72-1004

TRANSNUCLEAR

March 7, 2003
NUH03-03-13

Mr. L. Raynard Wharton
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike M/S O13-D-13
Rockville, MD 20852

Subject: Submittal of Revision 3 of Application for Amendment No. 6 to the NUHOMS® Certificate of Compliance No. 1004 (TAC NO. L23370)

Reference:

1. Telecon with the NRC staff on February 25, 2003 to Provide Additional Clarification of the ANISN Methodology Used to Determine Design Basis Source Terms for Generating Fuel Qualification Tables.

2. Revision 2 of Application for Amendment No. 6 to the NUHOMS® Certificate of Compliance No. 1004 (TAC NO. L23370).

Dear Mr. Wharton:

In response to a verbal request from the staff (Reference 1), Transnuclear, Inc. (TN) herewith provides the requested clarification and submits Revision 3 of Application for Amendment No. 6 to the NUHOMS® Certificate of Compliance No. 1004.

Also included are updated pages of Chapter 7 of NUHOMS® FSAR Revision 6 to provide a similar clarification for the shielding analysis of the NUHOMS®-24P and –52B systems.

Please replace the affected pages of the Revision 2 application (Reference 2) with the changed pages submitted herewith.

Should you or your staff require additional information to support review of this application, please do not hesitate to contact me at 510-744-6053.

Sincerely,

UBChopra

U. B. Chopra

Licensing Manager

Docket 72-1004

Enclosure: Eight Copies of Revision 3 of Application for Amendment No. 6 to the NUHOMS® Certificate of Compliance No. 1004 (Changed Pages Only).

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NMSS01

7.2.3 Fuel Qualification Tables

The purpose of this section is to document the methodology used to determine the cooling times required for PWR and BWR fuel assemblies, with various burnups and initial enrichments, for storage in the standardized NUHOMS® 24P and 52B systems. An acceptable fuel assembly meets the cladding temperature limits, overall heat generation limit, and design basis HSM and Transfer Cask (TC) surface dose rates presented in the Section 7.3.2. The methodology is based on preserving the following parameters for design basis fuel: the cladding temperature, the total dose rate on the exterior of the HSM and the TC radial surface thereby, assuring that the temperatures and dose rates calculated on and around the HSM and TC, using the design basis fuel source terms, remain bounding. The HSM roof surface dose rate is chosen for this evaluation because it represents the largest contribution to the exposure received by members of the public, both offsite and onsite. The TC radial surface is chosen because it represents the greatest surface area and dose rate on the TC surface to which workers are exposed during fuel loading operations.

For a wide range of assembly burnups and initial enrichments, the OCRWM Characteristics Database (CDB) [7.14] is used to determine the required cooling time to meet the decay heat and surface dose rate criteria described above for this evaluation. The results of this evaluation are provided in Fuel Qualification Table 3.1-8a and Table 3.1-8b.

Methodology

The standard NUHOMS® design basis fuel assemblies have a decay heat of 1.0 kW/assy for a PWR assembly and 0.37 kW/assy for a BWR assembly. A fuel assembly with a decay heat less than these design basis values results in maximum HSM, TC and DSC component temperatures less than those listed in Section 8.1.3. The maximum allowable fuel cladding temperature is a function of both the post irradiation cooling time and the fuel burnup. Allowable decay heats as a function of cooling time and burnups are based on the criteria in reference [7.15]. These decay heats result in maximum fuel cladding temperatures that are less than the corresponding cladding temperature limit.

Surface neutron dose rates are assumed to be directly proportional to the total neutron sources in the assemblies. The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. *For the ranges of burnups, initial enrichments, and cooling times in the fuel qualification tables,* ^{244}Cm *represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein. To account for the fact that the original analyses used a different neutron spectrum, the variation in the spectrum is accounted for by applying a 5% safety margin to all neutron results. Surface gamma dose rates are determined for the HSM and TC surfaces using the actual gamma spectrum applicable for each case.*

The BWR heavy metal weight used is 0.198 MTU per assembly. The PWR heavy metal weight used is 0.475 MTU per assembly to bound existing PWR fuel designs. Note that *this is an increase over that used for the design basis shielding analysis in Section* 7.3. *The increase in heavy metal loading is accounted for by increased cooling time to offset the increase in the source terms.*

The design basis HSM roof dose rate from Table 7.3-2 *is 48.6 mrem/hr. Although not a regulatory or operational limit, the HSM roof dose rate of 48.6 mrem/hr is an appropriate acceptance criterion for the purposes of this evaluation. The HSM surface dose rate criterion assures the design basis offsite dose rates remain bounding.*

The design basis transfer TC radial dose rate from Table 7.3-2 *is 591.8 mrem/hr. Like the HSM dose rate, this is not a regulatory or operational limit, but is considered an appropriate acceptance criterion for the purposes of this evaluation. The TC surface dose rate criterion assures the design basis occupational exposures remain bounding.*

For conservatism, all required cooling times are rounded to the next higher integral year. None of the safety margins in the design basis analyses are reduced in this evaluation. The acceptance criteria for this evaluation are that the cladding temperature is less than the applicable cladding temperature limit, that the HSM concrete temperatures are maintained, and that the surface dose rates are less than the design basis surface dose rates for both the HSM roof and TC side centerline.

PWR Fuel Evaluation – Decay Heat

The CDB provides decay heats in the units watts/MTIHM. For an assembly heavy metal loading of 0.475 MTU, the per assembly decay heat is determined using the relation,

$$DecayHeat = \frac{0.475 \cdot Q_{CDB}}{1000}$$

where Q_{CDB} *is the CDB decay heat in units of watts/MTIHM for at given burnup, initial enrichment and cooling time. The calculated decay heat is checked against the allowable decay heat as given in* Table 7.2-5. *If the calculated decay heat is too high then the cooling time is increased until the decay heat limit is met.*

Table 7.2-5
PWR Allowable Decay Heat Versus Cooling Time and Burnup

Cooling Time (years)	Allowable Decay Heat (kW/assy)	
	≤40 GWd/MTU	> 40 GWd/MTU ≤45 GWd/MTU
3	1.00	0.97
4	1.00	0.97
5	1.00	0.97
6	0.99	0.93
7	0.86	0 83
8	0.85	0.82
9	0.84	0.81
10	0.84	0.80
11	0.83	0.80
12	0.82	0.79
13	0.81	0 78
14	0.81	0.78
15	0.81	0 78
16	0.80	0.77
17	0.79	0.77
18	0.79	0 76
19	0.78	0 76
20	0.78	0 76
21	0.78	0 75
22	0.77	0.75
23	0.77	0 75
24	0.77	0.75
25	0.77	0.74
26	0.76	0.74
27	0.76	0.74
28	0.76	0.74
29	0.75	0.73
30	0.75	0.73

PWR Fuel Evaluation – Neutron Dose Rate

The CDB provides neutron sources in the units neutrons/s/MTIHM, for a given burnup, initial enrichment and cooling time. Neutron dose rates are determined for both the HSM roof and the TC side surface for every entry in the fuel qualification table. The HSM and TC neutron dose rates are each determined using the relation,

$$DoseRate = 1.05 \cdot \left(Dose_{DesignBasis}\right)\frac{0.475\left(Source_{CDB}\right)}{Source_{Section7.2}}$$

where $Dose_{DesignBasis}$ *is the HSM roof or TC sidewall dose rate for design basis fuel (0.4 mrem/hr HSM and 163.9 mrem/hr TC,* Table 7.3-2*),* $Source_{CDB}$ *is the neutron source from*

CDB for a given assembly, and $Source_{Section\ 7.2}$ is the per assembly source for the design basis fuel (2.23E-08 n/s/assy). As discussed above, the calculated dose rates include a 5% safety factor to account for the spectral differences relative to the design basis neutron source listed in Table 7.2-2.

PWR Fuel Evaluation – Gamma Dose Rate

The CDB provides gamma spectra in the units γ/s/MTIHM for each of 18 energy groups for a given burnup, initial enrichment and cooling time. The HSM and TC gamma dose rates are determined by (1) Mapping the CDB gamma source and spectrum into the Cask-81[7.7] gamma energy structure used in the shielding evaluation; (2) Multiplying by the number of assemblies in the DSC and by the heavy metal weight; (3) Dividing by the fuel region volume; (4) Multiplying the resulting source in each energy group by a response function to determine the dose rate contribution from the group to the total surface dose rate; and (5) Summing the dose rate contribution from each energy group. Each of these steps is described in detail below.

The 18 gamma-ray group CDB energy spectrum has been mapped to the Cask-81 18 group energy spectrum used in the ANISN shielding models. The energy group mapping is performed by assuming that the particles in each group are evenly distributed in logarithmic energy space. The total source strength is conserved. The formulae used to map the CDB energy structure into the structure used in the shielding evaluation are shown in Table 7.2-6.

Table 7.2-6
Formulae for Mapping Gamma Source Spectra

CDB Structure Group	CDB Structure E_{mean} (MeV)	Cask-81 Group Structure Group	Cask-81 Group Structure E_{upper} (MeV)	Mapping Formula CDB → Cask-81 Group Structure
a	9.500	23	10.000	a
b	7.000	24	8.000	0.722b
c	5.000	25	6.500	0.278b+0.450c
d	3.500	26	5.000	0.550c
e	2.750	27	4.000	d
f	2.250	28	3.000	e
g	1.750	29	2.500	f
h	1.250	30	2.000	0.648g
i	0.850	31	1.660	0.352g+0.297h
j	0.575	32	1.330	0.703h
k	0.375	33	1.000	0.626i
l	0.225	34	0.800	0.374i+0.349j
m	0.125	35	0.600	0 651j+0.290k
n	0.085	36	0.400	0.710k
o	0.058	37	0.300	0.585l
p	0.038	38	0.200	0.415l+m
q	0.025	39	0.100	n+0.762o
r	0.010	40	0.050	0 238o+p+q+r

The mapped gamma source is multiplied by the number of assemblies in each DSC (24) and by the design heavy metal weight (0.475 MTU) to determine the total source in each energy group inside the DSC for the case being evaluated. This value is then divided by the fuel region volume in the ANISN models, 8,073,120 cm^3 to determine the total volumetric source ($\gamma/s/cm^3$) in each energy group. The source-to-dose rate response functions, described below, are then applied to the source spectrum to determine gamma dose rates on the HSM and TC surfaces.

The ANISN discrete-ordinates computer code is used to generate source-to-dose rate response functions to convert the group sources to surface dose rates on the HSM and TC. The ANISN models used are identical to those used in Section 7.3.2. *Eighteen runs were performed for the HSM roof geometry and an additional 18 runs were performed for the TC side geometry. Each run includes a unit source (1 $\gamma/s/cm^3$) in a single energy group. The other input parameters, including geometry, materials, and flux-to-dose rate factors, are unchanged relative to the design basis analysis models. The gamma dose rate reported at the HSM (or TC) surface in each run represents the contribution from that energy group (per unit volumetric source) to the total gamma dose rate. The ANISN results, shown in* Table 7.2-7, *represent a response function which allows the gamma* *dose rates to be determined for each fuel qualification case, including spectral effects, without the need to perform additional ANISN runs. The total surface gamma dose rate is then calculated by multiplying the source in each group (discussed above) by the response function for that group, and summing the result for all eighteen groups*

Table 7.2-7

PWR HSM and TC Unit Gamma Source Response Functions

(mrem/hr per γ/s/cm^3)

Cask-81 Group	HSM Response Function	TC Response Function
23	6.55E-05	2.39E-05
24	4.86E-05	2.94E-05
25	3.04E-05	3.13E-05
26	1.63E-05	2.96E-05
27	7.67E-06	2.47E-05
28	3.03E-06	1.72E-05
29	1.28E-06	1.04E-05
30	4.45E-07	4.88E-06
31	1.48E-07	1.85E-06
32	3.34E-08	3.51E-07
33	5.32E-09	2.72E-08
34	9.66E-10	1.06E-09
35	9.33E-11	5.29E-13
36	3.06E-12	3 25E-18
37	1.95E-13	5.80E-20
38	1.88E-15	1.13E-32
39	3.42E-27	0.00E+00
40	0.00E+00	0.00E+00

The goal of this evaluation, however, is simply to compare the dose rates for various sets of fuel parameters to the dose rate for the design basis fuel parameters. The design basis dose rates reported in Table 7.3-2, *therefore, are scaled by the ratio of the calculated dose rate for each case to the calculated dose rate for the design basis case. The end result of this effort is to specifically account for the gamma spectrum for every case on the fuel qualification table.*

$$DoseRate = \left(Dose_{T7\,3\,2}\right)\frac{0.475\left(Dose_{case}\right)}{0.472\left(Dose_{DesignBasis}\right)}$$

Where $Dose_{T7\,3\,2}$ is the HSM roof or TC side gamma dose rate for design basis fuel (48.2 mrem/hr for the HSM and 427.9 mrem/hr for the TC) from Table 7.3-2, *$Dose_{Case}$ is the gamma dose rate determined by ANISN using the* Table 7.2-7 *response functions for the case being evaluated, and $Dose_{DesignBasis}$ is the dose rate determined above for the design basis source. The ratio of 0.475/0.472 scales the design basis dose rate up to a value* *consistent with a heavy metal weight of 0.475 MTU. The following is an example of the dose rate evaluation methodology for the design basis (40 GWd/MTU, 4.0 wt. %, 5-year cooled) case. The result of this evaluation is $Dose_{DesignBasis}$ as is used in the above equation.*

Table 7.2-8 *shows the calculation of $Dose_{DesignBasis}$ for both the HSM and TC. The first two columns, "CDB Energy" and "CDB (γ/s/MTIHM)" are the CDB energy group mean energies and gamma source, respectively for 4.0 wt. % ^{235}U, 40,000 MWd/MTU and 5-year cooling time. The "Shielding (γ/s/DSC)" column represents the total gamma source in a single DSC mapped into the 18 energy groups used in the shielding evaluation. The values in this column are determined by multiplying the CDB source by the heavy metal weight (0.475 MTU), the number of assemblies (24) and by the mapping function in* Table 7.2-6 *for the Cask-81 energy groups as labeled in the "Cask-81 Shielding Energy Group Column". As an example, the third value in the Cask-81 shielding energy group column (Cask-81 Group 25) is equal to (0.475*24)[(.278*1.99e6)+(.450*1.73e7)] = 9.485e7 γ/s/DSC where b=1.99E+06 for CDB energy 7.00 MeV and c=1.73E+07 for CDB energy of 5.00 MeV. The corresponding contribution to the HSM roof dose rate is then equal to the DSC source divided by the source volume and multiplied by the response function, (9.485e7/8,073,120)*3.04e-5 = 3.573e-4. This process is completed for each energy group and the sum shown at the bottom of* Table 7.2-8. *For the design basis PWR parameters of 4.0 wt. % ^{235}U, 40,000 MWd/MTU, and 5-year cooling the calculated "$Dose_{DesignBasis}$" for the HSM and TC are 82.48 mrem/hr and 846.3 mrem/hr, respectively.*

Table 7.2-8
Sample PWR Dose Rate Eva luation

CDB Energy (MeV)	CDB (γ/s/MTIHM)	Cask-81 Shielding Energy Group	Shielding (γ/s/DSC)	HSM Dose	Cask Dose
1.00E-02	3.63E+15	23	2.606E+06	2.115E-05	7.712E-06
2.50E-02	8.55E+14	24	1.638E+07	9.861E-05	5.975E-05
3.75E-02	9.17E+14	25	9.485E+07	3.573E-04	3.674E-04
5.75E-02	7.19E+14	26	1.082E+08	2.188E-04	3.973E-04
8.50E-02	4.66E+14	27	2.016E+11	1.916E-01	6.167E-01
1.25E-01	4.64E+14	28	1.578E+12	5.923E-01	3.352E+00
2.25E-01	3.88E+14	29	5.018E+13	7.931E+00	6.473E+01
3.75E-01	2.29E+14	30	6.439E+13	3.552E+00	3.895E+01
5.75E-01	6.18E+15	31	2.123E+15	3.884E+01	4.851E+02
8.50E-01	1.43E+15	32	4.942E+15	2.041E+01	2.150E+02
1.25E+00	6.17E+14	33	1.023E+16	6.745E+00	3.449E+01
1.75E+00	8.72E+12	34	3.071E+16	3.675E+00	4.017E+00
2.25E+00	4.40E+12	35	4.664E+16	5.391E-01	3.057E-03
2.75E+00	1.38E+11	36	1.852E+15	7.008E-04	7.447E-10
3.50E+00	1.77E+10	37	2.589E+15	6.241E-05	1.861E-11
5.00E+00	1.73E+07	38	7.131E+15	1.657E-06	1.003E-23
7.00E+00	1.99E+06	39	1.155E+16	4.890E-18	0.000E+00
9.50E+00	2.29E+05	40	6.351E+16	0.000E+00	0.000E+00
Total	1.591E+16	-	1.814E+17	8.248E+01	8.463E+02

Similar evaluations were performed for each entry (cooling time) in the fuel qualification table to verify that the decay heat and dose rate determinations for PWR assemblies with burnups ranging from 10,000 MWd/MTU to 45,000 MWd/MTU and initial enrichments from 2.0 wt. % ^{235}U to 4.0 wt. % ^{235}U.

BWR Fuel Evaluation

The BWR fuel evaluations are performed in the same manner to those described above for the PWR fuel. BWR assemblies are evaluated with burnups ranging from 15,000 MWd/MTU to 45,000 MWd/MTU and initial enrichments ranging from 2.0 wt. % to 4 wt. % ^{235}U. The calculated decay heat is checked against the allowable decay heat given in Table 7.2-9. *As with the PWR fuel if the calculated decay heat is too high then the cooling time is increased until the decay heat limit is met.*

Table 7.2-9
BWR Allowable Decay Heat Versus Cooling Time

Cooling Time (years)	Allowable Decay Heat (kW/assy)
3	0.370
4	0.370
5	0.370
6	0 350
7	0.310
8	0.310
9	0.296
10	0.296
11	0.296
12	0.296
13	0.296
14	0.296
15	0.290
16	0.290
17	0.290
18	0.290
19	0.280
20	0.280
21	0.280
22	0.280
23	0.280
24	0.280
25	0.280
26	0.280
27	0.280
28	0.270
29	0.270
30	0.270

The HSM and TC source-to-dose rate response functions are calculated for BWR fuel similar to the PWR fuel. Because the design basis shielding evaluation in Section 7.3.2 *only include fuel region number densities for PWR fuel, number densities for BWR fuel are required. The number densities generated for the 52B DSC evaluation are based on a GE BWR/4-6 7 X 7 GE-2 fuel assembly. This assembly was chosen because its heavy metal weight of 0.1947 MTU is closest to the BWR design basis of 0.198 MTU. The actual fuel assembly chosen has no impact on the results because the calculated dose rates are used only for comparison.*

*The BWR fuel assembly properties used in the number density calculations are taken from the CDB [*7.14*]. These include a heavy metal weight of 0.1947 MTU/assy, a* *zircaloy spacer weight of 2.029 kg/assy, an active fuel length of 144 in, a rod diameter of 0.563 in, a zircaloy cladding thickness of 0.032 in, and 49 fueled rods per assembly. The fuel channels and the DSC basket materials have been neglected. The mass and atom density for BWR fuel region is shown in* Table 7.2-10.

Table 7.2-10
BWR Fuel Region Densities

Element	Atomic Mass (g/mol)	Mass Density (g/cc)	Atom Density (atoms/b•cm)
O	15.9994	0.169	6.348E-03
Zr	91.22	0.275	1.818E-03
U-235	235.04	0.050	1.285E-04
U-238	238.05	1.204	3.046E-03

With the exception of the fuel region atom densities, the ANISN source-to-dose rate response function models for the BWR fuel are identical to those for PWR fuel. The HSM and TC gamma dose rate response functions for the BWR fuel are shown in Table 7.2-11. *For the $Dose_{DesignBasis}$ calculation, the BWR fuel parameters of 2.65 wt. % ^{235}U, 35,000 MWd/MTU, and 5-years cooling are used and the calculated gamma "$Dose_{DesignBasis}$" for the HSM and TC are 83.94 mrem/hr and 863.2 mrem/hr, respectively. These fuel parameters were also used to lookup the design basis neutron source per assembly from the CDB. This value is used in the evaluation to allow an "apples-to-apples" comparison of the neutron source for different fuel parameters. The $Dose_{DesignBasis}$ neutron and gamma sources for BWR assemblies are 1.83E+08 n/s/assy and 2.63E+15 γ/s/assy, respectively. Note that the design basis neutron source term is calculated using a different methodology than are used herein. The design basis BWR neutron source listed above is taken from the CDB for the design basis fuel parameters. This was done to provide an "apples-to-apples" comparison in the neutron scaling calculations.*

Table 7.2-11

BWR HSM and TC Unit Source Response Functions

(mrem/hr per $\gamma/s/cm^3$)

Cask-81 Group	HSM Response Function	TC Response Function
23	7.69E-05	2.80E-05
24	5.71E-05	3.45E-05
25	3.58E-05	3.68E-05
26	1.93E-05	3.50E-05
27	9.10E-06	2.93E-05
28	3.61E-06	2.04E-05
29	1.52E-06	1.24E-05
30	5.32E-07	5.83E-06
31	1.76E-07	2.20E-06
32	3.96E-08	4.17E-07
33	6.28E-09	3.22E-08
34	1.13E-09	1.24E-09
35	1.08E-10	6.10E-13
36	3.48E-12	3.70E-18
37	2.20E-13	6.55E-20
38	2.11E-15	1.28E-32
39	3.90E-27	0.00E+00
40	0.00E+00	0.00E+00

ATTACHMENT C

List of Changed Appendix N Pages

- *N.2-20*
- *N.5-1*
- *N.5-2*
- *N.5-2a (New Page)*
- *N.5-3 thru N.5-7*
- *N.5-7a thru N.5-7c (New Pages)*
- *N.5-13 thru N.5-15*
- *N.5-17*
- *N.5-37*
- *N.5-37a thru N.5-37c (New Pages)*
- *N.5-39*
- *N.5-42*
- *N.5-43*
- *N.5-60a thru N.5-60f (New Pages)*
- *N.5-81*
- *N.5-82*
- *N.10-2*
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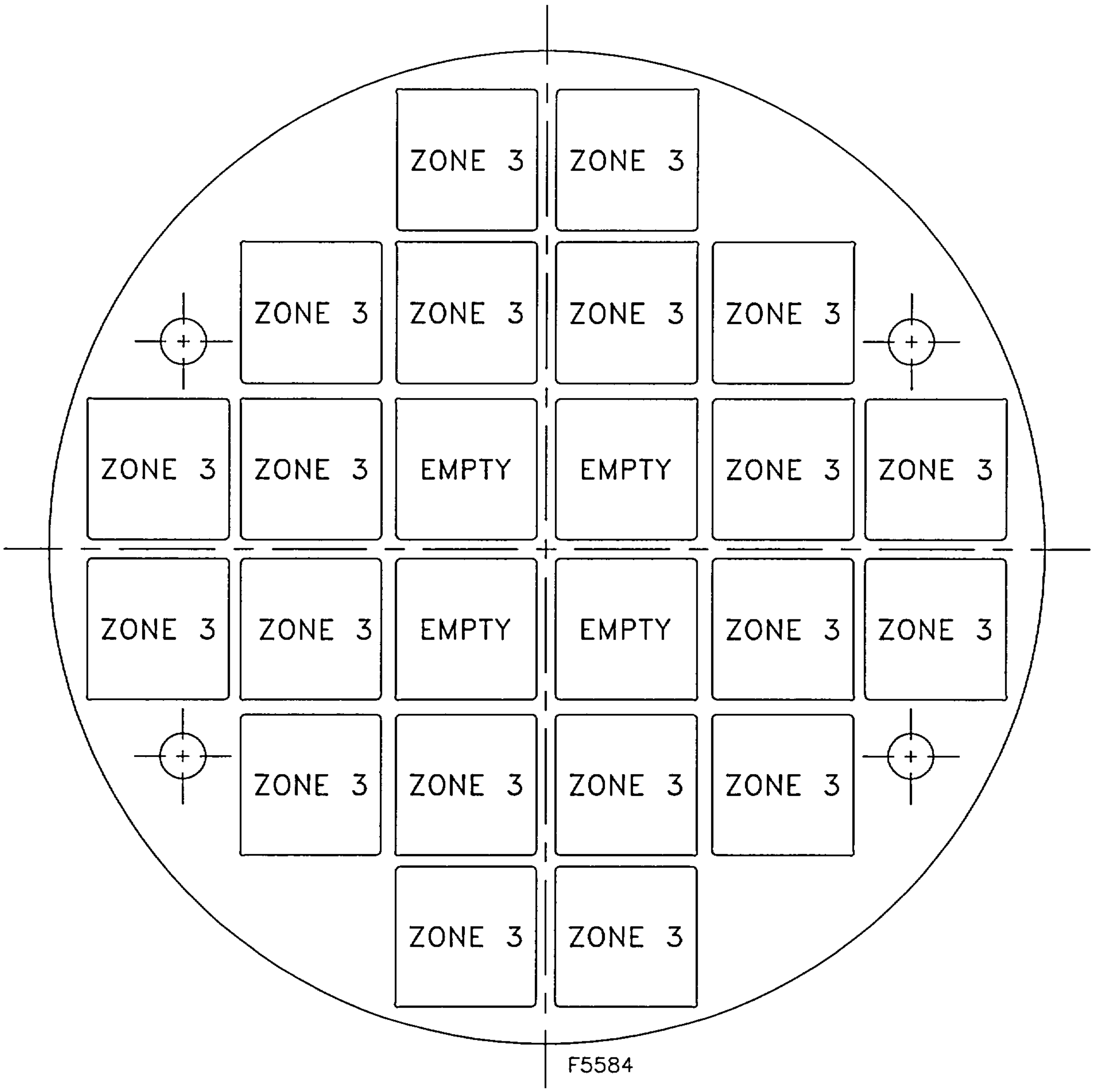


	Zone 1	Zone 2	Zone 3
Maximum Decay Heat (kW / FA)	NA	NA	1.3[(1)]
Maximum Decay Heat per Zone (kW)	NA	NA	24.0

1 Actual decay heat for each assembly must be determined to assure that the maximum decay heat for the canister is not exceeded

Figure N.2-2
Heat Load Zoning Configuration 2

N.5 Shielding Evaluation

The shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) for the 24P and 52B canisters is discussed in Sections 3.3.5, 7.0 and 8.0. The following evaluation specifically addresses the bounding dose rates due to design basis B&W 15x15 PWR fuel and Burnable Poison Rod Assemblies (BPRAs) loaded in a NUHOMS®-24PHB DSC. The fuel assemblies and BPRA characteristics are described in Section N.2-1. The shielding analysis is performed for the two DSC configurations (24PHBS and 24PHBL) of the NUHOMS®-24PHB System described in Section N.2.1. The basket layout for these two DSC configurations is identical except for the shield plug design and length of the DSC components. For shielding purposes, the 24PHBL DSC bounds the 24PHBS DSC because of the additional gamma source due to the BPRAs. Therefore, the dose rates calculated for the 24PHBL DSC with fuel plus BPRAs bound the dose rates for the 24PHBS DSC with fuel.

The design basis PWR fuel source terms are derived from the bounding fuel, B&W 15x15 Mark B 10 assembly design as described in Section N.5.2. The information provided in the Table N.5-1 is based on B&W 15x15 fuel. The types of spent fuel considered in this Appendix include the following:

- B&W 15x15 Mark B2, B3, B4, B4Z, B5, B5Z, B6, B7, B8, B9 and B10 fuel assemblies.
- B&W 15x15 reconstituted fuel assemblies with a maximum of 10 stainless steel rods or unlimited number of zircaloy clad lower enriched UO_2 rods instead of zircaloy clad UO_2 rods. (Note that lower enriched UO_2 rods are of similar design and behavior as the standard fuel rod aside from the uranium enrichment.) The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four. The reconstituted assemblies can be placed anywhere in the basket.
- Standard BPRA design for the B&W 15x15 class assemblies listed in Appendix J.

Note that while the B&W fuel types are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Table N.5-1 bound the specific manufacturers replacement fuel.

The design basis fuel source terms for this evaluation bound the source terms from fuel with the burnup/initial enrichment/cooling time combination given in Table N.2-3 through Table N.2-5 (with or without BPRAs and with or without reconstituted fuel assemblies) and located in the basket as shown in Figure N.2-1 and Figure N.2-2. This evaluation bounds the maximum dose rate on the surface of the HSM (Model 102) and Standard, OS197 or OS197H Transfer Cask (TC). The source terms for the BPRAs are the same as given in Appendix J for the B&W 15x15 BPRAs. The approach used to assure that the neutron and gamma source spectrum and the source terms used, bound the fuel allowed per the fuel qualification tables is consistent with that used for the Standardized NUHOMS®-24P and -52B canister designs *as described in Section 7.3.2.*

The NUHOMS®-24PHB DSC may store PWR fuel assemblies arranged in one of two alternate Heat Load Zoning Configuration s with a maximum decay heat of 1.3 kW per assembly and a maximum heat load of 24 kW per DSC. The Heat Load Zoning Configuration s are shown in Figure N.2-1 and Figure N.2-2. The NUHOMS®-24PHB System dose rates are calculated for these two configurations.

To calculate neutron and gamma source terms for Heat Load Zoning Configurations 1 and 2, the following burnup, initial heavy metal loading, minimum initial enrichment and cooling time combinations are used in this analysis:

- Zone 1: 55 GWd/MTU burnup, 3.4 wt. % U-235, 29-year cooling time for neutron and gamma source terms,
- Zone 2: 46 GWd/MTU burnup, 3.2 wt. % U-235 and 8-year cooling time for gamma source terms and 55 GWd/MTU burnup, 3.4 wt. % U-235, and 13.5-year cooling time for neutron source terms, and
- Zone 3: 46 GWd/MTU burnup, 3.2 wt. % U-235, and 5.5-year cooling time for gamma source terms and 55 GWd/MTU burnup, 3.4 wt. % U-235 and 8-years cooling time for neutron source terms.

These neutron and gamma source terms bound the dose rates due to fuel burnup, minimum initial enrichment and minimum cooling times given in the fuel qualification tables with initial heavy metal loadings up to 0.49 MTU. Initial enrichment in this section corresponds to the assembly average enrichment.

The Heat Load Zoning Configuration 2 produces the maximum dose rates on the surfaces of the HSM and TCs. *Configuration 2 produces higher dose rates on and around the HSM and TC because of the much shorter required cooling times for a given burnup and initial enrichment. The shorter cooling time results in a much higher ^{60}Co source in the end fittings which more than makes up for the four assemblies removed from the center of the DSC on the dose rates at the ends of the HSM and TC. The side dose rates are higher because the source terms are again larger for the shorter cooling times, which again make up for the missing four assemblies at the center of the DSC which provide negligible contribution to the side dose rate. Table* N.5-21 and *Table* N.5-22 *show a comparison of the dose rates on and around the HSM and TC respectively for the two configurations without BPRAs to demonstrate this point. These dose rates were calculated using DORT with the models described in Section* N.5.3.

*For the design basis shielding analysis, t*he bounding *fuel* gamma and neutron source terms *for Zone 3 (46 GWd/MTU burnup, 3.2 wt. % U-235, and 5.5-year cooling time for gamma source terms and 55 GWd/MTU burnup, 3.4 wt. % U-235 and 8-years cooling time for neutron source terms)* are used in the *DORT and MCNP* shielding models *for Configuration 2* to conservatively calculate dose rates for the NUHOMS®-24PHB System *with and without BPRAs*. For Heat Load Zoning Configuration 2, all twenty fuel assemblies in the DSC are modeled with neutron and gamma source terms consistent with 1.3 kW heat load. Therefore, these source terms result in conservative dose rates because the shielding analysis is based on a 26 kW heat load per DSC compared to the 24 kW per DSC design basis limit. *(Note that when loading for Configuration 2,* *the actual decay heat for each assembly must be determined to assure that the maximum decay heat load of 24 kW for the canister is not exceeded.)*

The design basis source terms for the B&W 15x15 BPRAs with up to 2 cycles burnup and 5-year cooling are taken from Appendix J. The properties used to calculate the design basis source terms for the authorized BPRAs are reproduced in Table N.5-2.

The cooling times for the reconstituted fuel assemblies are determined such that the source terms of reconstituted fuel assemblies are bounded by the standard fuel assemblies.

The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

N.5.1 Discussion and Results

The maximum dose rates for the NUHOMS®-24PHB system loaded with design basis PWR fuel assemblies (24 spent fuel assemblies for Configuration 1 or 20 spent fuel assemblies for Configuration 2) with and without BPRAs are provided in Table N.5-3. Table N.5-4 provides maximum and surface average dose rates on the HSM loaded with the NUHOMS®-24PHB DSC. Table N.5-3 provides a summary of the dose rates on and around the TC for 24PHB DSC transfer operations. *The dose rates in these tables are for the bounding Configuration 2 with and without BPRAs.*

A discussion of the method used to determine the design basis fuel and BPRA source terms is included in Section N.5.2. The model specification and shielding material densities are given in Section N.5.3. The method used to determine the dose rates due to design basis fuel assemblies with BPRAs in the NUHOMS®-24PHB DSC design configurations is provided in Section N.5.4. Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The shielding evaluation is performed with the DORT [5.2] code with the CASK-81 cross section library [5.3] and the MCNP code [5.10]. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section N.5.5.

N.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Table N.2-3 through Table N.2-5 and the bounding design basis fuel source terms suitable for use in the shielding calculations. The thermal and radiological source terms for the BPRAs are taken from Appendix J.

The B&W 15x15 Mark B10 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading. The neutron flux during reactor operation is peaked in the in-core (active fuel) region of the fuel assembly and drops off rapidly outside the in-core region. Much of the fuel assembly hardware is outside of the in-core region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the in-core (active fuel) region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table N.5-5. The light elements that make up the various materials for the various fuel assembly materials are taken from reference [5.4] and are listed in Table N.5-6. The bounding source terms are generated using a heavy metal weight of 0.488 MTU per assembly. However, the design basis heavy metal weight is 0.49 MTU per assembly. The difference in initial heavy metal weight between the bounding source terms and the design basis fuel is accounted for by increased cooling times. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the in-core regions neutron flux (fluence) correction factors are applied to light element composition for each region. The neutron flux correction factors are given in Table N.5-7 [5.13].

The fuel qualification tables are generated based on the decay heat limits and dose rate limits for the various Heat Load Zoning Configuration s shown in Figure N.2-1 and Figure N.2-2. SAS2H is used to calculate the minimum required cooling time as a function of assembly initial enrichment and burnup for the entries in the various fuel qualification tables. The total decay heat includes the contribution from the fuel as well as the hardware in the entire assembly. The fuel qualification table also accounts for the 8 watts design basis decay heat from the BPRAs. Because the decay heat generally increases slightly with decreasing enrichment for a given burnup, it is conservative to assume that the required cooling time for a higher enrichment assembly is the same as that for a lower enrichment assembly with the same burnup.

The 1-D discrete ordinates code ANISN [5.5] *and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3]* is used to demonstrate that the bounding source terms used in the evaluation result in dose rates on the surface of the HSM and TC greater than the dose rates due to fuels with burnup, initial heavy metal, initial enrichment and cooling time combinations given in the fuel qualification tables. The *ANISN results* due to the bounding *Configuration 2* source term*s* on the HSM roof determines the *"target* dose rate*"* for the HSM. Similarly, *ANISN results* on the side of the *TC* using the bounding *Configuration 2* source term*s* provides the *"target* dose rate*" for the TC.* The ANISN models are essentially identical to the appropriate DORT models for the locations of interest. This approach, *described in detail in Section* N.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® 24P and 52B canister, *as described in Section 7.2.3.* The radiological source terms generated in the SAS2H/ORIGEN-2 using 44-group ENDF/B-V library which includes more accurate evaluation for ^{154}Eu and ^{155}Eu, are used in the ANISN evaluations to calculate the surface dose rates. Heat load Configuration 2 (Figure N.2-2) produced the bounding total surface dose rate for both the HSM and TC.

A sample SAS2H/ORIGEN-S input file for the 55 GWd/MTU, 3.5 wt. % U-235 case is provided in Section N.5.5.1. It is conservatively assumed that a reactor operated at the maximum power from the beginning to the end of each cycle to maximize actinide production rate.

The cobalt concentration used in the various exposure regions and the total for entire fuel assembly was selected to maximize the gamma source terms.

For reconstituted fuel with zircaloy clad lower enriched uranium oxide rods, the assembly average enrichment produces the same total assembly decay heat, neutron and gamma source, where the assembly average enrichment can be calculated by taking the total grams of U-235 in the reconstituted assembly, divided by the total grams of uranium in the assembly, had it been configured in its reconstituted form prior to irradiation. Therefore, the appropriate fuel qualification table can be used directly to determine the minimum required cooling time by using the calculated average initial enrichment and the assembly average burnup. The resulting cooling time assures that the heat load, clad temperature and dose rate limits are maintained.

For reconstituted fuel with up to 10 stainless steel rods, a series of SAS2H calculations were performed to evaluate the effect of the increased cobalt content from the stainless steel rods. For a given burnup/initial enrichment and cooling time, the total neutron and decay heat are reduced because of the reduced heavy metal in the assembly. Therefore the surface neutron dose rates are reduced and the total decay heat is bounded by fuel that has not been reconstituted. The effect on the gamma source term and resulting gamma dose rate is evaluated using the ANISN models that are used to develop the fuel qualification tables. Based on the results of the ANISN evaluation, reconstituted fuel with up to 10 stainless steel rods can be stored using the same minimum cooling times as those shown in the fuel qualification tables or 9.0 years which ever is greater *as the contribution from the gamma source terms is bounded by fuel that has not been reconstituted.* Therefore reconstituted fuel with up to 10 stainless steel rods shall not be stored with less than 9.0 years cooling *to assure that both the neutron and gamma dose rates on the surface of the HSM and TC remain bounded by the Configuration 2 design basis evaluation.*

Boron concentration, moderator temperature and density values are selected in the depletion model to maximize buildup of isotopic activities such as ^{244}Cm resulting in conservative neutron source terms.

N.5.2.1 Gamma Source

Four SAS2H/ORIGEN-S runs are required to determine gamma source terms for the four exposure regions of interest for each fuel assembly; the bottom, in-core, plenum and top regions. The only difference between the runs is in Block #10 "Light Elements" of the SAS2H input and the 81$$ card in the ORIGEN-S input. Each run includes the appropriate "Light Elements" for the region being evaluated and the 81$$ card is adjusted to have ORIGEN-S output the total gamma source for the in-core region and only the light element source for the plenum, top and bottom nozzle regions.

The design basis source terms for the authorized BPRA designs, taken from Appendix J, are listed in Table N.5-11. The SAS2H/ORIGEN-S gamma source is output in the CASK-81 energy group structure shown in Table N.5-8 [5.3]. Gamma source terms for the in-core region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. These results for the bounding neutron and gamma source terms for various zones are given in Table N.5-9 and Table N.5-10.

Gamma source terms used in the shielding models are calculated by multiplying the assembly sources by the number of assemblies in the zone of interest and dividing by the appropriate heat load configuration zone volume. The appropriate assembly region volumes for the Heat Load Zoning Configuration zones are listed in Table N.5-12.

The volumes of the Zones for Configuration 1 are calculated as follows. Zone 1 encompasses the center four assemblies as shown in Figure N.2-1. The equivalent cross sectional area of this four assembly region is calculated such that the cross sectional area of the four fuel assembly compartments is conserved. The cross section of a fuel assembly compartment is 8.9 inches square. The cross sectional area is therefore $4*(8.9\ inches)^2 = 316.84\ in^2$ or $2,044\ cm^2$. This forms an equivalent radius of 25.52 cm. The lengths of the various assembly regions are given in Table N.5-5.

The volumes of the assembly regions in Zone 1 are therefore the product of the cross sectional area of Zone 1 and the length of the assembly region.

Zone 2 encompasses the middle ring of twelve assemblies as shown in Figure N.2-1. The equivalent cross sectional area of this twelve assembly ring is calculated such that the cross sectional area of the twelve fuel assembly compartments is conserved. The cross sectional area of Zone 2 is therefore $12*(8.9\ inches)^2 = 950.52\ in^2$ or $6,132\ cm^2$. This forms an equivalent annular region with an inner radius of 25.52 cm and an outer radius of 51.02 cm. The volumes of the assembly regions in Zone 2 are therefore the product of the cross sectional area of Zone 2 and the length of the assembly region.

The radius of Zone 3 is calculated by conserving the total area occupied and enclosed by the 24 fuel assemblies, including the guide sleeves, and guide sleeve wrappers, in the loaded DSC. The distance to the outer edge of each outer cutout in the spacer disc is used to calculate the area of this region. The resultant cross sectional area is $2,523.64\ in^2$ or $16,281\ cm^2$. The radius of the equivalent cylinder is 71.99 cm. Therefore the cross sectional area attributed to Zone 3 is $\pi(71.99^2-51.02^2) = 8,104\ cm^2$. The volumes of the assembly regions in Zone 3 are therefore the product of the cross sectional area of Zone 3 and the length of the assembly region.

The volumes of the Zone 3 for Configuration 2 are simply the sum of the volumes of Zone 2 and 3 from Configuration 1.

The outer ring of assemblies (Zone 3 for both configurations) control the dose rates on the surfaces of the HSM and the TC. Therefore, one would expect that Configuration 2 would result in the controlling shielding configuration because it allows 20 of the "hottest" fuel assemblies, and thus the strongest neutron and gamma source terms, to be configured around the edges of the DSC. This is demonstrated by the DORT calculations performed in support of this application. For models of Configuration 2, the center region (radius=25.51 cm) is modeled as void. (Note the radius' given above are modeled exactly as stated in the DORT models, and are rounded to the nearest cm in the ANISN models.) This is slightly conservative, as credit for the shielding that is provided by the guide sleeves and spacer disc ligaments in the center four cells is not accounted for in the model.

Almost 100% of gamma spectrum from light elements is in 0.70 to 1.33 MeV which corresponds exactly to two the most prominent lines of ^{60}Co. As for fission products, the main contributors after six (6) years with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh, ^{147}Sm, and ^{142}Ce are the strongest emitters at energies greater then 2.0 MeV. The accuracy of gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate then those for actinides because the calculation of their inventory has less uncertainty [5.1].

Shortly after discharge the emission in higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times after 10 years [5.1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of ^{244}Cm is calculated. Per SCALE 4.4 [5.1], reported experimental ^{244}Cm densities are accurate within ± 20%. The gamma emission intensity from Cm, which is proportional to the quantity of Cm in the actinide inventory, is bounded by this value. Uncertainty in the source strength in the gamma energy range 0.5 to 2.5 MeV is in the vicinity of 10 to 15 % [5.1].

N.5.2.2 Neutron Source Term

One SAS2H/ORIGEN-S run is required for each bounding source to determine the total neutron source terms for the in-core regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years . The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100 relative to ^{244}Cm. Thus the neutron spectrum for cooling times of interest is also totally dominated by ^{244}Cm in both spontaneous fission and (α,n) (~ 20% of total neutron source) components. The results for each burnup/initial enrichment/cooling time combination of interest are summarized in Table N.5-13.

Neutron source terms for use in the shielding models are calculated by multiplying the assembly sources by the number of assemblies in the in-core region of interest and dividing by the appropriate in-core heat load configuration zone volume. The appropriate assembly region volumes for heat load configuration zones are listed in Table N.5-12.

N.5.2.3 Axial Peaking

Axial peaking factors for both neutron and gamma sources in PWR fuel are taken from Reference [5.6]. These peaking factors are derived from work performed by the Department of Energy in support of its Topical Report for burnup credit [5.7]. The gamma peaking factors are shown as a function of the in-core region height in Table N.5-14. These factors are directly applied to each DORT interval in the fuel region. Neutron peaking factors in each zone are equal to the gamma factor raised to the fourth power to correctly account for the variation of neutron source with burnup. The burn-up profile from Reference [5.6] is intended to be conservative for burn-up credit evaluations for a fuel assembly of similar heavy metal loading, neutron spectrum, and total length. The fuel depletion (burn-up) is conservatively overestimated at the center of the fuel region. This conservatism is also applicable for shielding, where it is intended to conservatively overestimate the source at the middle of the fuel (or TC) where the shielding is the lowest. Thus, for the shielding evaluations, the burn up profile is also conservative.

N.5.2.4 Response Functions for Alternate Nuclear Parameters

To determine if fuel with a given burnup, *wt. %* enrichment and cooling time is bounded by the design basis shielding analysis, the total source term, *which includes the contribution from the fuel as well as the hardware in the entire assembly* in question *(including end fittings and plenum) is used to calculate its total dose rate and compared to the target dose rates* on the HSM roof and TC radial surface using a response function developed using the ANISN code. *This response function is only used to determine the relative strength of the various source terms from fuel assemblies to assure that the dose rates calculated on and around the HSM and TC, with DORT, using the design basis fuel source terms remain bounding.*

The BPRA contribution is fixed and is included in the design basis shielding evaluation as such and therefore is not included in the ANISN Response Function.

ANISN [5.5] determines the fluence of particles throughout one-dimensional geometric systems by solving the Boltzmann transport equation using the method of discrete ordinates. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order.

The ANISN code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration is resolved.

ANISN coupled with the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] and the ANSI/ANS-6.1.1-1977[5.11] flux-to-dose conversion factors is chosen to generate the response functions used to determine the relative strength of the various source terms from fuel assemblies to assure that the dose rates calculated on and around the

HSM and TC, with DORT, using the design basis fuel source terms remain bounding. ANISN provides an efficient method to calculate the response function.

The response functions are calculated using ANISN models to perform the evaluation for the fuel assembly parameters in the fuel qualification table. The ANISN model used to generate the HSM Response Function is a cut through the center of the DORT HSM roof model used for the shielding evaluation (for Configuration 2). The ANISN model used to generate the TC Response Function is a cut through the center of the DORT TC side model used for the shielding evaluation (for Configuration 2). Figure N.5-19 *and Figure* N.5-20 *provide sketches for the ANISN models of the HSM roof and TC centerline respectively.*

The material densities used in the ANISN models for the various model regions are listed in Table N.5-25. *These material densities are very similar to those used for the DORT and MCNP analysis, but are simplified to reduce the size of the ANISN input decks. Only the important elements of a give material are included and the gram density of the material is maintained.*

To generate the neutron, including (n,γ), response function, ANISN runs for the HSM roof and TC are run with a starting neutron source of one neutron per second per assembly with a ^{244}Cm spectrum. The resulting calculated total dose rates on the HSM and TC surfaces are the appropriate neutron and (n,γ) response functions. To generate the response function for each gamma group (CASK-81 group structure), ANISN runs are performed for the HSM and TC assuming one gamma per second per assembly in that group. The resulting ANISN calculated total dose rates on the HSM and TC surfaces are the appropriate gamma response functions. An example ANISN input file is included in Section N.5.5.4. *The HSM and Transfer Cask materials are very similar in all directions; the ANISN models accurately assess the relative source strengths to assure that all dose rates, as summarized in Section* N.5.4 *remain bounded by Configuration 2 and the Configuration 2 design basis source terms. While the ANISN models are based on Configuration 2, this does not affect the conclusions of the evaluation of the various burnup/initial enrichment/cooling time combinations.*

To *determine if the source term from a candidate assembly for a given burnup, wt.% enrichment and cooling time,* multiply the total neutron source in n/sec/assembly by the neutron response function given in Table N.5-15 *and* the group-wise source in γ/sec/assembly per group times the appropriate gamma group response function *given in* Table N.5-15 and sum the results, *thus accounting for the total, i.e. the neutron, (n,γ) and primary gamma contributions from the fuel assembly.* If the total dose rate is less than or equal to that determined *in the same way* for the design basis source term, then the minimum cooling time is adequate *for shielding purposes. Note that the decay heat limit must also be verified depending on fuel Zone.* If not, the cooling time is increased until the *target* dose rate *is* met for both the HSM and TC. *It should be noted that for Zones 1 and 2, decay heat is the limiting factor and for Zone 3, decay heat and/or shielding are the limits.*

The *target dose rate* calculated *with design basis neutron and gamma source terms*, using the response function, is 93.7 mrem/hr on the HSM roof surface and 1370.2 mrem/hr on the TC side surface. *The corresponding DORT calculated dose rates are 36 mrem/hr on the roof surface and 1026 mrem/hr on the TC cask surface. The ANISN calculated target dose rates are higher than those calculated by DORT at the corresponding location, due to the simplifying assumptions* *used in the ANISN models for the source and geometry. Calculation of these target dose rates is shown in Table* N.5-23. *Table* N.5-23 *lists the response function for the HSM and the TC, the total design basis source term for a single assembly and the corresponding target dose rates derived by multiplying the applicable response function by the source term and summing the results.*

To evaluate other burnup/initial enrichment/cooling time combinations, one obtains the total neutron and group-wise gamma source for the applicable burnup/initial enrichment/cooling time combination for a single assembly, which must include the contribution from the fuel as well as the hardware in the entire assembly. An example calculation is presented in Table N.5-24 *for the 55 GWd/MTU, 3.4 wt. % U-235, 8-year cooled fuel case shown in Table N.2-5. The combination of burnup, initial enrichment and cooling time is acceptable because the total decay heat is less than 1.3 kwatts and the total dose rates are less than 93.7 mrem/hr for the HSM and the 1370.2 mrem/hr for the TC.*

The response function is used to account for the substantial shift in the gamma spectrum over the range of burnup/initial enrichment/cooling time combinations included in the Fuel Qualification Tables provided in Chapter N.2. The important energy groups contributing to the total dose rate on and around the HSM and TC are groups 35 to 29 (0.6 – 2.5 Mev) as demonstrated in Table N.5-23 *and Table* N.5-24. *However depending upon cooling time most notably, the lower energy groups 38 to 40 dominate the total gamma source (gamma/sec) but make no contribution to the dose rate outside the HSM and TC. The response function is used to remove these low energy gammas from the evaluation. Table* N.5-26 *shows the fraction of the total number of primary gammas and corresponding contribution to the HSM and TC surface dose rate in groups 35 to 29 and 38 to 40 for the design basis source terms and for 55 GWd/MTU, 3.4 wt. % U-235, 8-year cooled fuel.*

One 3-D MCNP model is used to calculate dose rates in front and roof bird screens. The model is shown on Figure N.5-4A and Figure N.5-4B. Square labels designate MCNP geometry cells. Circular labels on the figures correspond to surfaces bounding the cells. Bounding surfaces are listed in Table N.5-19. The surface designators in the table obey conventions used in MCNP surface cards description, except for cylindrical ones. Only radii are shown in Table N.5-19. All cylinders in the MCNP model are parallel to x-axis, their axis pass through y=155.00 cm and z=31.84 cm.

Cells 21 and 28 represent 4 inner and 20 outer fuel assemblies respectively. Cell 17 is a stainless steel DSC structural shell. Bottom and Top Shield Plugs with associated inner and outer cover plates are designated as cells 19 and 18 respectively. These cells are filled with carbon steel.

The shielding analysis results predicted by the 2-D DORT methodology, described above, bounds the 3-D MCNP analysis result for the NUHOMS® System as documented in reference [5.14]. The 3-D MCNP methodology is also validated by comparison to actual measured dose rate data from installed NUHOMS® systems. The results show that the conservative assumptions used in the 2-D DORT analysis bound the 3-D results. Therefore, the results of the comparison also show that the 2-D DORT results are conservative as compared to measured data.

4. The DORT and MCNP results are used to calculate offsite exposures.

5. DORT models are also generated to determine the effects of accident scenarios including HSM sliding and loss of cask neutron shield *for the bounding Configuration 2.*

N.5.4.6 Assumptions

The following general assumptions are used in the analyses.

N.5.4.6.1 Source Terms

The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein. Surface gamma dose rates are calculated for the HSM and cask surfaces using the actual photon spectrum applicable for each case.

The PWR heavy metal weight is assumed to be 0.49 MTU per assembly to bound existing PWR fuel designs.

N.5.4.6.2 Shielding Materials

Source regions are homogenized (smeared) but cross sectional areas are preserved to simplify the shielding calculations.

The HSM reinforcing bars (rebars) have been included in smeared regions in the HSM walls and roof. The rebar steel is included in four inch thick regions for each face of each HSM surface.

This layered method of including the rebar in the shielding model is consistent with ANSI/ANS 6.4 guidelines.

N.5.4.6.3 HSM DORT 2-D Models

The DSC and fuel assemblies are positioned as close to the front door as possible to maximize the front wall dose rates.

Dose rates are calculated on HSM front, rear shield wall, roof surfaces, roof and front birdscreens. Reported dose rate distributions are along the lines of the intersection between these surfaces and vertical plane through DSC and HSM door axis. DORT X-Z model is used to calculate dose rates on end module shield wall. It assumes that air vents are infinite in horizontal direction which gives conservative dose rate value. In addition, because the fuel is modeled in 'X-Z' 2-D geometry, the surface area of the source is increased and the source points are pushed closer to the outside of the canister resulting in under prediction in the self shielding by the canister internals and the fuel itself which is conservative.

Fully symmetric S16 quadrature is used for all cylindrical models. Upward and downward biased, 420 and 400 direction quadrature sets [5.9] are used when calculating dose rates on the front and rear surfaces of HSM.

Embedments in the HSM concrete are neglected.

N.5.4.6.4 Cask DORT Models

The cask and DSC are modeled in cylindrical coordinates.

Three inches of supplemental neutron shielding and one inch of steel are placed on top of the DSC cover plates during welding.

During the accident case *(Configuration 2)*, the neutron shielding material in the cask neutron shield is assumed lost.

N.5.4.7 HSM Dose Rates

Dose rates on and around an HSM containing a design basis NUHOMS®-24PHB DSC are calculated using the DORT 2-D discrete-ordinates code. HSM surface dose rates are calculated using the three models which are designated as: "roof models," "floor models" and "X-Z models." The "roof" model calculates doses rates on the roof and on the top half of the front and back of the HSM above the horizontal plane through DSC and front door axis. The "floor" model calculates dose rates on the bottom half of front and back of the HSM below the horizontal plane just mentioned. "X-Z" model is for an estimate of dose rates along the line on surface of end module shield wall, running in vertical elevation through the middle of that wall.

Roof and floor models are shown in Figure N.5-1 and Figure N.5-2. As can be seen from the figures, halves of roof and floor models connected along line OO' represent the HSM. "X-Z" model is depicted on Figure N.5-3.

In the MCNP model, shown in Figure N.5-4A and Figure N.5-4B, equal size rectangular cells are placed in front of front bird screen and roof bird screen in the HSM models. These cells are not part of the geometry and used only to calculate the dose rate distribution functions in front of the bird screens. The thickness of the cells is 2.0 cm and 0.2 cm, respectively. Dose rates calculated in these volumes provide a representative dose rate distribution near bird screens.

BPRA contributions to the HSM surface dose rates have been taken from Appendix J. This evaluation has been performed only for Heat Load Zoning Configuration 2, as this option produces the bounding system dose rates.

The HSM results reported for fuel without BPRAs are scaled by appropriate BPRA scaling factors to estimate the dose rates for fuel with BPRAs. Scaling factors are calculated by adding the BPRA contribution at *the* locations *addressed in Appendix J* to the corresponding fuel contribution *without BPRA* at that location and then dividing by the fuel contribution *without BPRA. The scaling factors for the HSM back shield wall, front, roof, side, front bird screen, and roof bird screen are 1.107, 1.005, 1.093, 1.035, 1.035 and 1.035 respectively.*

N.5.4.8 Data Reduction and HSM Dose Rate Results

The dose rate distribution for each case is calculated by summing the neutron and gamma DORT results. Surface average dose rates for each HSM surface are calculated as discussed below.

The average dose rates from the DORT results are calculated using the following formula:

$$\bar{D} = \frac{\iint\limits_{Surface} D(x,y)dxdy}{\iint\limits_{Surface} dxdy} \qquad (5.2)$$

Or, expressed in the finite mesh approximation, this becomes:

$$\bar{D} = \frac{\sum_{i=1}^{IM-1}\sum_{j=1}^{JM-1} D(x_i,y_j)\cdot(x_{i+1}-x_i)\cdot(y_{j+1}-y_j)}{\sum_{i=1}^{IM-1}\sum_{j=1}^{JM-1}(x_{i+1}-x_i)\cdot(y_{j+1}-y_j)} \qquad (5.3)$$

where,

JM = number of fine meshes along axis (Z)

IM = number of fine meshes along axis (X)

$(x_i\ y_j)$ = a point in the center of cell (i, j)

$D(x_i,y_j)$ = dose rate at the middle of cell (i,j)

i, j = fine mesh indices.

Assuming that the dose rate distribution on the surfaces has only one-dimensional profile, Equation (3.2) for averaged values on those surfaces becomes:

N.5.4.8.4 HSM End Shield Wall Surface Dose Rates

The HSM end shield wall dose rates are calculated using the DORT X-Z model. The average dose calculation is performed in the same manner as was used on the HSM roof. The results are summarized in Table N.5-4.

N.5.4.8.5 HSM Front and Roof Bird Screen Surface Dose Rates

Dose rate values calculated with MCNP code models are provided in Table N.5-3. Gamma dose rate distribution at the HSM roof bird screen and front bird screen are shown in Figure N.5-7 and Figure N.5-8.

N.5.4.8.6 HSM Dose Rates for Fuel with BPRAs

The resulting dose rates on the HSM surfaces are given in Table N.5-3 and Table N.5-4 for fuel with BPRAs. Similarly, the dose rates including BPRAs adjacent to the HSM's birdscreens are shown in Table N.5-3.

N.5.4.9 TC Dose Rates

The NUHOMS® TC containing a NUHOMS®-24PHB DSC is modeled in cylindrical coordinates using material zones as shown in Figure N.5-5. The dose rates for a OS197 TC with a liquid neutron shield (shown in Figure N.5-5) bound the dose rates for the Standard TC with a solid neutron shield. The materials used in these zones are varied to model the various welding and decontamination cases during fuel loading, canister sealing and transfer operations.

The onsite transfer case includes all cask and DSC covers, air in the DSC cavity (air versus helium has no effect on the results), air in the cask/DSC annulus, and water in the neutron shield cavity. The decontamination model is similar except it includes water in the DSC and in the cask/DSC annulus; the cask cover and both top DSC covers are removed. The DSC inner top cover plate welding (Wet Welding) model is similar to the decontamination model except that the water inside the DSC is assumed to be lowered four inches below the top shield plug, the inner top cover plate is added and supplemental shielding consisting of three inches of NS-3 and one inch of steel is added. The DSC outer top cover plate welding case (Dry Welding) model is similar to the wet welding case except the water is removed from the DSC cavity and the outer top cover plate is added to the inner top cover plate and the supplemental shielding described above. The accident condition model *(Configuration 2)* is identical to the onsite transfer model except that the neutron shield is removed.

The results of the evaluation, with and without BPRAs, are provided in Table N.5-3 and shown graphically in Figure N.5-10 to Figure N.5-18 for the various cases and at various distances from the TC.

```
       92235.50 1.936e-4
       92238.50 4.839e-3
phys:p 15.0 j j
cut:p j j 0.50 0.25 j
```

> **cut** sets the weight cutoffs. If the weight of particle falls below ***0.25x(source sell importance/current cell importance)*** then there is a ***weight/(0.50x(current cell importance/source sell importance))*** probability that particle will survive. The weight of survived particle will be ***0.50x(source sell importance/current cell importance)***.

```
c     prdmp j j 1 4
print 110 120
nps 39114946
ctme  720
```

N.5.5.4 Sample ANISN Model (Neutron Response Function for HSM)

```
PWR HSM Roof Design Basis Source(Outer 1.3 kWt)
'  Group-by-group response function
'  Neutron 1 n/s/assembly
15$$
'      ID        ITH      ISCT       ISN       IGE
       32        0        3          8         2
'      IBL       IBR      IZM        IM        IEVT
       1         0        10         154       0
'      IGM       IHT      IHS        IHM       MS
       40        3        4          43        92
'      MCR       MTP      MT         IDFM      IPVT
       48        0        80         0         0
'      IQM       IPM      IPP        IIM       ID1
       1         0        0          40        0
'      ID2       ID3      ID4        ICM       IDAT1
       0         3        1          50        0
'      IDAT2     IFG      IFLU       IFN       IPRT
       0         0        0          1         1
'      IXTR
       0
16**
'      EV        EVM      EPS        BF        DY
       0.0       0.0      0.0001     1.420892  361.42
'      DZ        DFM1     XNF        PV        RYF
       0.0       0.0      0.0        0.0       0.5000
'      XLAL      EQL      XNPM       T
       0.0002    F0.0                T
14*
```

Cross Sections not listed for brevity.

```
  T
17**       26R0.0    54R4.904E-10    74R0.0
           26R0.0    54R4.169E-09    74R0.0
           26R0.0    54R1.147E-08    74R0.0
           26R0.0    54R5.717E-08    74R0.0
           26R0.0    54R1.448E-07    74R0.0
           26R0.0    54R1.915E-07    74R0.0
           26R0.0    54R4.806E-07    74R0.0
           26R0.0    54R3.935E-07    74R0.0
           26R0.0    54R9.616E-08    74R0.0
           26R0.0    54R4.966E-07    74R0.0
```

```
          26R0.0   54R8.850E-07    74R0.0
          26R0.0   54R7.846E-07    74R0.0
          26R0.0   54R3.615E-07    74R0.0
          26R0.0   54R1.557E-11    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          7Q154
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
          26R0.0   54R0.000E+00    74R0.0
        T
'       Zone-1 Zone-2 Zone-3
3**     26R0.0 24R1.0 30R1.0 74R0.0    39Q154
        T
1**     F0.0
4**     25I0.0  23I26  29I50  3I72 84.0 84.38 84.5 85.33 7I87 95.25
        9I95.65 45I106.68 198.12 200
5**     F1.0
6**   .00 .0604938  .0453704  .0453704  .0604938 .0604938 .0453704
      .0453704   .0604938  0.0  .0453704   .0462962  .0453704
      .0453704   .0462962  .0453704  0.0  .0453704  .0453704
      .0453704   .0453704  0.0  .0604938  .0604938
7**  -.9759000  -.9511897  -.7867958  -.5773503  -.2182179  .2182179
     .5773503  .7867958  .9511897  -.8164965  -.7867958  -.5773503
     -.2182179  .2182179  .5773503  .7867958  -.6172134  -.5773503
     -.2182179   .2182179  .5773503   -.3086067  -.2182179  .2182179
8$$     26R1  24R2  30R3  4R4  3R5  9R6  1R7 10R8 46R9 1R10
9$$       57    49    49   57   53   57   53   57   65   57
' MAT   1, 4      = H
' MAT   5, 8      = C
' MAT   9, 12     = O
' MAT  13, 16     = AL
' MAT  17, 20     = SI
' MAT  21, 24     = CA
' MAT  25, 28     = FE
' MAT  29, 32     = ZR
' MAT  33, 36     = PB
' MAT  37, 40     = U235
' MAT  41, 44     = U238
' MAT  45, 48     = FX
'
' MIXTURE  49, 52      = FUEL-RADIAL
' MIXTURE  53, 56      = STEEL
```

```
' MIXTURE  57, 60       = AIR
' MIXTURE  61, 64       = LEAD
' MIXTURE  65, 68       = CONCRETE
' MIXTURE  69, 72       = BOT NOZZLE
' MIXTURE  73, 76       = TOP NOZZLE
' MIXTURE  77, 80       = FUEL AXIAL
'
10$$
' FUEL-RADIAL
      49  50  51  52
      49  50  51  52
      49  50  51  52
      49  50  51  52
      49  50  51  52
' STEEL-
      53  54  55  56
' AIR-
      57  58  59  60
' LEAD-
      61  62  63  64
' CONCRETE
      65  66  67  68
      65  66  67  68
      65  66  67  68
      65  66  67  68
      65  66  67  68
      65  66  67  68
' BOT-NOZZLE
      69  70  71  72
      69  70  71  72
' TOP-NOZZLE
      73  74  75  76
      73  74  75  76
' FUEL-AXIAL
      77  78  79  80
      77  78  79  80
      77  78  79  80
      77  78  79  80
      77  78  79  80
11$$
' FUEL-RADIAL
       9  10  11  12
      25  26  27  28
      29  30  31  32
      37  38  39  40
      41  42  43  44
' STEEL-
      25  26  27  28
' AIR-
     9   10   11  12
' LEAD-
      33  34  35  36
' CONCRETE
        1   2   3   4
        9  10  11  12
       13  14  15  16
       17  18  19  20
```

```
      21   22   23   24
      25   26   27   28
' BOT-NOZZLE
      25   26   27   28
      29   30   31   32
' TOP-NOZZLE
      25   26   27   28
      29   30   31   32
' FUEL-AXIAL
     9    10   11   12
     25   26   27   28
     29   30   31   32
     37   38   39   40
     41   42   43   44
12**
' FUEL-RADIAL
         4R7.084E-3
         4R1.280E-3
         4R2.233E-3
         4R1.434E-4
         4R3.399E-3
' STEEL
         4R8.487E-2
' AIR
         4R5.28E-6
' LEAD
         4R3.296E-2
' CONCRETE
         4R7.770E-3
         4R4.386E-2
         4R2.389E-3
         4R1.581E-2
         4R2.916E-3
         4R3.128E-4
' BOT-NOZZLE
         4R9.781E-3
         4R7.029E-4
' TOP-NOZZLE
         4R4.031E-3
         4R8.469E-4
' FUEL-AXIAL
         4R7.084E-3
         4R2.017E-3
         4R2.233E-3
         4R1.434E-4
         4R3.399E-3
19$$     F3
22$$     F-45
23$$     7 8 9
         T         T
```

5.13 "Characteristics of Potential Repository Waste," DOE/RW-0184-R1, Volume 1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1992.

5.14 Letter from U. B. Chopra (TN) to Mary Jane Ross-Lee (SFPO), "Response to Request for Additional Information (RAI) and submittal of Revision 1 of Application for Amendment No. 5 to the NUHOMS® Certificate of Compliance No. 1004 (TAC No. L23343)," Enclosure 2 to NUH03-02-13 Response to Question 5-2, NUH03-02-13, February 21, 2002.

Table N.5-3
Summary of NUHOMS®-24PHB System Maximum Dose Rates

Dose Rate Location	*Configuration 2* Dose without BPRAs			*Configuration 2* Dose with BPRAs		
	Gamma (mrem/hr)	Neutron (mrem/hr)	Total[1] (mrem/hr)	Gamma (mrem/hr)	Neutron (mrem/hr)	Total[1] (mrem/hr)
HSM Roof	55	1	56	61	1	62
HSM Roof Birdscreen[4]	913	9[3]	N/A	945	9[3]	N/A
HSM End Shield Wall Surface	250	2	251	258	2	260
HSM Door Exterior Surface (centerline)	9	4	12	10	4	13
HSM Front Birdscreen[4]	458	6[3]	N/A	474	6[3]	N/A
HSM Back Shield Wall	4	0.1	4	5	0.1	5
Centerline Top DSC Cover Plate w/3"ns3+1" Steel Dry Welding	258	165	423	277	165	442
Outer Edge Centerline Top DSC (Peak Annulus)	5754	2.0	5755	6050	2.0	6052
Cask Surface (Radial) Contact Normal Condition	688	537	1148	738	537	1193
3 ft from Cask Surface (Radial) Normal Condition	252	192	442	272	192	462
Cask Surface (Radial) Contact Accident Condition	864	6132	6776	927	6132	7059
Cask Top Surface	253	254	354	266	254	365
Cask Bottom Surface[2]	101	1136	1193	101	1136	1193

Notes:

(1) Gamma and Neutron peaks do not always occur at same location therefore the total is not always the sum of the gamma plus neutron.

(2) The dose rates reported are the peak bottom surface dose rates excluding the grapple ring cutout area The peak bottom surface dose rate (703 mrem/hr gamma and 4937 mrem/hr neutron) is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 71 mrem/hr gamma, 227 mrem/hr neutron for a total average dose rate of 298 mrem/hr.

(3) Dose rates in front and roof bird screens are calculated with MCNP. An average and maximum values for gamma dose rates are calculated. Neutron dose rate is substantially lower than gamma component. Only an average neutron dose rate is calculated.

(4) *To assure the accuracy of the MCNP results, the MCNP calculated error is below 15%*

Table N.5-4
Summary of HSM Dose Rates with 24PHB DSC

Surface	Dose Rate Component	*Configuration 2* Dose Rate without BPRAs		*Configuration 2* Dose Rate with BPRAs	
		Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)	Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)
Rear(1)	Gamma	4	1.3	5	1.4
	Neutron	0.1	0.1	0.1	0.1
Front	Gamma	9	4.9	10	5.4
	Neutron	4	2.0	4	2.2
Roof	Gamma	55	25.4	615	27.7
	Neutron	1.0	0.5	1.0	0.5
Side(1)	Gamma	250	29.6	258	30.7
	Neutron	2.0	0.5	2.0	0.5
Front Bird Screen(3)	Gamma	458	261.3	474	270.6
	Neutron	Not calculated(2)	6.2	Not calculated(2)	6.2
Roof Bird Screen(3)	Gamma	913	408.9	945	423.3
	Neutron	Not calculated(2)	9.0	Not calculated(2)	9.0

(1) Includes 24 inch shield wall.

(2) See Table N 5-3, Note 3.

(3) To assure the accuracy of the MCNP results, the MCNP calculated error is below 15%

Table N.5-21
Configuration 2 / Configuration 1 Dose Rate Comparison - HSM

Surface	Dose Rate Component	Configuration 2 without BPRAs		Configuration 1 without BPRAs	
		Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)	Maximum Dose Rate (mrem/hr)	Surface Average Dose Rate (mrem/hr)
Rear(1)	Gamma	4	1.3	3.4	1.0
	Neutron	0.1	0.1	0 1	0.1
HSM Door Exterior Surface (Centerline)	Gamma	9	4.9	NA(2)	NA(2)
	Neutron	4	2.0	NA(2)	NA(2)
Roof	Gamma	55	25.4	42	18.4
	Neutron	1.0	0.5	0.9	0.5
Side(1)	Gamma	250	29.6	178	21.1
	Neutron	2.0	0.5	0.7	0.2

(1) Includes 24 inch shield wall

(2) Not calculated for Configuration 1. See Cask Bottom Surface Dose Rates reported in Table N.5-22, *for an equivalent comparison of Configuration 1 vs Configuration 2*

Table N.5-22
Configuration 2 / Configuration 1 Dose Rate Comparison - TC

Dose Rate Location	*Configuration 2 without BPRAs*		*Configuration 1 without BPRAs*	
	Gamma (mrem/hr)	*Neutron (mrem/hr)*	*Gamma (mrem/hr)*	*Neutron (mrem/hr)*
Cask Surface (Radial) Contact Normal Condition	*688*	*537*	*533*	*473*
3 ft from Cask Surface (Radial) Normal Condition	*252*	*192*	*187*	*170*
Cask Top Surface	*253*	*254*	*203*	*189*
Cask Bottom Surface(1)	*703*	*4937*	*463*	*3316*

Notes:
(1) The dose rates reported here are the peak bottom surface dose rates in the grapple ring cutout area.

Table N.5-23
"Response Function" Evaluation of Design Basis Source Terms
Configuration 2

Column A	Column B	Column C	Column D	Column E	Column F
Response Function Parameter	*HSM Response Function in mrem/hr per particle/sec per assembly*	*TC Response Function in mrem/hr per particle/sec per assembly*	*Zone 3 Design Basis Source Term particle/sec for single assembly*	*Column B* Column D HSM*	*Column C* Column D TC*
Neutron (1)	2.4748E-09	5.1990E-07	9.650E+08	2.4	501.7
Group 23(2)	2.4231E-10	9.6836E-11	3.370E+05	0.0	0.0
Group 24	1.7971E-10	1.2279E-10	1.590E+06	0.0	0.0
Group 25	1.1252E-10	1.3341E-10	8 090E+06	0.0	0.0
Group 26	6 0422E-11	1.2871E-10	2.020E+07	0.0	0.0
Group 27	2.8447E-11	1.0855E-10	9.450E+09	0 3	1.0
Group 28	1.1241E-11	7.5352E-11	7.530E+10	0 8	5.7
Group 29	4 7289E-12	4.4980E-11	1.900E+12	9.0	85.5
Group 30	1.6474E-12	2.0229E-11	9.930E+11	1.6	20.1
Group 31	5.4548E-13	7.1290E-12	6.460E+13	35.2	460.5
Group 32	1.2186E-13	1.1681E-12	2.290E+14	27.9	267.5
Group 33	1.9157E-14	6.6010E-14	3.590E+14	6 9	23.7
Group 34	3.4130E-15	1.4615E-15	2.730E+15	9.3	4.0
Group 35	3.2007E-16	6.2364E-16	8.070E+14	0.3	0.5
Group 36	9.8537E-18	6.9437E-18	6.470E+13	0.0	0.0
Group 37	5.9591E-19	2.8411E-21	9.470E+13	0.0	0.0
Group 38	4 6895E-21	2.1118E-29	3.320E+14	0.0	0.0
Group 39	1.9963E-31	6 9233E-34	4.170E+14	0.0	0.0
Group 40	7.0065E-45	5.6052E-45	2.080E+15	0.0	0.0

(1) Also accounts for (n,γ) contribution

(2) Group Structure for CASK-81 Library [5.3] (See Table N.5-8 *for group structure).*

Total Dose Rate, mrem/hr (sum of column) 93.7 1370.3

Maximum decay heat per assembly: Zone 1 = 0.7 kW
Zone 2 = 1.0 kW
Zone 3 = 1.3 kW

Table N.5-24
"Response Function" Evaluation of Sample Source Terms
55 GWd/MTU, 3.4 wt. % U-235, 8-year Cooled Fuel Case Configuration 2

Column A	Column B	Column C	Column D	Column E	Column F
Response Function Parameter	HSM Response Function in mrem/hr per particle/sec per assembly	TC Response Function in mrem/hr per particle/sec per assembly	55 GWd/MTU, 3.4 wt. % U-235 Enrichment, 8 Year Cooling Time Fuel Source Term particle/sec for single assembly	Column B* Column D HSM	Column C* Column D TC
Neutron (1)	2.4748E-09	5.1990E-07	9.680E+08	2.4	503.3
Group 23(2)	2.4231E-10	9.6836E-11	5.574E+05	0.0	0.0
Group 24	1.7971E-10	1.2279E-10	2.625E+06	0.0	0.0
Group 25	1.1252E-10	1.3341E-10	1.338E+07	0 0	0.0
Group 26	6 0422E-11	1.2871E-10	3.334E+07	0.0	0.0
Group 27	2.8447E-11	1.0855E-10	2.266E+09	0.1	0.2
Group 28	1.1241E-11	7.5352E-11	1.762E+10	0.2	1.3
Group 29	4.7289E-12	4.4980E-11	3.068E+11	1.5	13.8
Group 30	1.6474E-12	2.0229E-11	3.097E+11	0.5	6 3
Group 31	5.4548E-13	7.1290E-12	6.026E+13	32.9	429.6
Group 32	1.2186E-13	1.1681E-12	2.371E+14	28.9	277.0
Group 33	1.9157E-14	6.6010E-14	2.263E+14	4.3	14.9
Group 34	3.4130E-15	1.4615E-15	2.640E+15	9.0	3.9
Group 35	3.2007E-16	6.2364E-16	4.609E+14	0.1	0.3
Group 36	9.8537E-18	6.9437E-18	4.942E+13	0.0	0.0
Group 37	5.9591E-19	2.8411E-21	7.679E+13	0.0	0.0
Group 38	4 6895E-21	2.1118E-29	2.669E+14	0 0	0 0
Group 39	1.9963E-31	6.9233E-34	3.513E+14	0.0	0.0
Group 40	7.0065E-45	5.6052E-45	1.824E+15	0.0	0.0
			Total Dose Rate, mrem/hr (sum of column)	77.5	747.3
				<93.7	<1370.2

(1) *Also accounts for (n,γ) contribution.*

(2) *Group Structure for CASK-81 Library* [5.3] *(See* Table N 5-8 *for group structure).*

Decay Heat *1.29* kW, *<1.30 kW, therefore only allowed in Zone 3.*

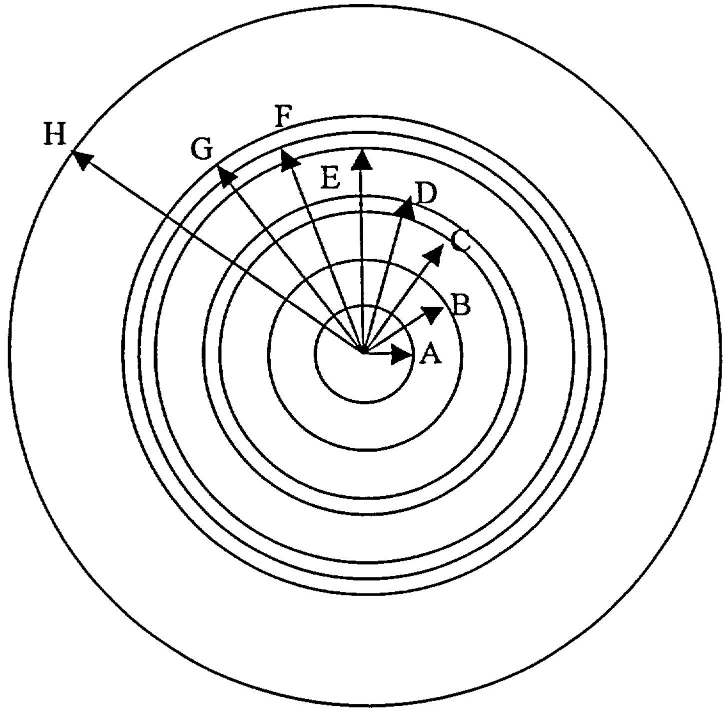
Table N.5-25
ANISN Material Densities

Element	*Material Atom Densities (atoms/barn-cm)*					
	Stainless Steel	*Air*	*Lead*	*Fuel*	*Water*	*Concrete*
H	*0.000E+00*	*0.000E+00*	*0.000E+00*	*0.000E+00*	*6.393E-02*	*7.770E-03*
C	*0.000E+00*	*0.000E+00*	*0.000E+00*	*0.000E+00*	*0 000E+00*	*0 000E+00*
O	*0.000E+00*	*5.280E-06*	*0.000E+00*	*7.084E-03*	*3.203E-02*	*4.386E-02*
Al	*0.000E+00*	*0.000E+00*	*0 000E+00*	*0.000E+00*	*0.000E+00*	*2.389E-03*
Si	*0.000E+00*	*0.000E+00*	*0.000E+00*	*0 000E+00*	*0.000E+00*	*1.581E-02*
Ca	*0 000E+00*	*0 000E+00*	*0.000E+00*	*0.000E+00*	*0 000E+00*	*2.916E-03*
Fe	*8.487E-02*	*0.000E+00*	*0.000E+00*	*2.017E-03*	*0 000E+00*	*3.128E-04*
Zr	*0.000E+00*	*0.000E+00*	*0.000E+00*	*2.233E-03*	*0.000E+00*	*0.000E+00*
Pb	*0.000E+00*	*0.000E+00*	*3.296E-02*	*0.000E+00*	*0.000E+00*	*0.000E+00*
U235	*0 000E+00*	*0.000E+00*	*0.000E+00*	*1.434E-04*	*0 000E+00*	*0.000E+00*
U238	*0 000E+00*	*0.000E+00*	*0.000E+00*	*3.399E-3*	*0.000E+00*	*0.000E+00*

Table N.5-26

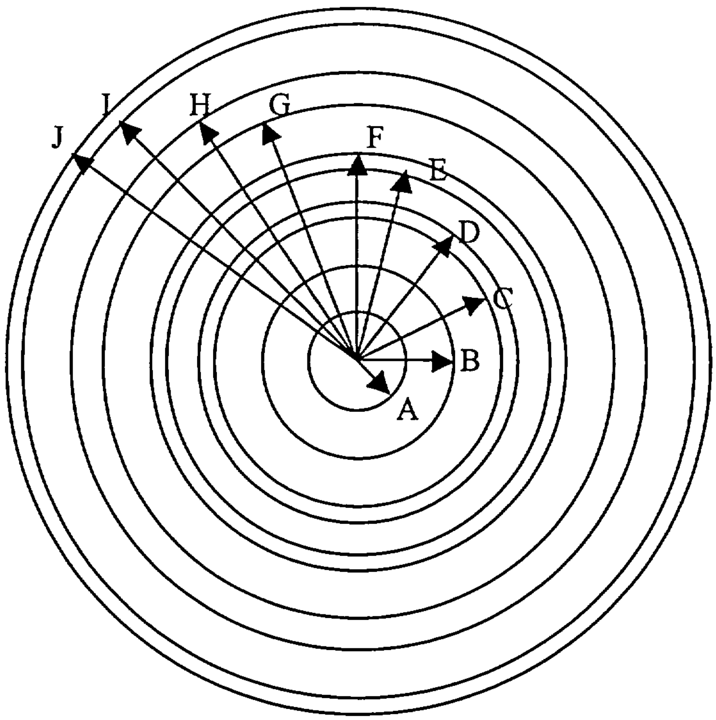
Relative Contribution of Source Terms to Dose Rates

Cask-81 Group	*Total γ/s/assembly*	Fraction γ/s/assembly	Response Function Dose Rate for HSM	Fraction of Dose Rate for HSM	Response Function Dose Rate for TC	Fraction of Dose Rate for TC
Configuration 2 Design Basis Source Terms						
35-29	4.192E+15	58%	90.2	96%	861 8	63%
38-40	2.829E+15	39%	0 0	0%	0 0	0%
Total	7.181E+15	100%	93 7	100%	1.370 2	100%
55 GWd/MTU, 3.4 wt. % U-235, 8-year Cooled Fuel						
35-29	3.868E+15	58%	80 2	98%	768 4	69%
38-40	2.710E+15	40%	0 0	0%	0 0	0%
Total	6 722E+15	100%	82.2	100%	1108 7	100%



Region	*Material*	*Outer Radius (cm)*	*Thickness (cm)*	*Thickness (in)*
A. Center Empty Region	*Air*	*26.0*	*26.0*	*10.2*
B. Source Region	*Fuel/Basket*	*72.0*	*46.0*	*18.1*
C. Gap between fuel/basket	*Air*	*84.0*	*12.0*	*4.7*
D. Canister Wall	*Stainless Steel*	*85.5*	*1.5*	*0.6*
E. Gap between DSC/HSM	*Air*	*95.25*	*9.75*	*3.83*
F. Thermal Shield	*Stainless Steel*	*95.65*	*0.4*	*0.157*
G. Gap	*Air*	*106.68*	*11.03*	*4.3*
H. Roof	*Concrete*	*198.12*	*91.44*	*36*

***Figure* N.5-19**
ANISN HSM Model



Region	Material	Outer Radius (cm)	Thickness (cm)	Thickness (in)
A. Center Empty Region	Air	26.0	26.0	10.2
B. 1.3 kW Source Region	Fuel	72.0	46.0	18.1
C. Gap between fuel/basket	Air	84.0	12.0	4.7
D. Canister Wall	Stainless Steel	85.33	1.5	0.6
E. Gap between DSC/Cask	Air	86.36	1.03	0.4
F. Inner Liner of Cask	Stainless Steel	87.67	1.31	0.52
G. Lead Gamma Shield	Lead	96.67	9	3.54
H. Cask Body	Stainless Steel	100.48	3.81	1.5
I. Neutron Shield	Water	108.59	8.11	3.19
J. Cask Skin	Stainless Steel	109	0.41	0.163

Figure **N.5-20**
ANISN TC Model

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 *(Configuration 2 from Chapter N.2)*. 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. *Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading and dose rates are not limits.*

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 *(Configuration 2 from Chapter N.2)*. 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 *(Configuration 2 from Chapter N.2)*. 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

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

Locations	Task Description	# of workers	Duration (hr)	Area Dose Rate (mrem/hr)	Total Exposure 24 PHBS (person-mrem)	Area Dose Rate w/BPRA (mrem/hr)	Exposure 24PHBL (person-mrem)
Auxiliary Building and Fuel Pool	Ready the DSC and TC for Service	2	4.00	0.00	0 00	0 00	0 00
	Place the DSC into the Transfer Cask	3	1 00	2 00	6.00	2.00	6.00
	Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2 00	2 00	8.00	2.00	8.00
	Fill the DSC Cavity with Water (borated for PWRs)	1	6 00	2.00	12.00	2.00	12.00
	Place the Cask Containing the DSC in the Fuel Pool	5	0 50	2.00	5 00	2.00	5 00
	Verify and Load the Candidate Fuel Assemblies into the DSC	3	5.00	2 00	30.00	2 00	30.00
	Place the Top Shield Plug on the DSC	2	1.00	2 00	4.00	2 00	4.00
	Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	5	0 50	2 00	5.00	2.00	5.00
		1	0.03	100.00	3.33	107.28	3.58
		1	1.00	67.13	67.13	72.02	72.02
Cask Decontamination Area	Decontaminate the Outer Surface of the Cask	1	1.75	67.13	117.48	72.02	126.03
		1	1.00	2.00	2.00	2.00	2.00
	Decontaminate the Top Region of the Cask and DSC	2	0.50	121.37	121.37	201.92	201.92
		1	1.00	2 00	2.00	2.00	2.00
	Drain Water Above DSC Shield Plug	1	0.08	100.00	8.33	107.28	8.94
	Remove Cask/DSC Annulus Seal and Set-Up Welding Machine	1	0.25	121.36	30.34	201.90	50 48
		1	1.25	98.89	123.61	164.52	205.65
		1	1.50	2.00	3 00	2.00	3.00
	Weld the Top Inner Top Cover to the DSC Shell and Perform NDE (PT)	2	6.00	2.00	24.00	2.00	24.00
		1	0.50	98 89	49.45	164 52	82.26
	Drain the Cask/DSC Annulus and the DSC Cavity	1	0 25	225.31	56.33	374 84	93.71
		1	0 02	423 99	7.07	705.38	11.76
		1	0.50	2.00	1.00	2.00	1.00
	Vacuum Dry and Backfill the DSC with Helium	Same as Draining			64 39		106 47
	Helium Leak Test the Shield Plug Weld	2	1.00	2.00	4 00	2.00	4 00
	Seal Weld the Prefabricated Plugs to the Vent	1	0 50	240.09	120.05	257.78	128.89
	Fit-Up the DSC Top Cover Plate	1	0 50	2.00	1.00	2.00	1.00
		1	0 50	240.09	120.05	257.78	128.89
	Weld the Outer Top Cover Plate and Perform NDE	1	1 25	240 09	300 11	257.78	322 23
		1	1.50	2.00	3.00	2.00	3 00
		2	14.00	2.00	56.00	2.00	56.00
		1	0 50	240 09	120 05	257.78	128.89
	Install the Cask Lid	2	1.00	56 27	112 54	59.16	118 31
Reactor/Fuel Building Bay	Ready the Cask Support Skid and Transport Trailer for Service	2	2.00	2.00	8.00	2.00	8.00
	Place the Cask onto the Skid and Trailer	2	0.50	200.00	200.00	214.56	214.56
	Secure the Cask to the Skid	1	0.25	200.00	50 00	214 56	53.64

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

Locations	Task Description	# of workers	Duration (hr)	Area Dose Rate (mrem/hr)	Total Exposure 24 PHBS (person-mrem)	Area Dose Rate w/BPRA (mrem/hr)	Exposure 24PHBL (person-mrem)
ISFSI Site	Ready the Cask Support Skid and Transport Trailer for Service	2	2.00	0.00	0.00	0 00	0.00
	Transport the Cask to ISFSI	6	1.00	0.00	0.00	0 00	0.00
	Position the Cask in Close Proximity with the HSM	3	1 00	0 00	0 00	0.00	0.00
	Remove the Cask Lid	2	1.00	162.79	325.58	171.14	342.29
	Align and Dock the Cask with the HSM	2	0 25	200 00	100 00	214.56	107.28
	Position and Align Ram with Cask	2	0.50	200.00	200.00	214 56	214.56
	Remove Ram Access Cover Plate	1	0 25	140.55	35.14	140.69	35.17
	Transfer the DSC from the Cask to the HSM	3	0.50	0.00	0 00	0.00	0.00
	Lift the Ram Back onto the Trailer and Un- Dock the Cask from the HSM	2	0 08	200.00	33.33	214 56	35.76
	Install HSM Access Door	2	0.50	106.52	106.52	107.39	107.39
Totals			66 22		2646		3075

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.

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

Two 1x10 Front To Front Array

Distance (meters)	Back Total Dose (mrem)	*1σ Error (mrem)*	*MCNP Relative Error*
6.096	13038	*58*	*0.004*
10	9692	*71*	*0.007*
20	5803	*56*	*0.010*
30	3863	*45*	*0 012*
40	2872	*46*	*0.016*
50	2171	*28*	*0.013*
60	1713	*23*	*0.013*
70	1382	*22*	*0.016*
80	1132	*17*	*0.015*
90	921	*14*	*0.015*
100	795	*25*	*0.032*
200	171	*6*	*0.037*
300	53	*4*	*0.073*
400	17	*0.6*	*0.038*
500	6	*0.2*	*0.032*
600	2	*0.1*	*0.047*

Distance (meters)	Side Total Dose (mrem)	*1σ Error (mrem)*	*MCNP Relative Error*
6.096	65274	*133*	*0.002*
10	43287	*119*	*0 003*
20	17571	*87*	*0.005*
30	9145	*46*	*0.005*
40	5624	*47*	*0.008*
50	3797	*31*	*0.008*
60	2731	*25*	*0.009*
70	2148	*94*	*0.044*
80	1583	*15*	*0.009*
90	1280	*15*	*0.012*
100	1016	*11*	*0.011*
200	197	*5*	*0.025*
300	62	*6*	*0.102*
400	18	*0.8*	*0.043*
500	7	*0.7*	*0.100*
600	3	*0.3*	*0.112*

2x10 Back To Back Array

Distance (meters)	Front Total Dose (mrem)	*1σ Error (mrem)*	*MCNP Relative Error*
6 096	93082	*174*	*0.002*
10	56142	*142*	*0.003*
20	21982	*62*	*0.003*
30	11596	*58*	*0.005*
40	7051	*36*	*0.005*
50	4698	*26*	*0.006*
60	3418	*29*	*0.008*
70	2538	*31*	*0.012*
80	1984	*31*	*0 016*
90	1550	*19*	*0.012*
100	1240	*15*	*0.012*
200	226	*4*	*0.020*
300	63	*2*	*0.026*
400	23	*1*	*0.054*
500	7	*0.4*	*0.048*
600	3	*0.1*	*0.042*

Distance (meters)	Side Total Dose (mrem)	*1σ Error (mrem)*	*MCNP Relative Error*
6 096	100758	*237*	*0.002*
10	49618	*131*	*0.003*
20	15730	*63*	*0.004*
30	7869	*40*	*0.005*
40	4844	*35*	*0.007*
50	3305	*33*	*0.010*
60	2399	*24*	*0.010*
70	1791	*17*	*0.009*
80	1455	*45*	*0.031*
90	1125	*15*	*0.013*
100	904	*12*	*0.013*
200	181	*4*	*0.020*
300	52	*2*	*0.040*
400	16	*0.6*	*0.038*
500	6	*0.2*	*0 035*
600	3	*0.2*	*0 082*

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

Distance (meters)	Gamma Dose Rate (mrem/hr)	*Gamma MCNP 1σ error*	Neutron Dose Rate (mrem/hr)	*Neutron MCNP 1σ error*	Total Dose Rate (mrem/hr)	*Combined MCNP 1σ error*
6.10E+00	9.28E+00	*0.0019*	1.35E+00	*0.0068*	1.06E+01	*0.0019*
1.00E+01	5.62E+00	*0.0026*	7.93E-01	*0.0088*	6.41E+00	*0.0025*
2.00E+01	2.21E+00	*0.0028*	2.96E-01	*0.0116*	2.51E+00	*0.0028*
3.00E+01	1.17E+00	*0.0052*	1.51E-01	*0.0180*	1.32E+00	*0.0050*
4.00E+01	7.16E-01	*0.0050*	8.92E-02	*0.0231*	8.05E-01	*0.0051*
5.00E+01	4.80E-01	*0.0058*	5.61E-02	*0.0205*	5.36E-01	*0 0056*
6.00E+01	3.51E-01	*0.0089*	3.87E-02	*0.0256*	3.90E-01	*0.0084*
7.00E+01	2.61E-01	*0.0122*	2.91E-02	*0.0553*	2.90E-01	*0.0123*
8.00E+01	2.06E-01	*0.0168*	2.08E-02	*0.0369*	2.27E-01	*0.0156*
9.00E+01	1.61E-01	*0.0129*	1.62E-02	*0.0342*	1.77E-01	*0.0121*
1.00E+02	1.29E-01	*0.0130*	1.25E-02	*0.0312*	1.42E-01	*0.0122*
2.00E+02	2.38E-02	*0.0210*	2.04E-03	*0.0506*	2.58E-02	*0.0197*
3.00E+02	6.44E-03	*0.0269*	7.30E-04	*0.0877*	7.17E-03	*0.0258*
4 00E+02	2.29E-03	*0.0582*	2.93E-04	*0.1265*	2.58E-03	*0.0536*
5 00E+02	7.41E-04	*0.0537*	1.01E-04	*0.0522*	8.43E-04	*0.0477*
6 00E+02	2.72E-04	*0.0402*	5.21E-05	*0.1517*	3.24E-04	*0 0416*

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

Distance (meters)	Gamma Dose Rate (mrem/hr)	*Gamma MCNP 1σ error*	Neutron Dose Rate (mrem/hr)	*Neutron MCNP 1σ error*	Total Dose Rate (mrem/hr)	*Combined MCNP 1σ error*
6.10E+00	1.31E+00	*0.0046*	1.76E-01	*0.0160*	1.49E+00	*0.0045*
1.00E+01	9.76E-01	*0.0081*	1.31E-01	*0.0142*	1.11E+00	*0.0073*
2.00E+01	5.83E-01	*0.0104*	7.98E-02	*0.0258*	6.62E-01	*0.0097*
3.00E+01	3.91E-01	*0.0126*	4.99E-02	*0.0318*	4.41E-01	*0.0117*
4.00E+01	2.91E-01	*0.0173*	3.64E-02	*0.0415*	3.28E-01	*0.0161*
5.00E+01	2.22E-01	*0.0133*	2.57E-02	*0.0423*	2.48E-01	*0.0127*
6 00E+01	1.78E-01	*0.0141*	1.79E-02	*0.0429*	1.96E-01	*0.0134*
7.00E+01	1.43E-01	*0.0172*	1.49E-02	*0.0432*	1.58E-01	*0.0161*
8.00E+01	1.18E-01	*0.0163*	1.14E-02	*0 0459*	1.29E-01	*0.0154*
9.00E+01	9.65E-02	*0.0159*	8.65E-03	*0 0409*	1.05E-01	*0.0150*
1.00E+02	8.41E-02	*0.0345*	6.65E-03	*0.0343*	9.07E-02	*0.0321*
2.00E+02	1.79E-02	*0.0380*	1.59E-03	*0.1496*	1.95E-02	*0.0370*
3.00E+02	5.40E-03	*0.0795*	5.99E-04	*0.1548*	6.00E-03	*0.0732*
4.00E+02	1.70E-03	*0.0382*	2.04E-04	*0.1613*	1.90E-03	*0.0382*
5 00E+02	5.76E-04	*0.0342*	8.45E-05	*0.0816*	6.61E-04	*0.0316*
6 00E+02	2.25E-04	*0.0403*	4.62E-05	*0.1906*	2.71E-04	*0.0466*

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

2x10 Back-to-Back Array

Distance (meters)	Gamma Dose Rate (mrem/hr)	*Gamma MCNP 1σ error*	Neutron Dose Rate (mrem/hr)	*Neutron MCNP 1σ error*	Total Dose Rate (mrem/hr)	*Combined MCNP 1σ error*
6.10E+00	1.12E+01	*0.0024*	3.27E-01	*0 0113*	1.15E+01	*0.0024*
1.00E+01	5.47E+00	*0.0027*	1.94E-01	*0.0134*	5.66E+00	*0.0026*
2.00E+01	1.71E+00	*0.0041*	8.41E-02	*0.0203*	1.80E+00	*0.0040*
3.00E+01	8.48E-01	*0.0049*	5.02E-02	*0.0359*	8.98E-01	*0.0050*
4.00E+01	5.20E-01	*0.0072*	3.27E-02	*0 0374*	5.53E-01	*0.0071*
5.00E+01	3.52E-01	*0.0084*	2.52E-02	*0.0950*	3.77E-01	*0.0101*
6.00E+01	2.57E-01	*0.0105*	1.70E-02	*0.0380*	2.74E-01	*0.0101*
7.00E+01	1.91E-01	*0.0091*	1.32E-02	*0.0561*	2.04E-01	*0.0093*
8.00E+01	1.55E-01	*0.0328*	1.07E-02	*0.0889*	1.66E-01	*0.0312*
9.00E+01	1.20E-01	*0.0136*	8.30E-03	*0 0549*	1.28E-01	*0.0132*
1.00E+02	9.62E-02	*0.0132*	7.06E-03	*0.0807*	1.03E-01	*0.0135*
2.00E+02	1.93E-02	*0.0205*	1.33E-03	*0.0738*	2.07E-02	*0.0198*
3.00E+02	5.46E-03	*0.0423*	4.94E-04	*0.1261*	5.96E-03	*0.0402*
4.00E+02	1.71E-03	*0.0409*	1.72E-04	*0.0836*	1.88E-03	*0.0379*
5.00E+02	5.74E-04	*0.0350*	7.88E-05	*0.1348*	6.53E-04	*0.0348*
6.00E+02	2.32E-04	*0.0495*	5.57E-05	*0.3695*	2.88E-04	*0.0818*

Two 1x10 Front-To-Front Arrays

Distance (meters)	Gamma Dose Rate (mrem/hr)	*Gamma MCNP 1σ error*	Neutron Dose Rate (mrem/hr)	*Neutron MCNP 1σ error*	Total Dose Rate (mrem/hr)	*Combined MCNP 1σ error*
6.10E+00	6.89E+00	*0.0020*	5.63E-01	*0.0113*	7.45E+00	*0.0020*
1.00E+01	4.62E+00	*0.0027*	3.18E-01	*0.0170*	4.94E+00	*0.0028*
2.00E+01	1.88E+00	*0.0050*	1.22E-01	*0.0255*	2.01E+00	*0.0049*
3 00E+01	9.79E-01	*0.0049*	6.45E-02	*0.0322*	1.04E+00	*0.0050*
4.00E+01	5.96E-01	*0.0059*	4.56E-02	*0.0887*	6.42E-01	*0.0083*
5.00E+01	4.06E-01	*0.0084*	2.72E-02	*0.0338*	4.33E-01	*0.0082*
6 00E+01	2.91E-01	*0.0095*	2.06E-02	*0.0429*	3.12E-01	*0.0093*
7.00E+01	2.30E-01	*0.0464*	1.48E-02	*0.0406*	2.45E-01	*0.0437*
8.00E+01	1.69E-01	*0 0089*	1.20E-02	*0 0668*	1.81E-01	*0.0094*
9.00E+01	1.37E-01	*0.0120*	9.24E-03	*0.0683*	1.46E-01	*0 0120*
1.00E+02	1.08E-01	*0.0103*	7.59E-03	*0.0773*	1.16E-01	*0.0109*
2.00E+02	2.08E-02	*0.0218*	1.70E-03	*0.1972*	2.25E-02	*0.0250*
3.00E+02	6.65E-03	*0.1090*	4.53E-04	*0.0616*	7.10E-03	*0.1021*
4.00E+02	1.92E-03	*0.0444*	1.92E-04	*0.1691*	2.11E-03	*0.0432*
5.00E+02	6.60E-04	*0.0377*	1.64E-04	*0.4772*	8.23E-04	*0.0996*
6.00E+02	2.66E-04	*0.0470*	6.75E-05	*0.5197*	3.33E-04	*0.1118*

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 *Configuration 2, from Chapter N.2, of* 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.

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 *for Configuration 2, from Chapter N.2,* 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. *This corresponds to the exposure to an individual at a distance of 100 meters of approximately 57 mrem for the assumed eight hour duration also 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.