

## Real-Time SNF Cask External Dose Calculation Using RAPID

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**Abstract** - A new capability of RAPID (Real-time-Analysis for Particle-transport In-situ Detection) tool with real-time calculation on the Spent Nuclear Fuel (SNF) Cask external dose calculation is introduced in this paper. The GBC-32 Westinghouse  $17 \times 17$  Fuel Assemblies (FAs) is used as the calculation model. The fuel composition introduced here is 50 GWD/MTU burnup with no cooling time. In this study, neutron dose is calculated at two locations on the cask surface, both at axial midplane, but at different azimuthal angles of 0.0 and 41.28. The RAPID and MCNP calculated doses are in good agreement considering that the RAPID algorithm uses a multigroup cross-section library for the adjoint calculation. RAPID obtains its solution in about 10 min on a single-processor laptop, while MCNP requires over 19 hours using a 16-processor cluster.

### I. INTRODUCTION

To ensure criticality safety and material safeguards of a Spent Nuclear Fuel (SNF) facility, accurate calculations of radiation dose is required. We have developed a novel tool, RAPID (Real-time-Analysis for Particle-transport In-situ Detection), based on Multi-stage Response-function Methodology (MRT) [1]. The tool utilizes the Fission Matrix (FM) [2] approach to calculate the total source (due to both subcritical multiplication and intrinsic source), i.e., fission densities, and the adjoint function methodology to calculate dose value. Unlike the time-consuming traditional Monte Carlo calculations, the RAPID tool is capable of calculating accurate eigenvalue, 3D fission densities, dose values for different locations and dosimeter types real time that will be highly beneficial for online monitoring applications. RAPID utilizes databases of pre-calculated FM coefficients and importance functions which allow for solution via linear systems of equations which avoids the source convergence or power-iteration correlation issues which plague the standard eigenvalue Monte Carlo methods [3].

In our previous work, the capability of RAPID to calculate accurate system eigenvalue and pin-wise, axially-dependent fission densities for the GBC-32 Cask [4] and a SNF storage pool [5] has been demonstrated.

In this study, the capability of RAPID for calculating accurate radiation dose using the adjoint methodology [6] is demonstrated. A database of importance function distributions as a function of different parameters (i.e. burnup, cooling time, and dosimeter position) is generated using the 3-D parallel TITAN deterministic transport code system [7].

### II. THEORY

The structure of the RAPID code system is discussed, followed by the adjoint function methodology and the determination of dosimeter FOV.

#### 1. RAPID code structure

The RAPID code system is described as follow:

#### A. Pre-Calculations

1. Burnup calculations using SCALE/TRITON [8]
2. Generation of FM coefficient database through a set of MCNP [9] fixed-source calculations
3. Dosimeter Field of View calculation
4. Calculation of the important functions using TITAN and generate a RAPID inner built important functions database

#### B. Real-Time Calculations

1. Evaluation of pin-wise axial neutron source distribution using FM approach [2]
2. Calculation of dose using Adjoint Function Methodology [6]

### 2. The Fission Matrix approach

For a sub-critical system such as SNF casks fission source due to the subcritical multiplication caused by intrinsic sources (spontaneous fission and alpha-n interaction) is obtained using Eq.1 .

$$F_i = \sum_{j=1}^N (a_{i,j}F_j + b_{i,j}S_j), \quad (1)$$

where  $F_j$  represent the fission neutron source strength density in fuel pin  $j$ ,  $S_j$  is the intrinsic neutron source in the fuel pin  $j$ ,  $a_{i,j}$  is the number of fission neutrons directly produced in fuel pin  $i$  due to a fission neutron born in the fuel pin  $j$ ,  $b_{i,j}$  is the same as  $a_{i,j}$  except is for the intrinsic neutron source.  $N$  is the total number of computation cells (i.e. fuel pin segments). This linear system of equations is solved iteratively. FM coefficients,  $a_{i,j}$  and  $b_{i,j}$ , are calculated through a set of fixed source Monte Carlo calculations.

### 3. Dose Calculations by Adjoint Function Methodology

The adjoint function methodology for calculation of dose can be written as:

$$R = \langle Q\psi^* \rangle \quad (2)$$

where  $\langle \rangle$  is the Dirac notation, which refers to integration over all the independent variables (space, energy, and direction),  $R$  is the dose,  $Q$  is the total neutron source (i.e. due to both subcritical multiplication fission neutron source and intrinsic neutron source), the  $\psi^*$  is the importance function of specific dosimeter position. The importance function is determined by solving the importance (adjoint) function equation given by Eq.3 using the TITAN code system.

$$H^*\psi^* = Q^* \quad (3)$$

where  $H^*$  refers to the adjoint operator [6], the  $\psi^*$  refers to the importance function of specific dosimeter position, and  $Q^*$  refers to the importance source.

### 4. Field of View

It is desired to see how deep the dosimeter can “see” into a system. For this purpose, the Group-wise Contribution (GC) is defined as:

$$GC_g = \frac{\sum_i \psi_{ig}^* Q_{ig} V_i}{\sum_i \sum_g \psi_{ig}^* Q_{ig} V_i} \quad (4)$$

The FOV is defined as:

$$FOV_i = \frac{\sum_g \psi_{ig}^* Q_{ig} V_i}{\sum_i \sum_g \psi_{ig}^* Q_{ig} V_i} \quad (5)$$

where  $i$  refers to spatial location (i.e. fuel pin location),  $g$  refers to the energy group,  $Q_{ig}$  refers to the source in  $i^{th}$  spatial location and  $g^{th}$  energy group,  $\psi_{ig}^*$  refers to the importance function in  $i^{th}$  spatial location and  $g^{th}$  energy group, and  $V_i$  refers to the volume of the  $i^{th}$  spatial location (i.e. fuel pin).

## III. GBC-32 CASK MODEL

### 1. MCNP Model

The GBC-32 cask is loaded with 32 Westinghouse 17x17 OFA/V5 fuel assemblies (FA) of different burnups [10]. For this, the cask has been loaded with FAs of 50 GWD/MTU burnt fuel with no cooling time. The model consists of the 32 FA contained in a stainless steel canister. The canister is equipped with Boral plates encased in Aluminum clads between the FAs, and it is placed in a Stainless Steel cylinder. For this benchmark, the cask is flooded with water. The model is depicted in Fig.1. The detailed fuel assembly configuration is shown in Fig. 2. The MCNP models in this study are used for RAPID FM coefficient generation and the forward fixed-source dose reference MCNP calculations. Note that the control volume of fixed-source dose reference calculations is the same as the TITAN importance model, which is discussed in the following section.

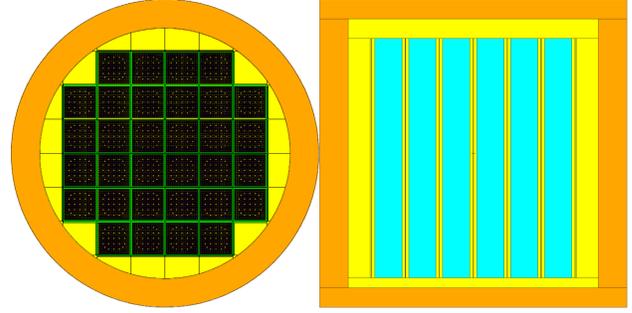


Fig. 1:  $xy$  (at mid active height) and  $xz$  (at  $y = 0$ ) cuts of the GBC-32 Cask model.

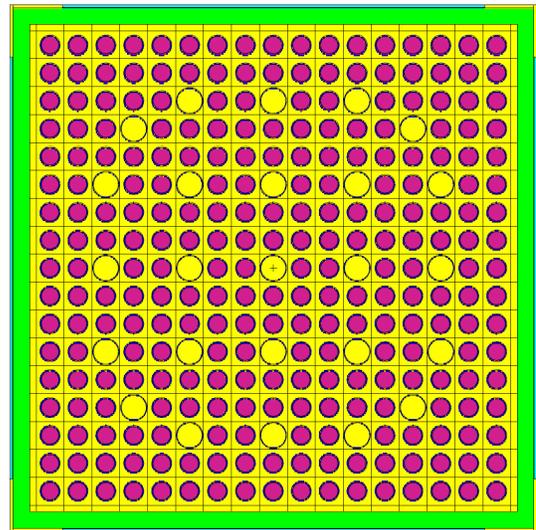
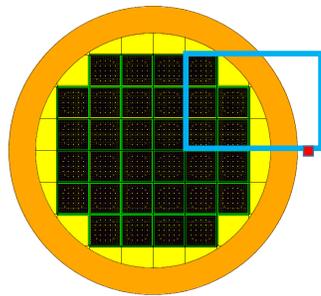


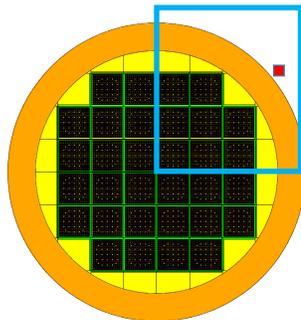
Fig. 2:  $xy$  (at mid active height) cut of the unit cell of the GBC-32 Cask model.

### 2. TITAN Importance Function Model

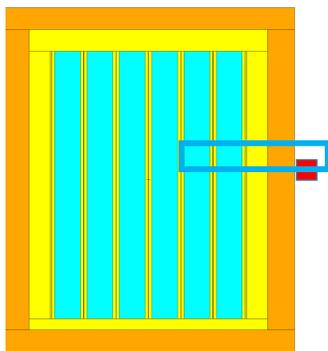
As discussed earlier, it is important to determine the FOV of a dosimeter so that both forward Monte Carlo and adjoint deterministic become more efficient while preserving solution accuracy. Here, we consider two neutron dosimeters of the same size  $2.5 \text{ cm} \times 2.5 \text{ cm} \times 5.0 \text{ cm}$ . These dosimeters are placed on the cask surface, both at axial midplane, but at two azimuthal angles of  $0.00^\circ$  and  $41.28^\circ$ , as depicted in Fig.3. We have determined the FOV by increasing the number of assemblies. This analysis resulted in a calculation volume identified by the blue square overlaid on the cask diagram in Fig.3. The Fig.3 (a) shows the FOV of the dosimeter located at  $0.00^\circ$ , and the Fig.3 (b) shows the  $41.28^\circ$  neutron dosimeter. Fig. 3 (c) depicts the axial height considered for both dosimeter locations.



(a) Radial projection of 0.00° dosimeter.



(b) Radial projection of 41.28° dosimeter.

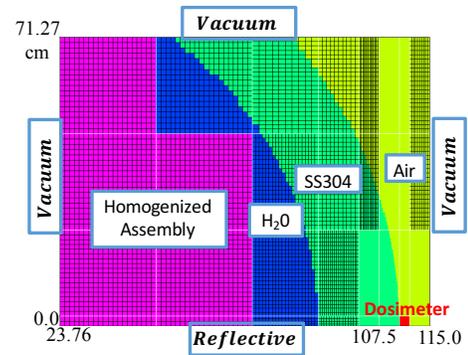


(c) Axial projection

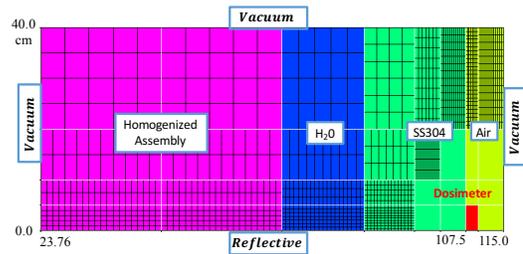
Fig. 3: Schematic of the GBC-32 Cask with Dosimeter.

Our analysis resulted in two TITAN models for the two dosimeters, as shown in Figs. 4 and 5. For the 0.00° dosimeter, model size is  $91.24 \times 71.27 \times 40.00 \text{ cm}^3$ , and the mesh sizes vary from 0.1 cm to 4.0 cm. For the 41.28° dosimeter, the model size is  $103.27 \times 115.0 \times 40.0 \text{ cm}^3$ , with mesh sizes ranging from 0.04 cm to 5.94 cm. For both models, we examine different angular quadrature orders, and use the multigroup BUGLE-96 cross-section library [11] with 47 neutron groups and anisotropic scattering order of P3.

Note that the homogenized fuel assembly material used in the importance function calculations are all fresh fuel composition. It is assumed that the variation of importance function due to burnt fuel material composition changes is negligible.

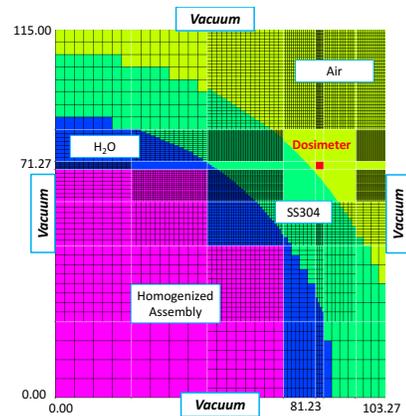


(a) xy Projection.

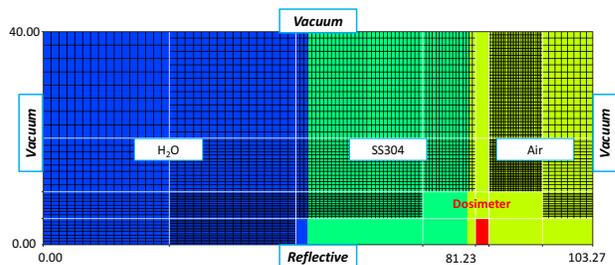


(b) z Projection

Fig. 4: TITAN GBC-32 Cask Model 0.00° dosimeter.



(a) xy Projection.



(b) z Projection

Fig. 5: TITAN GBC-32 Cask Model 41.28° dosimeter.

### A. Importance Source Spectrum

For this analysis, to determine neutron dose, we utilize the flux-to-dose conversion factors from ANSI/ANS-6.1.1 standard [12] as the importance source. By interpolating the conversion factors, a 47-group source is obtained. The comparison between original and interpolated flux-to-dose conversion factors is shown in the Fig. 6.

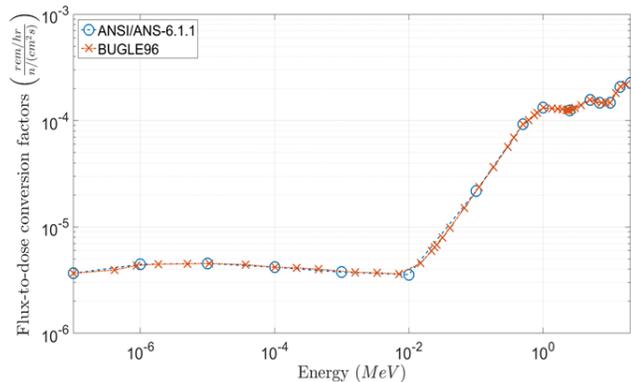


Fig. 6: Flux-to-dose conversion factor

## IV. RESULTS AND ANALYSIS

This section is divided into four parts, 1) the importance function calculated by TITAN and the neutron source calculated by RAPID, 2) the calculated dose using Eq. 2, 3) the group-wise contribution of dose, and 4) the spatial dosimeter FOV.

### 1. Importance function/Neutron Source

The fission neutron source from subcritical multiplication calculation using RAPID is presented in Fig. 7. It indicates that the fission neutron source drops one order of magnitude from center of the cask to the edge of the cask.

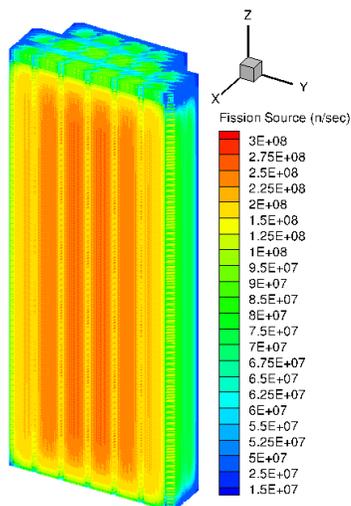
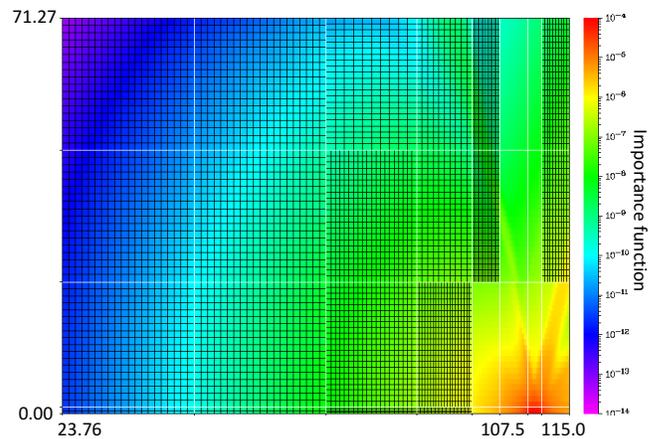
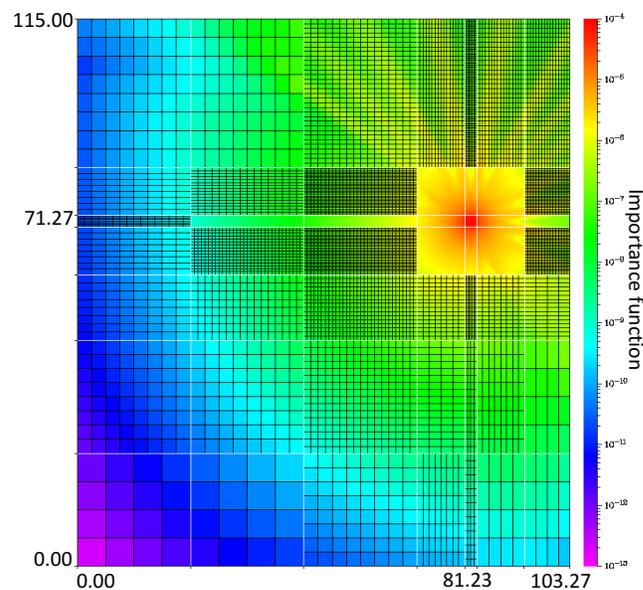


Fig. 7: Fission Source of 50 GWD/MTU no cooling FAs Radial Half Cask (neutrons/sec)



(a) 0.00° dosimeter.



(b) 41.28° dosimeter.

Fig. 8: Importance function of Group 8 in BUGLE96 (4.966 MeV to 6.605 MeV).

To examine the effect of the angular quadrature, we have examined different quadrature orders including S8 and S10 for the 0.00° dosimeter, and S8, S12, and S16 for the 41.28° dosimeter. Figs. 8 (a) and 8 (b) show the adjoint function distribution for a fast group (4.966 MeV to 6.605 MeV) with S8 angular quadrature for the 0.00° and 41.28° dosimeters, respectively. Both diagrams show the ray effects that attributed to the use of limited number of directions.

For the fission source, we use the Watt fission spectrum [9] to generate 47-group spectra for the fresh (pure Uranium-235) and burnt fuels. Fig. 9 compares the two spectra. The burnt-spectra is somewhat harder because of the presence of the Plutonium-239.

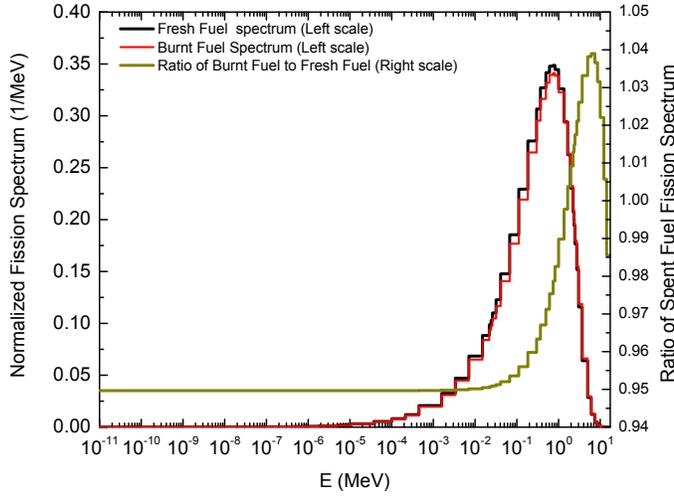


Fig. 9: Fission Spectrum

## 2. Dose Calculations

Using the fission source and adjoint function in Eq. 2, RAPID determines the dose at different locations. Using models shown in Figs. 4 and 5, reference fixed-source Monte Carlo calculations are performed. The results of the RAPID (for different quadrature orders) and MCNP calculations for the two dosimeters are presented in Tables I and II.

TABLE I: Dose rate of burnt fuel cask at the 0.00° dosimeter

Code	Quadrature	Dose rate (rem/hr)	Diff (%)
RAPID	S8	10.24	27.20
RAPID	S10	10.32	28.20
MCNP <sup>††</sup>	–	8.05±0.24 <sup>†</sup>	–

<sup>†</sup> 1  $\sigma$  <sup>††</sup> 3×10<sup>9</sup> particles

TABLE II: Dose rate of burnt fuel cask at the 41.28° dosimeter

Code	Quadrature	Dose rate (rem/hr)	Diff (%)
RAPID	S8	9.96	22.06
RAPID	S12	10.05	23.16
RAPID	S16	10.09	23.65
MCNP <sup>††</sup>	–	8.16±0.24 <sup>†</sup>	–

<sup>†</sup> 1  $\sigma$  <sup>††</sup> 3×10<sup>9</sup> particles

The above results indicate that an S8 quadrature set is adequate for these dosimeters. It is also demonstrated that for both dosimeter locations, the RAPID results are within 20-30 % of the MCNP predictions. This overestimation may be at-

tributed to the use of multigroup cross-section in the RAPID adjoint function calculation, versus the use of continuous-energy cross-section in the Monte Carlo calculation. As our previous studies on the reactor pressure vessel fluence calculation has demonstrated similar differences [13]. However, we are planning to examine this difference further.

The computer time for the pre-calculations and dose calculations are listed in the Tables III and IV. Note that for calculation of the FM coefficients and importance functions, we have used 16 and 8 processors, respectively. The RAPID dose calculation is performed on a personal computer with single processor, while the MCNP dose calculation is performed on a 16-processor cluster. Table III indicates that for this study, the computer wall-clock time for pre-calculations for the importance function with dosimeters located at 0.00° and 41.28° are about 12 hours and 29 hours on 8 processors, respectively. The higher computer time for the 41.28° dosimeter model is attributed to the fact that a significant higher number of meshes (by a factor 5) are used for this model.

TABLE III: Wall clock time for the RAPID pre-calculation

Dosimeter	# of Processors	Time (mins)
Multigroup TITAN importance function calculation		
0.00°	8	685.00
41.28°	8	1751.22 <sup>†</sup>
Fixed-source MCNP calculation for FM coefficient		
–	16	40.20

<sup>†</sup> Longer computer time is due to higher number of meshes

Table IV compares the wall-clock time for calculating the neutron dose at different locations using the RAPID and MCNP code systems.

TABLE IV: Comparison of wall-clock times for dose calculation using RAPID and MCNP

Dosimeter	# of Processors	Time (min)	Speedup
RAPID			
0.00°	1	7.62 <sup>††</sup>	145
41.28°	1	11.30 <sup>††</sup>	101
MCNP			
0.00°	16	1107.80	–
41.28°	16	1142.30	–

<sup>††</sup> This time include 3.7 min for the calculation of the fission source.

The MCNP calculations for both 0.00° and 41.28° dosimeter locations require about 19 hours on a 16-processors cluster. The calculation time of RAPID for 0.00° and 41.28° dosimeters are 7.62 and 11.30 minutes, respectively. The RAPID has speed up 145 and 101 times for 0.00° and 41.28° dosimeters, respectively. It should be mentioned that the fission source for MCNP calculation was obtained via RAPID subcritical multiplication calculation. The MCNP requires 12 days to converge on source calculation while it is only few minutes for RAPID calculation. This has been studied in our previous work [4]. It is important to note that after generating the database, RAPID can determine dose values at different locations and for different fuel burnups and cooling time in a few minutes as indicated in the Table IV.

### 3. Group-wise Contribution

Here, we examine the group-wise contribution of the dose by using Eq.4. Figure 10 presents the group-wise fractional dose contribution of two dosimeter locations. The 0.00° dosimeter shows a higher contribution of faster neutrons. This can be attributed to less moderation of neutrons as compared to that for the 41.28°. Also, it is demonstrated that only groups 1 to 19 contribute. This means that actually the computer time for the pre-calculation of the adjoint-function (given in Table III) can be reduced significantly.

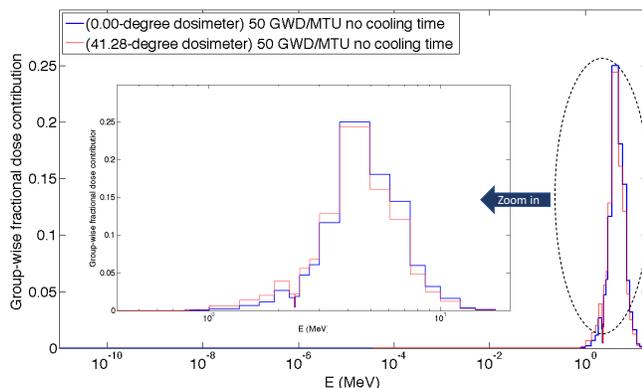


Fig. 10: Group-wise Contribution

### 4. Field of View study

The spatial FOV, given by Eq.5, are calculated based on the "Effective Distance", which is illustrated in the Fig. 11. The accumulated spatial dose value is calculated by backwardly tracing from dosimeter location. The FOV is shown in Fig. 12. It is obvious that the control volume is sufficient if the distance from the dosimeter location is about 72 cm.

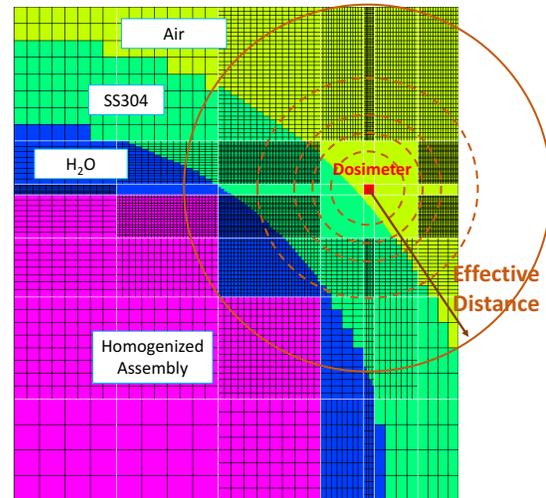


Fig. 11: Effective Distance

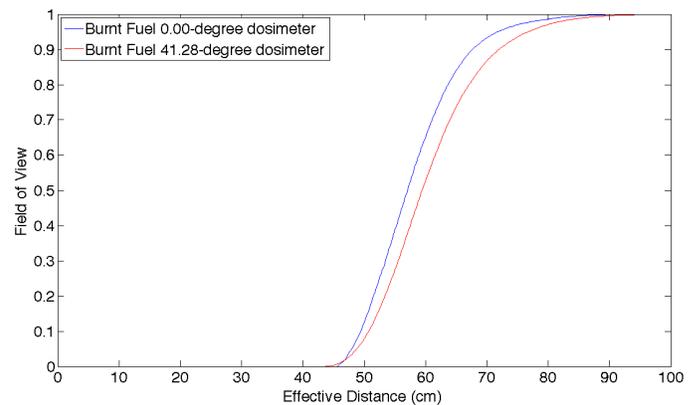


Fig. 12: Dosimeter FOV

## V. CONCLUSION

A new capability for determination of neutron/gamma dose based on the adjoint function methodology has been developed into the RAPID code system. The RAPID is capable to calculate radiation dose with about 10 minutes on a single-processor laptop while MCNP requires about 19 hours on a 16-processor cluster. A speedup about 100 to 150 is observed. A difference about 20% to 30% difference between RAPID and MCNP calculation is observed. This may be attributed to the effect of multigroup cross-section used on importance function calculation. Further investigation on this effect will be carefully analyzed.

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