	1.07E-02	1.66E-01
9.00E+01	1.20E-01	8.30E-03	1.28E-01
1.00E+02	9.62E-02	7.06E-03	1.03E-01
2.00E+02	1.93E-02	1.33E-03	2.07E-02
3.00E+02	5.46E-03	4.94E-04	5.96E-03
4.00E+02	1.71E-03	1.72E-04	1.88E-03
5.00E+02	5.74E-04	7.88E-05	6.53E-04
6.00E+02	2.32E-04	5.57E-05	2.88E-04

Two 1x10 Front-To-Front Arrays

Distance (meters)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
6.10E+00	6.89E+00	5.63E-01	7.45E+00
1.00E+01	4.62E+00	3.18E-01	4.94E+00
2.00E+01	1.88E+00	1.22E-01	2.01E+00
3.00E+01	9.79E-01	6.45E-02	1.04E+00
4.00E+01	5.96E-01	4.56E-02	6.42E-01
5.00E+01	4.06E-01	2.72E-02	4.33E-01
6.00E+01	2.91E-01	2.06E-02	3.12E-01
7.00E+01	2.30E-01	1.48E-02	2.45E-01
8.00E+01	1.69E-01	1.20E-02	1.81E-01
9.00E+01	1.37E-01	9.24E-03	1.46E-01
1.00E+02	1.08E-01	7.59E-03	1.16E-01
2.00E+02	2.08E-02	1.70E-03	2.25E-02
3.00E+02	6.65E-03	4.53E-04	7.10E-03
4.00E+02	1.92E-03	1.92E-04	2.11E-03
5.00E+02	6.60E-04	1.64E-04	8.23E-04
6.00E+02	2.66E-04	6.75E-05	3.33E-04



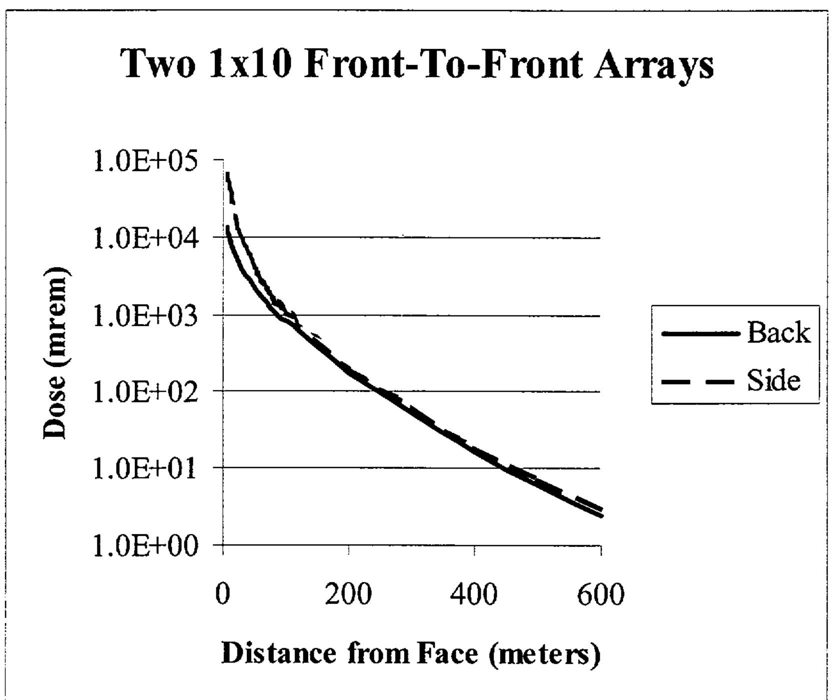


Figure N.10-1
Annual Exposure from the ISFSI as a Function of Distance, 24PHB System

N.11 Accident Analyses

This section describes the postulated off-normal and accident events that could occur during transfer and storage of the NUHOMS®-24HBS and 24HBL DSCs. Sections which do not affect the evaluation presented in Chapter 8 are identified as "No change." Detailed analysis of the events are provided in other sections and referenced herein.

N.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]. Off-normal 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.

The off-normal conditions considered for the NUHOMS®-24PHB DSC are off-normal transfer loads, extreme temperatures and a postulated release of radionuclides.

N.11.1.1 Off-Normal Transfer Loads

No change. The limiting off-normal event is the jammed DSC during loading or unloading from the HSM. This event is described in Section 8.1.2. Other off-normal events are bounded by the jammed DSC.

N.11.1.1.1 Postulated Cause of Event

See Section 8.1.2. The probability of a jammed DSC does not increase with the NUHOMS®-24PHB DSC, since the outside diameter of the DSC is the same as the NUHOMS®-24P DSC.

N.11.1.1.2 Detection of Event

No change. See Section 8.1.2.

N.11.1.1.3 Analysis of Effects and Consequences

A detailed evaluation of this event is presented in Section N.3.6.2 and is summarized below.

As stated in Section N.3.6.2.1, the 24PHB DSC shell and plate thicknesses are identical to the 24P DSC, thus the stresses due to the jammed canister event are identical. In addition, this stress is considered secondary and is enveloped by other handling stresses.

N.11.1.1.4 Corrective Actions

No change. See Section 8.1.2.

N.11.1.2 Extreme Temperatures

No change. The off-normal maximum ambient temperature of 125°F is used in Section 8.1.2.2. For the NUHOMS®-24PHB System, a maximum temperature of 117°F is used. Therefore, the analyses in Section 8.1.2.2 bound the NUHOMS®-24PHB System.

N.11.1.2.1 Postulated Cause of Event

No change. See Section 8.1.2.2.

N.11.1.2.2 Detection of Event

No change. See Section 8.1.2.2.

N.11.1.2.3 Analysis of Effects and Consequences

The thermal evaluation of the NUHOMS®-24PHB System for off-normal conditions is presented in Section N.4.5. The 100°F normal condition with insolation bounds the 117°F case without insolation for the DSC in the TC. Therefore the normal condition maximum temperatures are bounding.

The structural evaluation of the 24PHB DSC for off-normal temperature conditions is presented in Section N.3.6.2.2.

N.11.1.2.4 Corrective Actions

Restrictions for onsite handling of the TC with a loaded DSC under extreme temperature conditions are presented in Technical Specifications 1.2.13 and 1.2.14. There is no change to this requirement as a result of addition of the NUHOMS®-24PHB DSC.

N.11.1.3 Off-Normal Releases of Radionuclides

The NUHOMS®-24PHB DSC is designed and tested to the leak tight criteria of ANSI N14.5 [11.2]. Therefore the estimated quantity of radionuclides expected to be released annually to the environment due to normal or off-normal events is zero.

N.11.1.3.1 Postulated Cause of Event

In accordance with the Standard Review Plan, NUREG-1536 [11.3] and ISG-5 Rev. 1 [11.4] for off-normal conditions, it is conservatively assumed that 10% of the fuel rods fail.

N.11.1.3.2 Detection of Event

Failed fuel rods would go undetected, but are not a safety concern since the canister is designed and tested to leak tight criteria.

N.11.1.3.3 Analysis of Effects and Consequences

The bounding off-normal pressure for the NUHOMS®-24PHB DSC is calculated with the DSC in either the HSM or in the TC in Section N.4.5.4 as 15.2 psig which is below the 20 psig pressure used in the stress analyses. The NUHOMS®-24PHB DSC stresses due to these pressures are below the allowable stresses for off-normal conditions, as shown in Section N.3.6.2.2.

The NUHOMS®-24PHB DSC is designed and tested to the leak tight criteria of ANSI N14.5 [11.2]. Therefore the estimated quantity of radionuclides expected to be released annually to the environment due to normal or off-normal events is zero.

N.11.1.3.4 Corrective Actions

None required.

N.11.1.4 Radiological Impact from Off-Normal Operations

The NUHOMS®-24PHB DSC is designed and tested to the leak tight criteria of ANSI N14.5. The off-normal conditions have been evaluated in accordance with the ASME B&PV code [11.5]. The resulting stresses are below the allowable stresses. There will be no breach of the confinement boundary due to the off-normal conditions. Therefore, the estimated quantity of radionuclides expected to be released annually to the environment due to off-normal events is zero.

N.11.2 Postulated Accidents

N.11.2.1 Reduced HSM Air Inlet and Outlet Shielding

N.11.2.1.1 Cause of Accident

No change. See Section 8.2.1.1.

N.11.2.1.2 Accident Analysis

There are no structural consequences that affect the safe operation of the NUHOMS®-24PHB System resulting from the separation of the HSMs. The thermal effects of this accident results from the blockage of HSM air inlet and outlet openings on the HSM side walls in contact with each other. This would block the ventilation air flow provided to the HSMs in contact from these inlet and outlet openings. The increase in spacing between the HSM on the opposite side from 6 inches to 12 inches, will reduce the ventilation air flow resistance through the air inlet and outlet openings on these side walls, which will partially compensate the ventilation reduction from the blocked side. However, the effect on the NUHOMS®-24PHB DSC, HSM and fuel temperatures is bounded by the complete blockage of air inlet and outlet openings described in Section N.11.2.7. The radiological consequences of this accident are described in the paragraph below.

N.11.2.1.3 Accident Dose Calculations

The off-site radiological effects that result from a partial loss of adjacent HSM shielding is an increase in the air scattered (skyshine) and direct doses from the 12 inch gap between the separated HSMs. The air scattered (skyshine) and direct doses are reduced from the gap between the HSMs that are in contact with each other. On-site radiological effects result from an increase in the direct radiation during recovery operations and increased skyshine radiation. Table 8.2-2 shows the comparisons of the increased dose rate as a function of distance due to the reduced shielding effects of the adjacent HSM for the 24P DSC with 5-year cooled design basis fuel. Table N.11-1 provides a similar table for the NUHOMS®-24PHB System. For the NUHOMS®-24PHB System, the dose received by a person located 100 meters away from the NUHOMS® installation for eight hours a day for five days (estimated recovery time) would be 11 mrem. The increased dose to an off-site person for 24 hours a day for five days located 600 meters away would be about 0.08 mrem. Thus, the 10CFR72 requirements for this postulated event are met.

N.11.2.1.4 Corrective Actions

No change. See Section 8.2.1.4.

N.11.2.2 Earthquake

N.11.2.2.1 Cause of Accident

No change. See Section 8.2.3.1.

N.11.2.2.2 Accident Analysis

Section 8.2.3.2 describes the analyses performed to demonstrate that the NUHOMS® System withstands the design basis seismic event. Section N.3.7.3 presents the changes to this analysis resulting from the addition of NUHOMS®-24PHB DSC. As documented in Section N.3.7.3, the HSM and the TC seismic evaluation bounds the 24PHB DSC payload. The results of this analysis show that the seismic stresses are well below allowables and, thus, the leak-tight integrity of the canister is not compromised. The basket stresses are also low and do not result in deformations that would prevent fuel from being unloaded from the canister.

N.11.2.2.3 Accident Dose Calculations

The design earthquake does not damage the NUHOMS®-24PHB System. Hence, no radioactivity is released and there is no associated dose increase due to this event.

N.11.2.2.4 Corrective Actions

After a seismic event, the NUHOMS® HSMs and TC would be inspected for damage. Any debris would be removed. An evaluation would be performed to determine if the cask were still within the licensed design basis.

N.11.2.3 Extreme Wind and Tornado Missiles

N.11.2.3.1 Cause of Accident

The determination of the tornado wind and tornado missile loads acting on the HSM are detailed in Section 8.2.2.

N.11.2.3.2 Accident Analysis

An evaluation of the HSM and TC with respect to tornado winds and tornado missiles is presented in Section 8.2.2. As discussed in Section N.3.7.2, the analysis presented in Section 8.2.2 is bounding for the NUHOMS®-24PHB System.

N.11.2.3.3 Accident Dose Calculations

The NUHOMS®-24PHB DSC is designed and tested as a leak-tight containment boundary. Tornado wind and tornado missiles do not breach the containment boundary. Therefore, there is no increase in site boundary dose due to this accident event.

N.11.2.3.4 Corrective Actions

After excessive high winds or a tornado, the HSM's and TC would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the system was still within the licensed design basis.

N.11.2.4 Flood

N.11.2.4.1 Cause of Accident

No change. See Section 8.2.4.1.

N.11.2.4.2 Accident Analysis

The HSM is evaluated for flooding in Section 8.2.4. This evaluation is bounding for the NUHOMS®-24PHB DSC as described in Section N.3.7.4. The canister is designed and tested to be leak tight. The stresses in the canister due to the design basis flood are well below the allowable stresses for Service Level C of the ASME Code Subsection NB [11.5]. Therefore, the NUHOMS®-24PHB DSC will withstand the design basis flood without breach of the confinement boundary.

N.11.2.4.3 Accident Dose Calculations

The radiation dose due to flooding of the HSM is negligible. The NUHOMS®-24PHB DSC is designed and tested as a leak tight containment boundary. Flooding does not breach the containment boundary. Therefore radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the encroaching flood water. See also Section 8.2.4.3.

N.11.2.4.4 Corrective Actions

No change. See Section 8.2.4.4.

N.11.2.5 Accidental TC Drop

N.11.2.5.1 Cause of Accident

See Section N.3.7.5.1.

N.11.2.5.2 Accident Analysis

The evaluation of the NUHOMS®-24PHB DSC shell and basket assemblies due to an accidental drop is presented in Section N.3.7.5. As documented in Section N.3.7.5, the TC has been evaluated for a payload that bounds the 24PHB DSC payload, and therefore is not affected by the 24PHB DSC. As shown in Section N.3.7.5, the DSC shell and basket stress intensities are within the appropriate ASME Code Service Level D allowable limit, thus the DSC basket and shell maintain their structural integrity.

For the case of a liquid neutron shield, a complete loss of neutron shield is evaluated at the 100°F ambient condition with full solar load. It is conservatively assumed that the neutron shield jacket is still present but all the liquid is lost. The maximum DSC shell temperature is 378°F. The maximum cask inner shell, cask outer shell, and cask neutron shield jacket temperatures are bounded by analyses presented in Section 8.1.3.3. These bounding temperatures are 393°F, 384°F and 238°F respectively. The DSC shell temperatures and hence fuel cladding temperature are bounded by the HSM blocked vent case shown in Section N.4. Accident thermal conditions, such as loss of the liquid neutron shield, need not be considered in the load combination evaluation. Rather the peak stresses resulting from the accident thermal conditions must be less than the allowable fatigue stress limit for 10 cycles from the appropriate fatigue design curves in Appendix I of the ASME Code [11.5]. Similar analyses of other NUHOMS® TCs have shown that fatigue is not a concern. Therefore, these stresses in a TC with a liquid neutron shield need not be evaluated for the accident condition.

N.11.2.5.3 Accident Dose Calculations for Loss of Neutron Shield

The postulated accident condition for the on-site TC assumes that after a drop event, the water in the neutron shield is lost for TC with liquid neutron shield. To be conservative, neutron shield from the solid neutron shield TC is also assumed to be lost. The loss of neutron shield is modeled using the normal operation models described in Section N.5.4 by replacing the neutron shield with air.

The accident condition dose rates are summarized in Table N.11-2 and Figure N.11-1 for the 24PHB DSC loaded with design basis fuel plus BPRAs.

A comparison of the results in Table N.11-2 and Table N.5-4 demonstrates a maximum cask surface contact dose rate increase from 1.22E+03 mrem/hr to 7.06E+03 mrem/hr. These dose rates are approximately 3.3 times those reported in Section 8.2.5.3.2. Therefore, one would expect that the additional dose rate to an average on-site worker at an average distance of fifteen feet would also increase from 310 mrem/hr to 1023 mrem/hr. Similarly the exposure to off-site individuals at a distance of 2000 feet would also be expected to increase from 0.04 mrem for an assumed eight hour exposure to 0.13 mrem. This exposure is still well within the limits of 10CFR72 for an accident condition.

N.11.2.5.4 Corrective Action

No change. See Section 8.2.5.4.

N.11.2.6 Lightning

No change. The evaluation presented in Section 8.2.6 is not affected by the addition of the NUHOMS®-24PHB DSC to the NUHOMS® System.

N.11.2.7 Blockage of Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the HSM ventilation air inlet and outlet openings on the HSM side walls.

N.11.2.7.1 Cause of Accident

No change. See Section 8.2.7.1.

N.11.2.7.2 Accident Analysis

This event is evaluated in Section 8.2.7.2 . The section below describes the additional analyses performed to demonstrate the acceptability of the system with the NUHOMS®-24PHB DSC. The thermal evaluation of this event is presented in Section N.4.6. The temperatures determined in Section N.4.6 are used in the structural evaluation of this event, which is presented in Sections N.3.7.7 and N.3.4.4.3.

N.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. This is bounded by the evaluation of the NUHOMS® System with the 24P canister. See Section 8.2.7.3.

N.11.2.7.4 Corrective Action

No change. See Section 8.2.7.4.

N.11.2.8 DSC Leakage

The NUHOMS®-24PHB DSC is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials. The analyses of normal, off-normal, and accident conditions have shown that no credible conditions can breach the DSC shell or fail the double seal welds at each end of the DSC. The NUHOMS®-24PHB DSC is designed and tested to be leak tight. Therefore DSC leakage is not considered a credible accident scenario. See Section N.7.3.

N.11.2.9 Accident Pressurization of DSC

N.11.2.9.1 Cause of Accident

The bounding internal pressurization of the NUHOMS®-24PHB DSC is postulated to result from cladding failure of the spent fuel in combination with the blocked inlet and outlet vents and the consequent release of spent fuel rod fill gas and free fission gas. The evaluation conservatively assumes that 100% of the fuel rods have failed.

N.11.2.9.2 Accident Analysis

The pressure due to this case is evaluated in Section N.4.6. The maximum pressure calculated is 62.8 psig. The accident pressure is conservatively assumed to be 68 psig in the structural load combinations presented in Table N.2-8.

N.11.2.9.3 Accident Dose Calculations

There is no increase in dose rates as a result of this event.

N.11.2.9.4 Corrective Actions

This is a hypothetical event. Therefore no corrective actions are required. The canister is designed to withstand the pressure as a Level D condition. There will be no structural damage to the canister or leakage of radioactive material as a result of this event.

N.11.2.10 Fire and Explosion

N.11.2.10.1 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 NUHOMS®-24PHB System based on a fuel fire. The source of fuel is postulated to be from a ruptured fuel tank of the TC transporter tow vehicle. The bounding capacity of the fuel tank is 300 gallons and the bounding hypothetical fire is an engulfing fire around the TC. Direct engulfment of the HSM is highly unlikely. Any fire within the ISFSI boundary while the DSC is in the HSM would be bounded by the fire during TC movement. The HSM concrete acts as a significant insulating fire wall to protect the 24PHB-DSC from the high temperatures of the fire.

N.11.2.10.2 Accident Analysis

The evaluation of the hypothetical fire event is presented in Section N.4.6.3. The fire thermal evaluation is performed primarily to demonstrate the confinement integrity and fuel retrievability of the 24PHB-DSC. This is assured by demonstrating that the DSC temperatures and internal pressures will not exceed those of the blocked vent condition (see Section N.11.2.7) during the fire scenario. Peak temperatures for the NUHOMS®-24PHB System components are summarized in Section N.4.6.3.

N.11.2.10.3 Accident Dose Calculations

The 24PHB-DSC confinement boundary will not be breached as a result of the postulated fire/explosion scenario. Accordingly, no 24PHB-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 TC neutron shielding should the fire occur while the 24PHB-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 N.11.2.5.3 for evaluation of the dose consequences of a cask drop.

N.11.2.10.4 Corrective Actions

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

N.11.3 References

11.1 American Nuclear Society, ANSI/ANS-57.9, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1992.

11.2 N14.5-1997, "Leakage Tests on Packages for Shipment," February 1998.

11.3 NUREG-1536, "Standard Review Plan for dry Storage Casks, Final Report," US Nuclear Regulatory Commission, January 1997.

11.4 ISG-5, Rev. 1, Confinement Evaluation.

11.5 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, 1983 including W85 addenda.

Table N.11-1
Comparison of Total Dose Rates for HSM with and without Adjacent HSM Shielding Effects

Distance from Nearest HSM Wall, 2x10 Array (meters)	Normal Case Dose Rate(1) (mrem/hr)	Accident Case Dose Rate(1) (w/o adjacent HSM shielding effects) (mrem/hr)
10	6.4	12.8
100	0.14	0.28
500	8.4×10^{-4}	1.7×10^{-3}
600	3.2×10^{-4}	6.4×10^{-4}

(1) Air scattered plus direct radiation

Table N.11-2
TC Bounding Accident Dose Rate Results

Cask			
	Side (mrem/hr)	**Top (mrem/hr)**	**Bottom (mrem/hr)**
Neutron	6.13E+03	4.90E+02	4.91E+03
Gamma	9.26E+02	2.75E+02	7.03E+02
Total	7.06E+03	7.65E+02	5.61E+03

1-Meter from Cask			
	Side (mrem/hr)	**Top (mrem/hr)**	**Bottom (mrem/hr)**
Neutron	2.25E+03	1.68E+02	4.90E+02
Gamma	3.27E+02	3.72E+01	1.47E+02
Total	2.58E+03	2.05E+02	6.37E+02

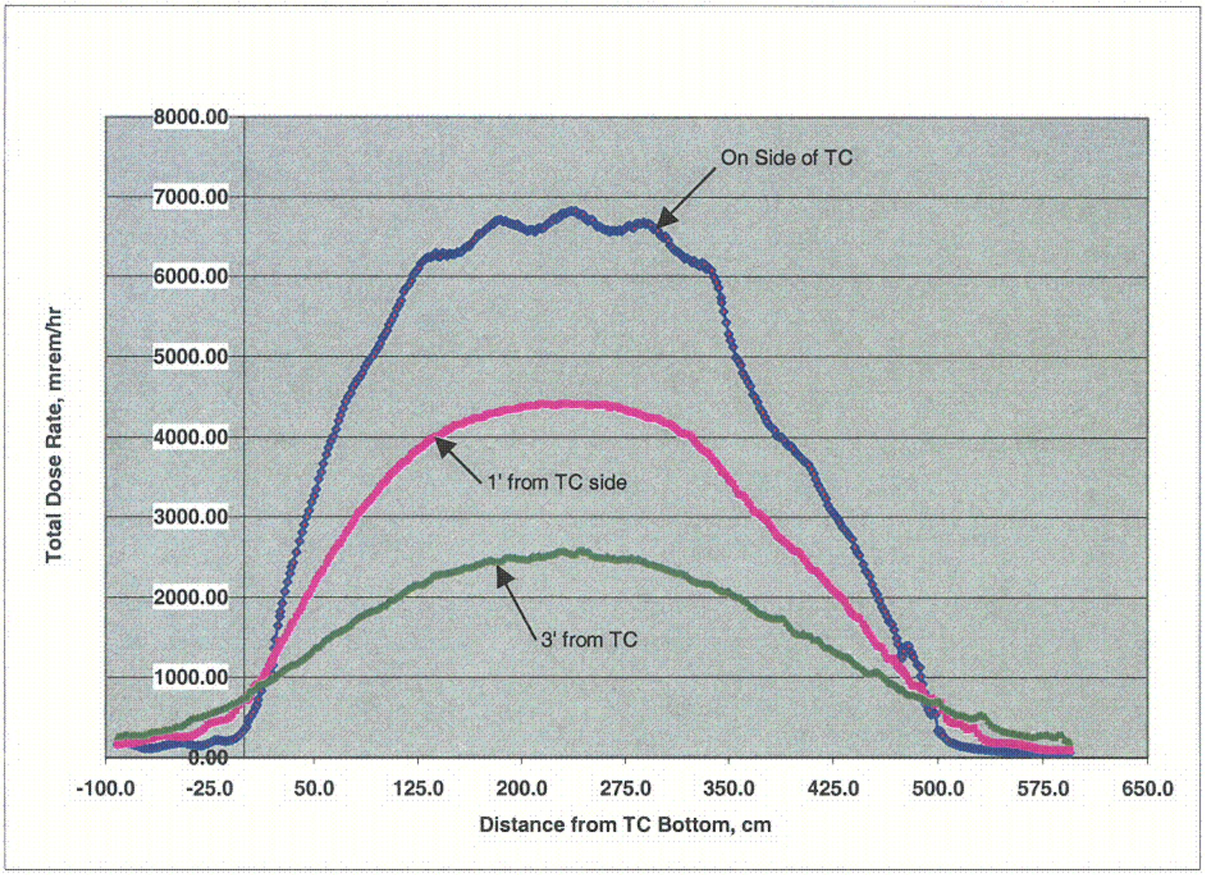


Figure N.11-1
TC Bounding Accident Dose Rate Distribution

C01

N.12 Conditions for Cask Use - Operating Controls and Limits or Technical Specifications

Attachment B of this application provides the suggested changes to the Technical Specifications due to the addition of high burnup fuel to the NUHOMS® System.

N.13 Quality Assurance

No change.