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COMPARISON OF SHIELDING CALCULATION CODES FOR USED FUELS TRANSPORT/STORAGE CASKS: CASE STUDY WITH TK-26

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Abstract

For the design and license application for transport and storage casks, 2D calculation codes such as DORT, are commonly used in Japan. However, in order to model much more accurately the casks and its contents, and to reduce conservatism, more advanced computational technique*s,* such as 3D Monte Carlo codes, are considered. MCNP, becoming increasingly popular and accepted in the Japanese nuclear industry and the Japanese Nuclear Regulation Authority (NRA), is the 3D code chosen by Transnuclear, Ltd. for new shielding calculations.

Our last paper presented at the PATRAM 2016 studied the influence of these codes on a simplified model. This paper tries to expand on this research by using the actual design of a dual purpose (transport and storage) cask to confirm the advantages of using a very accurate model, made possible by using MCNP. The TK-26 cask, loaded with 26 used PWR fuel assemblies, is selected for this purpose.

The 2D code used in this study is DORT 3.2 (multigroup library: 200N-47G based on ENDF/B-VII.1 from the SCALE6.1 system) released by ORNL (Oak Ridge National Laboratory), and MCNP5 v1.60 (continuous energy library: ENDF/B-VII.0) released by LANL (Los Alamos National Laboratory). For the MCNP5 calculation, ADVANTG which is released by ORNL is applied to generate the effective automated variance reduction parameters to reduce the CPU time.

This paper discusses the following topics with respect to the shielding analysis of casks:

- theoretical differences between the DORT and MCNP codes,
- advantages, limitations and main assumptions of each code,
- geometry creation
- comparison of the dose rates calculated with each code

Even if the capabilities of the two calculation codes are vastly different, the assumptions and associated margins make them both suitable for shielding calculations. However, as the dose rates calculated by MCNP are basically more accurate, a shift towards MCNP is currently underway in our company. Future prospect of research would be the study of the individual impact of geometry modeling (accurate modeling vs homogenizing) and the cross-sections libraries (multigroup vs continuous energy) in more details.

Introduction

After the Great East Japan Earthquake of March 11th, 2011, Japanese regulations were tightened. Now the Japanese nuclear power plants are restarting one by one, 9 are already in operation and 16 others applied for a restart to the Japanese Nuclear Regulation Authority (NRA). However the nuclear power plants pools are almost full, with this ever increasing activity, storage and transport solutions are needed. Dry storage and transport is the most important part of the Japanese approach to used fuel management. To ensure the safety of operators, radiation protection is tightly regulated by the NRA. In order to design a transport and storage cask such as the TK-26, shown in Figure 1, as efficiently and safely as possible, both DORT [1] and MCNP [2] to model casks and their content are used.

Transport and storage casks are designed to ensure that safety prevail during routine and accidental conditions of transport. To achieve this goal, mechanical, thermal, sub-critical, containment and radiation protection criteria needs to be fulfilled. The present works focus on one of the most critical area: radiation protection. This paper shows the procedures and approach used by Transnuclear, Ltd(TNT) to model casks as well as a comparison between the dose rate, around the TK-26 Dual Purpose Cask, calculated by 3D Monte Carlo code MCNP and the 2D deterministic calculation code DORT, expanding in a previous paper[3] that focused on classical benchmark problems [4].

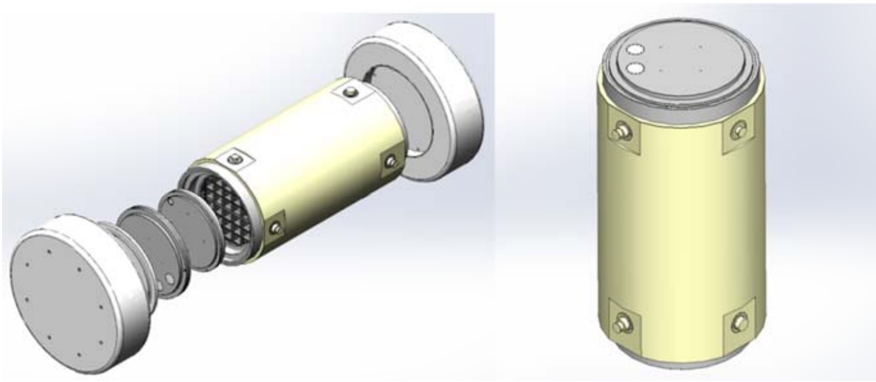

Figure 1: Transport (left) and Storage (right) configuration of the TK-26 cask

Theoretical differences between the DORT and MCNP codes

DORT main application is the calculation concerning deep penetration transport of both neutrons and photons. The fluxes of particles due to radiation sources, their interactions with the medium as well as incident particles from external sources are calculated and give the desired results in one or two dimensional geometric system and in a multigroup energy structure. The deterministic code DORT was developed based directly on the earlier DOT codes, by ORNL. To solve the Boltzmann transport equation, the method of discrete ordinates (Sn method) is used. For the discrete ordinates method, balance equations are solved for oriented flow of particles in each cell of a space mesh as well as in each energy group. Iterations are performed as needed, until all implicitness problems are solved (in the coupling of cells, the directions, or the energy groups).Depending on the complexity of the problem, running time can varies from few minutes to several hours.

MCNP is a Monte Carlo radiation-transport code designed to track various particle types over broad ranges of energies, using either continuous energy or multigroup energy structures. It is a time dependant general-purpose code, with the capacity to model in three dimensions. The Monte Carlo method relies on repeated random samplings, in MCNP case the particles are followed from birth to disappearance. The large quantity of particles generated leads to statistically significant numerical results. In this study ADVANTG is used, this tool produces automatically variance reduction parameters and source bias for MCNP calculations.

DORT and MCNP can be used for the same purpose: the design of dual purpose casks; however the theory behind them is quite different. MCNP is a 3D Monte Carlo code whereas DORT is a 2D calculation deterministic code, these differences implies the need to use a different set of assumptions. These assumption and there conservativeness can easily be assessed to prove the validity of using both codes.

The energy spectrum comes from ORIGEN2.2UPJ calculations in both cases. For DORT, to solve the Boltzmann equation, adjacent energies are grouped together and an average cross section is used, this can leads to problem due to resonance treatment. To mitigate this problem, TNT uses a 200 neutrons and 47 gammas groups energy structure [5] (ENDFB-VII library); the gamma spectrum is built from the ORIGEN original 18 groups thanks to the use of cubic splines. For MNCP the ORIGEN data can directly be used. Even if the original data is the same, the way it is treated differs.

MCNP can model casks in three dimensions while DORT can only accommodate two dimensions. Even as TORT would have been a natural progression, the switch to MCNP instead is due to a desire to keep up with the latest technologies. Trunnions, fuel and baskets as well as copper fins are difficult to model in 2D, homogenization or additional model are often required.

Advantages and limitations of each code

The deterministic code DORT and the Monte Carlo code MCNP both have advantages and disadvantages [6]. However for most of the problems encountered, ways to mitigate these issues were developed during the long life of these two codes. For MCNP, the use of a Monte Carlo method may create statistical problems for convergence and source scanning for instance; add-ons, plugins, tools or advanced features of MCNP, such as the ADVANTG[7] tool, can be used to solve or mitigate these problems. The advantages and limitations of each code can be found in the Table 1.

Table 1: Advantages and disadvantages of each code

Item	Monte Carlo method MCNP	Deterministic method DORT
1.Flux and convergence	(+) True value of the flux is calculated, only limited by convergence and cross section data	(+) Depends on the choice of parameter (Sn order, epsilon, iterations, group energy among others)
2.Calculation number and CPU time	(-) Only one model per particle is needed for each case but it is quite time expensive to get a suitable convergence	(-) Each calculation is relatively short but several models are needed
3.Source specification	(-) Spread source areas may cause statistical problems (Scanning). The use of ADVANTG mitigates this problem well	(+) Spread source areas causes no problems
4. Source energy spectrum	(+-) Energetically wide-ranged sources may cause statistical problems. The use of ADVANTG or/and the use of experience to give a defined structure to the source spectrum mitigates this problem well	(+/-) arbitrary energy ranges of source terms can be handled however too wide energy groups may cause problems (rays of cobalt), to mitigate this problem a 200n-47g energy groups structure was used for the TK-26
5. Radiation transport in vacuum/air	(++) No problem	(-) Ray-effect is cause of problems

6. Radiation transport through complex geometry	(-) Statistical problems may occur and additional calculations could be necessary. The use of ADVANTG mitigates this problem well	(-) Problems due to mesh definition as well as ray effect
7. Radiation transport for deep penetration	(+-) Absorbent material may cause Importance problems may arise however the use of ADVANTG mitigates these problems well	(+ +) No problem
8. 3D geometries description	(++) Any 3D geometries can be modeled	(--) Additional calculations are necessary to model 3D geometries, homogeneous basket, fuel and copper fin (generally where streaming occurs)
9. Cross-sections	(++) No problem due to the use of a continuous energy data library	(-) Multi-group energy spectrum can cause problems but it is heavily mitigated by the use of 200n-47g energy groups structure

Geometry creation

Japanese regulation and IAEA recommendation concerning radiation protection are different. Japanese regulation is stricter than the recommendation of the IAEA in term of radiation protection. For the routine transport configuration, the criteria in Japan are 100µSv/h at one meter from the surface of the cask and 2000µSv/h at its surface.

In DORT, defining the geometry and material is a three step process in our method while MCNP is more straightforward.

The trunnion is the most difficult part to model, in order to achieve this feat in the two dimensional code DORT, several models are defined. The models serve to characterize the dose rate around the whole cask. Geometries based on vertical slices of the casks, slices where trunnions are and where they are not are defined for each configuration (routine, normal and accidental transport as well as storage).

First step: Determination of homogeneous material densities

- Detailed R-Theta models are created. These models (such as the one in Figure 2) are used as reference
- One dimensional models representative of the center of the various angles are created
- The one dimensional models material atomic composition is modified in order to obtain a dose rate exceeding the detailed model
- These material densities will be used in the next step

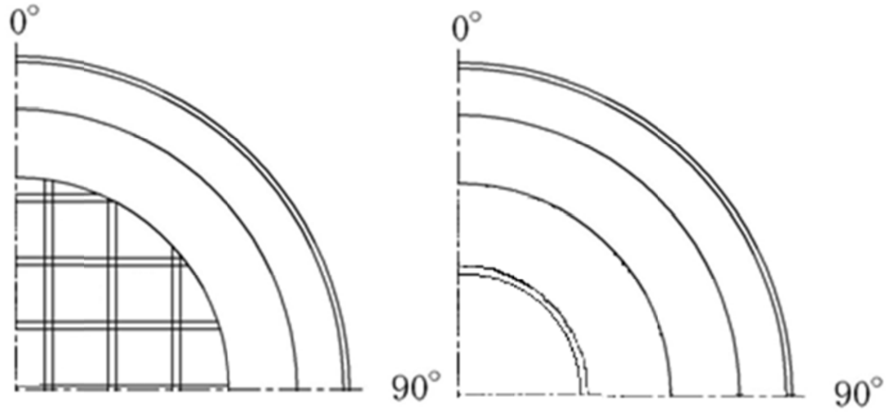

Figure 2: Plan (left), Detail R-Theta model with equivalent material density definition (right)

<u>Second step: R-Z model of the cask</u>

- Similarly to step one, several model of the cask are created for the various angles
- The atomic densities defined in the previous steps are used in the corresponding angle models

The main assumptions are:

- The cask is symmetrical by rotation
- Major holes, screw holes and tightening system elements removing shielding are assumed to have a rotational symmetry
- No models include trunnions, as shown in Figure 3; they will be added in the next step

To determine the radiuses of the basket areas, the volumes of fuel assemblies and basket are evaluated and radiuses corresponding to similar volumes are defined. All the atomic elements quantities are evaluated and used to define homogeneous areas. The need to create these homogeneous areas is one of the limitations of the 2D codes and a reason that TNT is currently evolving to use 3D simulation codes.

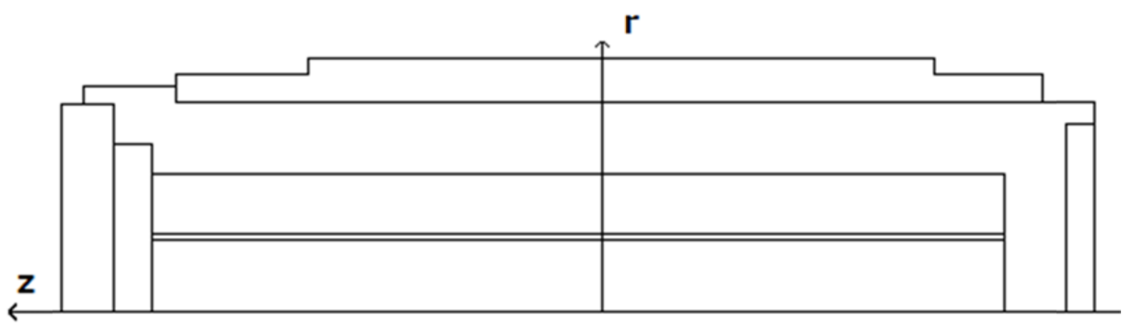

Figure 3: DORT full cask R-Z model Trunnion

<u>Third step: Streaming effect at the trunnions</u>

- Trunnions models corresponding to each step 2 R-Z models are made

- Two models are created for each cases: one flat model without trunnion and one trunnion model
- The positive values of the dose rate differences between the trunnion and flat models are calculated
- The dose rate obtained is added to the dose rate values of the step 2 R-Z model. Doing so, streaming effect on the trunnion areas can be considered in DORT

The assumptions taken are the following:

- Flux taken from adequately chosen meshed of the R-Z full cask model correspond to the flux in the first meshes of the R-Z trunnion model
- The trunnion can be modeled using a R-Z model centered on trunnion axis

As MCNP code can model three dimensional geometries, the model geometry and materials are based on the real geometry and materials of the cask, as shown in Figure 4. Contrary to the DORT modeling, conservatisms caused by restrictive hypothesis to tie together each step different models do not exist.

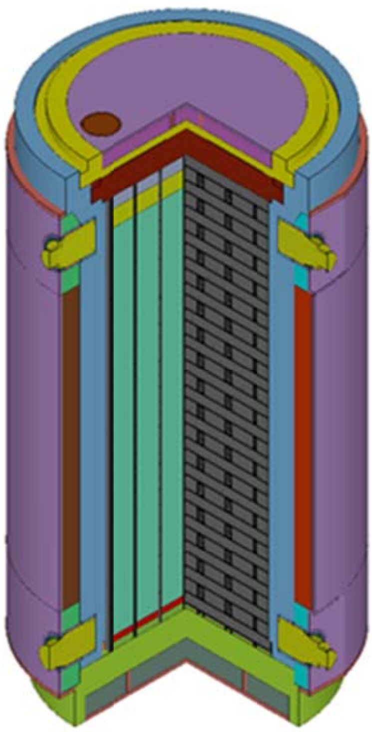

Figure 4: MCNP geometry of the TK-26 cask

Comparison of the dose rates around the TK-26 calculated by DORT and MCNP

Using MCNP, the local dose rate (among others) can be calculated at point of interest thank to the use of suitably positioned tallies. For DORT, the dose rate is calculated in two steps, in the first step, the DORT calculation provides neutron and gamma fluxes data. Based on these data, the dose rate will be determined in a second step, thanks to the use of tools such as FALSFT[8] (at 1m).

The calculation workflow is described below in Figure 5; of course the shielding model, material and sources will be different according to the code used.

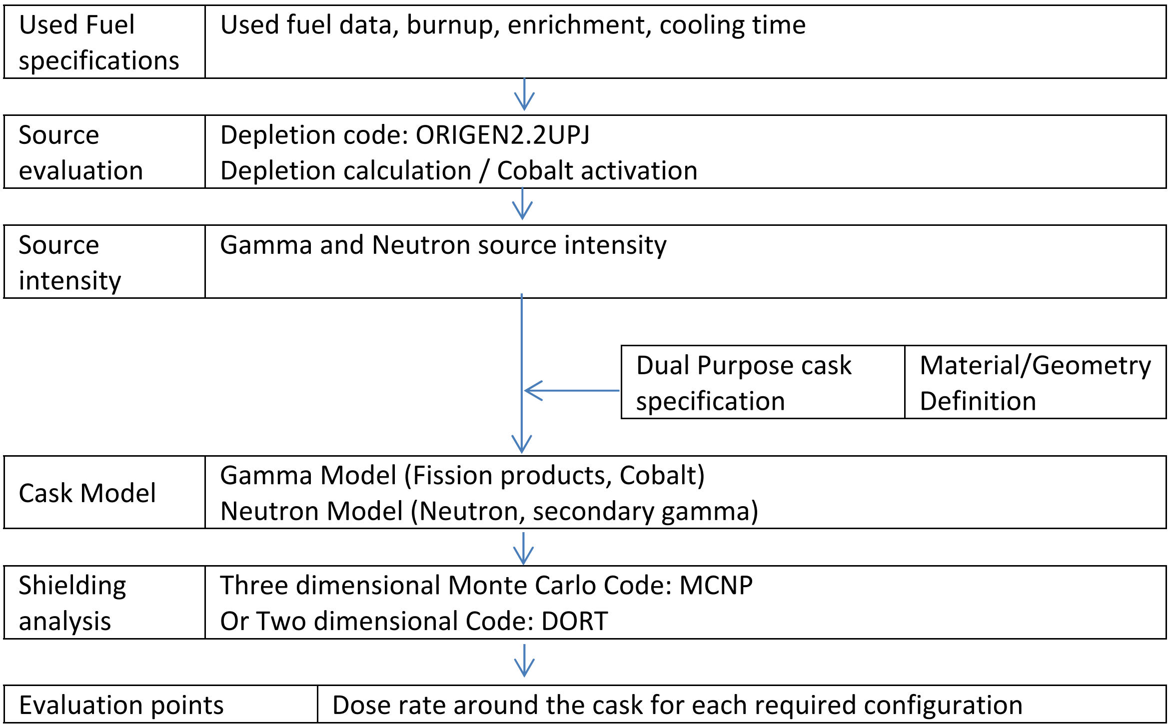

Figure 5: Flow of the shielding calculation

In order to compare DORT and MCNP, MCNP results will be taken as reference and the point with the maximum dose at surface as well as the closest trunnion area will be studied in further details, both at surface and at 1m (NRA criteria).

The dose rate can be found in the Table 2 for results at the surface and Table 3 for results at 1m from the surface of the cask. Explanation on the position of the calculated dose can be found in Figure 6; as per recommendation of the MCNP user's manual, a target of a maximum 10% statistical error at tallies is set. For the TK-26, the maximum dose rate for routine conditions of transport can be found at the surface perpendicular to the cask axis on the bottom side of the cask.

Neutron and gamma dose rate calculated by DORT are higher thanks to the conservative hypothesis explained above. However, at 1m, for neutron, in some cases DORT is below MCNP, this can be explained by the oversimplified cask model, especially concerning the shape of the basket as well as ray effect, even after mitigation efforts. Furthermore as the results are statistically close to each other, less than 10% in this case and statistical variation and errors can be considered as a potential cause. For the trunnion part, hypothesises on the flux made for DORT calculations are overly conservative. Secondary gamma dose rates, which contribution is minimal (less than 10%), are lower; the cause is under further investigation. DORT is generally more conservative than MCNP.

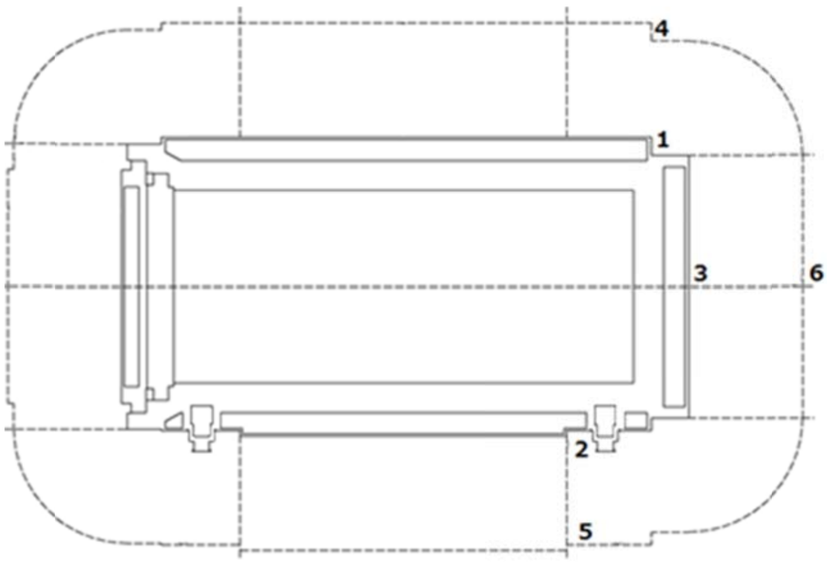

Figure 6: Position of the dose calculated

Table 2: Dose rate at the cask surface

Dose rate in µSv/h	1) Lower side part		2) Lower trunnion		3) Bottom part	
	MCNP	DORT	MCNP	DORT	MCNP	DORT
FP	2.6	3.9	26.0	69.9	19.8	17.7
Co	8.7	7.9	11.9	28.7	86.1	122.6
Secondary gamma	7.9	5.5	19.9	13.9	14.7	8.4
Neutron	1177.1	1231.0	381.3	537.1	55.1	51.0
Total	1196.3	1248.3	439.1	649.5	175.7	199.7
Ratio DORT/MCNP	1) Lower side part		2) Lower trunnion		3) Bottom part	
Fission products	1.51		2.69		0.89	
Co	0.91		2.41		1.42	
Secondary Gamma	0.70		0.70		0.57	
Neutron	1.05		1.41		0.93	
Total	1.04		1.48		1.14	

Table 3: Dose rate at 1m from the cask surface

Dose rate in µSv/h	4) Lower side part		5) Lower trunnion		6) Bottom part	
	MCNP	DORT	MCNP	DORT	MCNP	DORT
FP	6.5	7.3	11.9	20.1	8.4	8.0
Co	9.0	11.9	8.7	19.5	38.9	48.2
Secondary gamma	3.1	2.5	7.2	4.8	4.7	2.9
Neutron	56.2	46.3	28.7	32.5	21.8	20.8
Total	74.8	68.0	56.5	76.9	73.8	79.9
Ratio DORT/MCNP	4) Lower side part		5) Lower trunnion		6) Bottom part	
Fission products	1.13		1.70		0.95	
Co	1.33		2.26		1.24	
Secondary Gamma	0.81		0.67		0.62	
Neutron	0.82		1.13		0.95	
Total	0.91		1.36		1.08	

Conclusion

DORT is good for dual purpose cask calculation. MCNP is better. The theory behind MCNP allows to get rid of the conservationism present when using the DORT code. MCNP models, closer to the reality allow to develop high performance casks. As TNT is the first company to propose a full MCNP Safety Analysis Report to the NRA, it is pioneering the cask industry in Japan.

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