Enclosure 6 to TN E-25820

Transnuclear, Inc. Calculation NUH06L-0501, "OS197L 75 Ton Transfer Cask As-Built Configuration Shielding Analysis," Revision 3 (without disks)

AREVA TRANSNUCLEAR INC.	**Form 3.2-1 Calculation Cover Sheet** TIP 3.2 (Revision 2)	Calculation No.: NUH06L-0501
		Revision No.: 3
		Page: 1 of 19

DCR NO (if applicable) : NUH06L-031	**PROJECT NAME: NUHOMS® OS197L Light Onsite Transfer Cask**
PROJECT NO: NUH06L	**CLIENT: Transnuclear, Inc.**

CALCULATION TITLE:

OS197L 75 Ton Transfer Cask As-Built Configuration Shielding Analysis

SUMMARY DESCRIPTION:

1) Calculation Summary

This calculation summarizes the salient features of Calculation NUH06L-0500 for the shielding configurations of interest as defined by the final as-built design configuration.

Revision 3 of this calculation is the insertion of a new Appendix that details the ALARA methodology that was incorporated into the design of the OS197L Transfer Cask. No new MCNP calculations are performed in this revision.

2) Storage Media Description

1 CD-ROM attached. The CD utilized in Revision 2 is unchanged in Revision 3.

If original issue, is licensing review per TIP 3.5 required? N/A since this is a revision

Yes ☐ No ☐ (explain below) Licensing Review No.: ____________

The DCR NUH06L-031 performs the TIP 3.5 reviews as required.

Software Utilized:	Version:
MCNP	5v1.2
MCNP	5v1.4

Calculation is complete: Yevgeny Terekhin (for Rev. 3) Originator Name and Signature: YEVGENIY TEREKHIN	Date: 12/20/07
Calculation has been checked for consistency, completeness and correctness: A. Prakash Checker Name and Signature: PRAKASH NARAYANAN	Date: 12/20/2007
Calculation is approved for use: PETER SHIH Project Engineer Name and Signature: P. Shih	Date: 12/20/07

REVISION SUMMARY

REV.	DATE	DESCRIPTION	AFFECTED PAGES	AFFECTED DISKS
0	3/15/2006	Initial issue	12	0
1	3/31/2006	To correct two typographical errors on page 11 of this calculation.	11	None
2	5/22/2006	Added Appendix A, confirmatory MCNP evaluation of the Accident	2-5 and 12-16	1
3	12/20/07	*Inserted Appendix B, the documentation of the ALARA methodology for design of the OS197L TC*	*2-5 and 12-19*	0

The OS197L is a general purpose, light weight (75 Tons) on-site transfer cask. A detailed shielding evaluation of the TC with various configurations is performed in Calculation NUH06L-0500 at normal and accidents conditions. This calculation summarizes the salient features of Calculation NUH06L-0500 for the shielding configurations of interest as defined by the final as-built design configuration. In summary, the dose rates around the transfer cask for NUH06L-0500 configurations, "B" and "F" are included in this calculation. Moreover, the accident dose rates are revised to reflect a total loss of neutron shield shells (inner and outer shell, thus reducing the total effective shield thickness from 3" to 2.68"). A "scaling" factor methodology is employed to determine the impact of reduced steel thickness for configuration "B" and "F" for accident dose rates.

This calculation is revised (revision 1) to correct two typographical errors in Page 11, Section 6.4 in revision 0. One of the corrections was to change "2000 feet" to "100 m" and the other change is to correct the SAR reference for NUHOMS® -32PT to Appendix M and not Appendix N.

Revision 2 of this calculation is an inclusion of Appendix A, the confirmatory MCNP evaluation of the accident. The design basis MCNP model from reference [2] is modified to reduce the total shield thickness by 0.45" of stainless steel and the accident condition dose rates are calculated. These values are compared to those calculated from the "scaling factor" methodology.

Revision 3 of this calculation is an insertion of a new Appendix B that describes the ALARA methodology that was incorporated into the design of the OS197L TC. This is included in support of the response to RAI-1 [7] for Amendment 11 application to CoC 72-1004 licensing submittal.

TABLE OF CONTENTS

LIST OF TABLES

1. PURPOSE

The purpose of this summary calculation is to calculate the radiation dose rates around the OS197L Transfer Cask (TC) with a fully loaded NUHOMS®-32PT DSC containing 32 design basis PWR fuel assemblies. A comprehensive shielding evaluation for the OS197L TC with a variety of shielding configurations is documented in reference [2]. This summary calculation provides the radiation dose rates as a function of distance from the TC surface for the configurations identified as "B" and "F" from the design basis reference [2] calculation.

The accident dose rates determined in the reference [2] calculations are re-evaluated for a scenario involving the total loss of neutron shielding, including the 0.5" of skin shells (inner plus outer shell thicknesses). These dose rates are, however, evaluated based on an exponential treatment of the the dose rate ratios. Finally, this calculation also provides a brief discussion on the expected dose rates during the operation of the above described DSC/TC system. Also addressed in this calculation is an assessment of the effect of the use of the OS197L cask on payloads including other C of C 72-1004 DSCs (all DSCs covered by reference [1]).

The MCNP ([4], [5]) evaluation of the accident with a reduction in the stainless steel thickness by 0.45" is evaluated in Appendix A of this calculation. The purpose of this revision (revision 2) of this calculation is to document the MCNP evaluation.

The ALARA methodology incorporated into the final design of the OS197L TC, primarily in the selection of Configuration B, is described in Appendix B of this calculation.

2. REFERENCES

2.1 Referenced Documents, Calculations, Publications etc.

1. UFSAR "Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel", Rev. 9.

2. Transnuclear Calculation NUH06L-0500, Revision 1, "Design of Integral Radiation Shield for On-Site Transfer Cask OS197-L and Calculation of Occupational Exposure due to 32PT DSC Design Basis Fuel."

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

4. MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, Volume II: User's Guide, LA-CP-03-0245, 2003.

5. MCNP5 v1.20 Computer Code Verification Record. Test Plan: TN File No. QA 040.230.0001, Test Report: TN File No. QA 040.230.0002.

6. "MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, January 2006.

7. *Letter from Joseph M. Sebrosky, Senior Project Manager, USNRC to Donis Shaw, Licensing Manager, Transnuclear, Inc. "REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF AMENDMENT 11 TO THE STANDARDIZED NUHOMS® SYSTEM (TAC NO. L24080)," Docket No. 72-1004, TAC No. L24080.*

8. *Transnuclear Design Report, NUH06L-0103, Revision 0, "NUHOMS® OS197L 75 Ton Design Report."*

2.2 List of Acronyms and conventions

DSC	Dry Shielded Canister
UFSAR	Updated Final Safety Analysis Report
TC	Transfer Cask, refers to OS-197 cask in general and its modifications unless specified otherwise
TN	Transnuclear Inc.

3. METHODOLOGY AND DESIGN INPUTS

3.1 Methodology

The shielding evaluation documented in reference [2] performs a comprehensive shielding evalulation of the DSC / TC system in a variety of shielding configurations. A detailed description of these shielding configurations is provided in Section 8, Table 16 of reference [2]. These shielding configurations are based on a variation in the thickness of the shielding materials (lead and stainless steel) in the TC. The dose rate results for the shielding configurations identified as "B" and "F" are extracted from reference [2] and are shown in the results of this summary calculation. The dose rates from the UFSAR (reference [1]) and those based on the OS197 TC (MCNP model of the UFSAR configuration, used as a baseline for comparison of the effects of the OS197L MCNP model with respect to the UFSAR configuration) are also shown for comparison. A description of the calculational methodology for determining these rates is provided in reference [2].

In addition, the accident dose rates for these shielding configurations are estimated for an accident involving a complete loss of the detachable neutron shield shell (inner and outer shells). Note that for accidents, the "F" configuration defaults to the "B" configuration because temporary shielding is not credited. This configuration results in a total steel thickness of 2.68" and is expected to result in higher accident dose rates than what was

analyzed in reference [2] which employed a total steel thickness of 3.0". The effect of a reduction in the stainless steel thickness to the accident dose rates is evaluated in this calculation. A simple exponential treatment of the dose rate attenuation in steel is employed to determine the accident dose rates.

The exponential attenuation method is based on the following formula:

$$D_t = D_0 * EXP(-\mu * t)$$

Where D_t is the shielded dose rate (shielding material thickness, t)
D_0 is the reference dose rate (without any shielding)
μ is the attenuation coefficient

Basically, if there is a radioactive source that results in a dose rate D_0 at an arbitrary location which is surrounded by a shielding material of thickness, *t,* and attenuation coefficient, μ, then the dose rate at the outer surface of the shielding material is D_t.

When dose rates based on two different thicknesses are known, then the attenuation coefficient can be determined, and subsequently, the dose rate for an unknown thickness can be determined. Above all, when dose rates based on two different thicknesses are known, then the reference dose rate (D_0) is not required. The accident dose rates for a reduced steel thickness are determined based on the above method.

3.2 Design Inputs

The dose rates as a function of distance for normal conditions are directly obtained from Table 14 of reference [2]. The configurations labeled "B" and "F" are those that are relevant for this purpose. Configuration "F" is identical to configuration "B" except that it contains temporary shielding.

For the accident conditions, the results shown in Table 15 of reference [2] are utilized as starting design input. The results for configuration "B" (3.0" steel thickness) and configuration "H" (6.0" steel thickness) are utilized to determine the accident dose rates for the configuration "B" with reduced steel thickness.

Section 1.1 of reference [2] indicates that the minimum steel thickness of the OS197-L TC without the neutron shield shell is 2.68 inches. However, for conservatism the design input for this value is conservatively utilized as 2.55 inches. Therefore, the accident dose rates will be calculated with a stainless steel shell thickness of 2.55 inches.

4. ASSUMPTIONS AND CONSERVATISMS

All assumptions and conservatisms applicable to the calculational methodology employed in the reference [2] calculations are assumed to be valid since these results are directly utilized in this summary calculation. In addition, the following assumptions and conservatisms are employed in this calculation.

1. The thickness of the Stainless Steel utilized to calculate the accident dose rates is conservatively assumed to be 2.55″ while the actual thickness is 2.68″.

2. The accident dose rates are calculated based on exponential attenuation of the gamma and neutron particles. This assumption is reasonable and combined with a conservative assumption for the thickness of the material is expected to result in acceptable dose rates.

5. CALCULATIONS

5.1 Attenuation Calculations

The accident dose rates from Table 15 of reference [2] are utilized to determine the attenuation coefficients (or exponential ratios). The dose rates due to the 3″ and 6″ shield shells can be expressed utilizing the formula described in Section 3.1 and these expressions are shown below:

$$D_3 = D_0 * EXP(-\mu * 3)$$
$$D_6 = D_0 * EXP(-\mu * 6)$$

where D_3 is the dose rate with a 3″ thick steel shell and
D_6 is the dose rate with a 6″ thick steel shell.

Combining these equations would result in the elimination of the reference dose rate (D_0) and yield an expression for the attenuation coefficient, μ. This expression is shown below:

$$\mu = (1/(6-3)) * Ln(D_3/D_6)$$

[redacted] These calculations are shown in Table 1. The attenuation coefficients are shown as “ratios” for both neutrons and gamma-rays and these values are shown in **bold** in Table 1.

Table 1 Calculation of the Shielding Attenuation Coefficients

Transfer Cask Configu-ration	Dose Rate Component	Dose Rates at Different Distances from Side Surface							
		On Side Surface		4.57 meters (15')		100 meters		609.9 meters (2000')	
		Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
B	Neutron	3,176.08	0.02	157.33	0.01	0.17	0.09	8.18E-05	0.55
	Gamma	83,570.33	0.06	6,999.14	0.02	7.84	0.1	1.80E-02	0.63
	Total	86,690.95	0.06	7,151.74	0.02	8	0.1	1.81E-02	0.63
H	Neutron	1,166.75	0.01	49.9	0.02	0.054	0.02	1.44E-05	0.24
	Gamma	3,531.41	0.04	244.5	0.08	0.32	0.04	3.96E-04	0.37
	Total	4,686.84	0.03	263.79	0.05	0.37	0.03	4.12E-04	0.36
Ratio	**Neutron**	**0.33381**		**0.38277**		**0.38227**		**0.57902**	
	Gamma	**1.05466**		**1.11811**		**1.06622**		**1.27224**	

The dose rates due to the 2.55″ and 3.00″ shield shells can be expressed utilizing the formula described in Section 3.1 and these expressions are shown below:

$$D_{2.55} = D_0 * EXP(-\mu * 2.55)$$
$$D_{3.00} = D_0 * EXP(-\mu * 3.00)$$

where $D_{2.55}$ is the dose rate with a 2.55″ thick steel shell (needs to be calculated) and $D_{3.00}$ is the dose rate with a 3.00″ thick steel shell (already available).

Combining these equations would result in the elimination of the reference dose rate (D_0) and yield an expression for the required dose rate, $D_{2.55}$. This expression is shown below:

$$D_{2.55} = D_{3.00} * Exp(\mu * 0.45)$$

where, μ is the attenuation coefficient (or ratio) as determined in Table 1 above.

6. RESULTS

[redacted]

6.1 Dose Rates for Normal Conditions

The dose rates as a function of distance for the TC is shown in Table 2. The SAR configuration relates to the DSC/TC configuration employed in the FSAR, reference [1]. The OS-197 configuration relates to the same configuration except that the shielding analysis is performed utilizing a 3D calculational methodology (MCNP computer code) as well as incorporating additional features of the DSC model that could not be modeled in the original 2D analysis and demonstrates the underlying conservatism in the FSAR methodology. The dose rates for the configurations "B" and "F" are also shown in Table 2 since they represent the as-built configuration for the OS197L TC.

Table 2 Maximum Dose Rates for Normal Conditions

Transfer Cask Configu-ration	Dose Rate Compo-nent	Dose Rates at Different Distances from Side Surface							
		On Side Surface		4.57 meters (15')		100 meters		609.9 meters (2000')	
		Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
SAR	Neutron	261.0	N/A	Not Calc.	N/A	Not Calc.	N/A	Not Calc.	N/A
	Gamma	784.0	N/A	Not Calc.	N/A	Not Calc.	N/A	Not Calc	N/A
	Total	950.0	N/A	Not Calc	N/A	Not Calc	N/A	0.01	N/A
OS-197	Neutron	102.52	0.03	7.20	0.01	0.006	0.06	7.09e-6	0.45
	Gamma	248.4	0.04	20.25	0.03	0.03	0.04	5.29e-5	0.13
	Total	346.5	0.04	25.90	0.03	0.03	0.03	5.67e-5	0.13
B	Neutron	247.54	0.04	18.19	0.02	0.018	0.10	2.19e-5	0.61
	Gamma	53,031.83	0.05	3,905.74	0.04	4.52	0.07	9.70e-3	0.69
	Total	53,249.75	0.05	3,921.85	0.04	4.53	0.07	9.70e-3	0.69
F	Neutron	28.27	0.02	1.98	0.01	0.002	0.03	1.31e-6	0.50
	Gamma	93.66	0.05	11.25	0.03	0.02	0.07	2.44e-5	0.28
	Total	121.19	0.04	13.02	0.03	0.02	0.07	2.57e-5	0.26

6.2 Dose Rates for Accident Conditions

The dose rates as a function of distance for accident conditions are shown in Table 3 below. As discussed in Section 6.1 , the dose rates for SAR and OS-197 accident configurations are included for comparison. The dose rates for configuration "B" with the 2.55″ stainless steel thickness is calculated by applying the attenuation factor methodology to the reference [1] dose rates. As explained in Section 3.1 , the accident dose rates for the "F" configuration are not calculated separately since, they are bounded by those for the "B" configuration. Note that the relative errors for the "B" configuration are not calculated and therefore are not shown in Table 3.

Table 3 Maximum Dose Rates for Accident Conditions

Transfer Cask Configu-ration	Dose Rate Compo-nent	Dose Rates at Different Distances from Side Surface							
		On Side Surface		4.57 meters (15')		100 meters		609.9 meters (2000')	
		Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
SAR	Neutron	3,780.0	N/A	Not Calc.	N/A	Not Calc.	N/A	Not Calc.	N/A
	Gamma	1,070.0	N/A	Not Calc.	N/A	Not Calc.	N/A	Not Calc	N/A
	Total	4,640.0	N/A	Not Calc	N/A	Not Calc	N/A	0.01	N/A
OS-197	Neutron	1,282.2	0.01	66.00	0.02	0.067	0.01	1.87e-5	0.22
	Gamma	290.6	0.06	30.6	0.06	0.04	0.08	5.14e-5	0.19
	Total	1569.3	0.01	84.00	0.04	0.10	0.03	6.48e-5	0.19
B	Neutron	3,691		187		0.202		1.06e-4	
	Gamma	134,328		11,576		12.67		3.19e-2	
	Total	138,019		11,763		12.87		3.20e-2	

6.3 Discussion on Occupational Dose Rates

The occupational dose rates with the OS197L (75 Ton) cask are not calculated explicitly. Rather, the occupational dose rates during loading and transfer with this TC is judged to be bounded by those documented in the FSAR. This assertion is made because the shielding characteristics of the OS197L cask is not different from the one documented in the FSAR during all modes of operation where the 75 Ton maximum weight restriction is imposed. In other words, operations that do not require the 75 Ton maximum weight restriction (all operations other than transfer of the cask from the pool to the decontamination area and from the decontamination area to the trailer) are performed in such a way that a combination of remote crane operation, temporary shielding and other ALARA practices result in dose rates that are bounded by those calculated in the FSAR. The operations that require the 75

Ton maximum weight restriction are expected to be performed either remotely or with absolutely minimum personnel exposure. Therefore, the total occupational dose rates for the loading and transfer with the OS197L TC are bounded by those determined in the FSAR.

6.4 Use of Dose Rates Calculated for 32PT-DSC in the OS197L Cask for other OS197L Payloads

The calculations provided above and in NUH06L-0500 are specific to the OS197L with the 32PT-DSC payload. The ratios of dose rates (Configuration "B" versus OS-197 and Configuration "F" versus OS-197) are applicable to other OS197L payloads. These ratios may be applied to the payload for any DSC in order to estimate the corresponding dose rate for the OS197L (surface dose rates as well as dose rates at various distances, including off-site dose rates) when loaded with the specific DSC payload.

Note: Offiste dose rates, for the cask drop loss of neutron shield accident, for all DSCs in C of C 72-1004, with the exception of the 24PTH DSC, do not consider attenuation of the cask dose through air (they conservatively calculate the site boundary dose based on a $1/r^2$ non-attenuated dose rate estimate). The use of the ratio of OS-197 dose rates to the OS197L configuration "B" and "F" dose rates can be applied to other DSC surface dose rates and site boundary dose rates as follows:

For dose rates associated with activities in the decontamination area, transfer from the fuel handling building to the ISFSI or site boundary:

DSC dose rate for OS197L activities =

$$\text{DSC dose rate for OS197 activities (reference [1])} \times \left[\frac{[\text{Dose rate for applicable OS197L cask configuration (from Sections 6.1 \& 6.2)}]}{[\text{OS197 dose rate for the NUHOMS}^{\circledR}\text{ 32PT DSC from Appendix M of the FSAR}]}\right]$$

For example the site boundary total dose for a cask drop loss of neutron shield accident at 100 m for the 24PHB DSC in an OS197L cask would be:

57 mrem (Section N.11.2.5.3 of UFSAR) x [12.87*8 (configuration "B" accident dose rate at 100 m times 8 hours)]/[42 mrem (Section M.11.2.5.3 of UFSAR)] = 140 mrem

The above dose is based on the canister with the highest offsite dose consequences as documented in the UFSAR. This increase in dose constitutes an increase in the accident dose of 1.7% of the margin between the original dose reported and the maximum allowable offsite dose for an accident (5 Rem) [(140-57)/(5000-57)].

7. SUMMARY AND CONCLUSIONS

Shielding configurations with the OS197L on-site transfer cask containing the NUHOMS®-32PT design basis assemblies for normal and accident configurations have been determined. The configurations corresponding to "B" and "F" detailed in the reference [1] calculations have been reported in this summary calculation. A simple, exponential attenuation method is employed to determine the accident dose rates following a complete loss of the removable neutron shield shell. Note that this scenario was not evaluated in the reference [2] calculations. The results of the confirmatory MCNP calculations for the accident dose rates for configuration "B" indicate that the "scaling" factor methodology compares very closely (most cases, conservatively) with MCNP.

Based on the discussion in Section 6.3 regarding the conduct of loading and transfer operations, it is concluded that the occupational dose rates would be bounded by those determined in the FSAR.

A method for quantifying the effect of OS197L dose rates when used with other DSC payloads is provided in Section 6.4. The maximum offsite dose consequences for all of the DSCs described in the FSAR are due to the NUHOMS® -24PHB DSC. Applying the method described in Section 6.4 to this DSC with the OS97L TC configuration "B" yields an increase in the off-site dose rate by about 80 mrem which represents a 1.6% reduction in the available margin to the 5 rem dose limit and is considered to have a negligible impact on the dose consequences.

An ALARA evaluation documented in Appendix B of this calculation demonstrates that the current design configuration of the OS197L TC (Configuration B) provides for the most optimum configuration within the design constraints imposed by all disciplines.

8. APPENDIX A: MCNP EVALUATION

8.1 Methodology, MCNP Models and Assumptions

The shielding evaluation documented in reference [2] performs a comprehensive shielding evalulation of the DSC / TC system in a variety of shielding configurations. A detailed description of these shielding configurations is provided in Section 8, Table 16 of reference [2]. These shielding configurations are based on a variation in the thickness of the shielding materials (lead and stainless steel) in the TC. For the purpose of this evaluation, however, only one configuration detailed in the above reference is utilized. This configuration is the "OS-197L" configuration B under accident conditions. The evaluation performed in this appendix is for the same configuration B under accident conditions except that the thickness of the stainless steel is reduced by 0.45" consistent with the evaluation performed in this calculation.

The MCNP models from reference [2], "abg.mi" and "abn.mi" for gamma and neutron dose rate evaluations are used as starting models for the evaluation in this appendix. The stainless steel thickness is reduced by 0.45" – the cells utilizing surface numbers 20 and 21 are modeled with air instead of stainless steel. This ensures that the total amount of stainless steel credited in this evaluation is consistent with what is modeled in this calculation package. The "scaling factor" methodology is considered acceptable if the results based on the MCNP evaluation are within 5% of those predicted in Table 3. It is expected that the "scaling factor" methodology would yield acceptable if not conservative results in comparison to the MCNP results.

The 3-D Monte Carlo particle transport computer code, MCNP5 (reference [4], validation and verification reference [5]), is utilized to determine the dose rates. MCNP is a state of the art computer code and has been utilized by Transnuclear for shielding evaluations in NRC approved applications, including the calculations documented in the UFSAR, reference [1]. MCNP5 has also been utilized in the reference [2] calculations.

[redacted]

[redacted]

All assumptions and conservatisms applicable to the calculational methodology employed in reference [2] calculations are assumed to be valid since the modification to the MCNP models are considered minimal. No other specific assumptions or design inputs are utilized. The neutron dose rate evaluations are performed with a conservative reduction in the steel thickness by 0.50″ while the gamma dose rate evaluations are performed with the actual reduction in the steel thickness by 0.45″.

8.2 MCNP Results for Accident Conditions

The MCNP calculated dose rates as a function of distance are shown in Table 4, below. These accident dose rates are compared to those calculated using the "scaling" factor (also shown in Table 3) and are also shown in Table 4. These comparisons indicate that the "scaling" factor methodology compares very closely (most cases, conservatively) with MCNP.

Table 4 Dose Rates Comparison for Accident Conditions

Transfer Cask Configu-ration	Dose Rate Compo-nent	Dose Rates at Different Distances from Side Surface							
		On Side Surface		4.57 meters (15')		100 meters		609.9 meters (2000')	
		Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
B (Table 3)	Neutron	3,691		187		0.202		1.06e-4	
	Gamma	134,328		11,576		12.67		3.19e-2	
	Total	138,019		11,763		12.87		3.20e-2	
B (MCNP)	Neutron	3,377.15		183.19		0.194		7.55e-5	
	Gamma	132,424.6		10,841.6		13.01		1.51e-2	
	Total	135,259.8		11,024.8		13.18		1.52e-2	

Note: The relative errors in the MCNP calculations are generally within acceptable levels except for the tallies at 609.9m. This is consistent with the reported MCNP errors in reference [2].

9. APPENDIX B: ALARA EVALUATION

9.1 Methodology and Design

The OS197L TC is designed to provide adequate structural, thermal and radiation protection to the DSC and its contents. In addition, the OS197L is designed to maintain dose rates ALARA on and around the TC during loading and transfer operations. This includes selection of appropriate shielding geometry and materials.

The methodology for performing an ALARA evaluation is outlined below:

- *list the various design requirements of the OS197L TC,*
- *determine the limiting design features that are absolutely necessary to maintain the "protection" functions (structural and thermal) of the OS197L TC,*
- *determine if there is additional allowance in design to enhance ALARA,*
- *identify alternate material and geometric designs for this purpose, if available,*
- *perform sensitivity shielding calculations to determine the effect of alternate designs, and*
- *implement the design that results in ALARA enhanced dose rates and also satisfies the requirements of other disciplines.*

The primary design requirements of the OS197L TC are:

- *The maximum weight of the TC and its contents shall be below 75 Tons [8].*
- *The minimum inner diameter of the TC is approximately the same as that for the OS197 TC – 68 inches [1].*
- *Use of liquid neutron shield to satisfy the 32PT DSC thermal performance requirements [1].*

The limiting design features that are absolutely necessary to maintain the "protection" functions of the OS197L TC are listed below:

- *The UFSAR, reference [1], requires that the TC (based on the OS197 TC design) is designed with, at a minimum, a stainless steel inner shell with a thickness of 0.5 inches and a stainless steel outer shell with a thickness of 1.5 inches, to provide adequate structural protection.*
- *The two-piece neutron shielding design of the OS197L TC requires a total stainless steel thickness of 0.45 that includes the inner (0.25 inches) and outer (0.20 inches) neutron shield shells. The drawings for the OS197L TC are listed in the design report [8].*

Since the design of the OS197L TC (configuration B) contains a stainless steel thickness of 2.68 inches for the inner shell [8] and structural shell (excluding the neutron shielding steel thickness), only 0.68 inches of steel (or material equivalent by weight) is available to enhance ALARA. Due to the requirement of enhanced gamma shielding, the materials for consideration are steel and lead. For configuration B, the design consists of a single stainless steel shell with a thickness of 2.68 inches that provides structural, thermal and radiation protection of the DSC and its contents.

9.2 Design Alternative

Based on meeting the required structural and thermal performance requirements described above, an alternative design can be conceptualized for enhancing ALARA since the 0.68 inches of steel could to replaced with a different material. For this alternative design, the TC will consist of a stainless steel inner shell with a thickness of 0.5 inches, a lead gamma shield shell with a thickness of 0.45 inches (maximum based on replacing 0.68 inches of steel with lead) and a stainless steel outer shell with a thickness of 1.5 inches. This alternate design is equivalent to configuration B from a weight standpoint.

9.3 Analysis, Results and Conclusions

Sensitivity calculations were performed in calculation NUH06L-0500, revision 1, [2] to determine the effect of lead thickness on the dose rates for the OS197L TC with the 32PT DSC. Two configurations were investigated: one with a lead shell thickness of 0.50 inches (configuration A) and another with a lead shell thickness of 0.75 inches (configuration C). Both these configurations considered an inner steel shell with a thickness of 0.5 inches and an outer steel shell with a thickness of 1.5 inches. Due to the relatively insignificant contribution of the neutron dose rates to the total dose rates, only the results from the gamma dose rates calculations will be utilized to perform the dose rate comparison for these different configurations. The results for these sensitivity calculations are obtained from Table 8 of reference [2] and are shown below:

Configuration A, Gamma Dose rate = 86.3 rem/hour
Configuration B, Gamma Dose rate = 95.6 rem/hour
Configuration C, Gamma Dose rate = 47.7 rem/hour

As concluded in reference [2], Configuration C provides the best option for shielding as the dose rates are approximately 50% of those from either Configuration A or Configuration B. However, Configuration C that results in the lowest dose rates is not considered for ALARA enhancement because the thickness of lead far exceeds the maximum allowable thickness of 0.45 inches (from Section 9.2, above) and therefore does not meet the structural performance requirements.

The results for Configuration A indicate that the dose rates with a lead thickness of 0.5 inches are only slightly lower (approximately 10%) than that for Configuration B. A simple exponential extrapolation for lead thickness indicates that the dose rates with a lead thickness of 0.45 inches are expected to be within the same range as those for Configuration B. These dose rates are calculated assuming that the density of lead within the TC at 100% of theoretical density with no porosity. In actual practice, porosity free pouring of lead at a thickness of 0.45 inches or less over a 150 inch length is difficult to achieve and will likely result in an effective lead thickness of 0.40 inches or lower. Also, pouring of lead in a 0.45" cavity causes fabrication challenges and is not normally recommended.

In addition, use of a three shell configuration (inner, lead and outer) results in the degradation of the thermal performance of the TC due to introduction of additional thermal resistance introduced by the presence of extra gaps between the various "layers" or shells.

In conclusion, an alternate configuration for the OS197L TC is evaluated for ALARA purposes and is determined to have the following features:

- *Shielding design features that are similar to Configuration B assuming no further shielding performance degradation due to lead pouring effectiveness and/or lead slump assumptions,*
- *Degraded thermal performance compared to Configuration B, and*
- *More robust structural performance due to the presence of a single 2.68" stainless steel shell compared to the OS197 TC.*

Therefore, the optimum design of the OS197L TC, from an ALARA standpoint, is also due to Configuration B.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]