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BENCHMARK CALCULATION OF CANDU END SHIELDING SYSTEM

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ABSTRACT

A shielding analysis was performed for the end shield of CANDU 6 reactor. The one-dimensional discrete ordinate code ANISN with a 38-group neutron-gamma library, extracted from DLC-37D library, was used to estimate the dose rate for the natural uranium CANDU reactor. For comparison, MCNP-4B calculation was performed for the same system using continuous, discrete and multi-group libraries. The comparison has shown that the total dose rate of the ANISN calculation agrees well with that of the MCNP calculation. However, the individual dose rate (neutron and gamma) has shown opposite trends between ANISN and MCNP estimates, which may require a consistent library generation for both codes.

INTRODUCTION

In the direct use of spent PWR fuel in CANDU (DUPIC)[1] fuel cycle, the spent PWR fuel is reused in the existing CANDU reactor. For the DUPIC fuel CANDU core, it was recommended to adopt a 2-bundle shift refueling scheme, which in turn results in a different power distribution from the conventional natural uranium core. In general, the power shape is more flattened as the number of fuel bundles shifted per refueling operation is reduced. The flatter power distribution implies steeper flux gradients towards the ends and the radial edge of the core. When the flux gradient is steeper, the neutron leakage increases both axially and radially, which results in higher radiation fluxes throughout the end-shield and, consequently, the dose rates outside the end-shield can be increased.

The primary shield analysis for CANDU 6 reactor has been performed by the discrete ordinates and the point kernel method. Recently, the Monte Carlo method is used for the limited cases such as the fuel channel analysis and the radiation penetration calculation through the top shield. In this study, we have analyzed the end shield system by the ANISN code using conventional 38-group neutron-gamma library generated from DLC-37D[2] and compared the result to Monte Carlo calculation.

CALCULATION MODEL

Geometry

The end shields are horizontally oriented cylinders at each end of the reactor as shown in Fig.1. They consist basically of three zones which are the calandria side tube sheet, the carbon steel balls in light water region, and the fueling machine side tube sheet. The overall thickness of the end shield is 91.44 cm and the tube sheets are made of stainless steel 304L. Outside the tube region, there is a concrete wall of 106.68 cm thick. The air space of 1600 cm thick between the fueling machine side tube and the concrete wall is not modeled explicitly in this calculation, but the dose rate can be calibrated by an attenuation factor when the actual dose rate outside the concrete is needed.

Source Term

The axial fission density distribution on the horizontal mid-plane of the reactor was obtained from the time-average calculation[3]. The normalized fission density at the core center is 7.797×10^{11} fissions/cm^3sec or 723.125 kWth/bundle. The source distribution was calculated by interpolating fission density distribution in the core.

ANISN Model

In order to calculate neutron flux, gamma and neutron dose distribution through the end shield of CANDU 6 reactor, a one-dimensional discrete ordinates transport code ANISN was used for the slab geometry. The number of meshes and the dimension of each region used in this calculation are given in Table I and the atomic density of the each region is given in Table II. The numbers of neutron and delayed gamma per fission are 2.62 and 6.35, respectively. The orders of scattering and angular quadrature chosen for the transport calculation are P3 and S16, respectively. A 38-group neutron-gamma-coupled cross-section library was used for ANISN calculation. The neutron energy group consists of 7 fast groups (En > 0.82 MeV), 19 intermediate groups (0.82 MeV > En > 0.414 eV) and 1 thermal group (En < 0.414 eV).

MCNP Model

In order to validate the ANISN calculation, a Monte Carlo code MCNP-4B was used. In order to keep the consistency with the ANISN calculation, only one-dimensional model was developed.

For MCNP model of delayed gamma, it was assumed that the amount of delayed gamma leakage through the end shield is negligible. For the variance reduction in MCNP calculation, the population control methods like the particle splitting and Russian roulette are typically applied. However, for the calculation with a continuous energy library, the weight cutoff option (CUT) was also used in order to improve the convergency.

For the cross-section libraries of MCNP, the multi-group library MGXSNP, the discrete library DRMCCS, and the continuous library RMCCS were used to see the sensitivity of group structure. The MGXSNP library is comprised of 30-group neutron and 12-group photon data primarily based on ENDF/B-V for 95 nuclides. The multi-group data library was produced from MENDF5 and MENDF5G by CRSRD code. The DRMCCS is a discrete library with 262 energy groups based on ENDF/B-V. The continuous library RMCCS is based on ENDF/B-V too and MCPLIB02 is used for gamma calculation.

RESULTS, CONCLUSION AND RECOMMENDATIONS

The total dose estimated by the ANISN is a little higher than that by the MCNP as shown in Fig 2. Considering the standard deviation of MCNP calculation, the total dose rate predicted by ANISN with DLC-37D library is in a good agreement with that by MCNP with a little conservatism. However, if we look at the individual dose rate such as neutron and gamma, the neutron dose is higher for the ANISN calculation while the trend is opposite for the gamma dose as shown in Figs. 3 and 4. Therefore, the good agreement of total dose estimation between ANISN and MCNP is actually a compensation effect of the discrepancy of individual dose rate.

In order to investigate more, the ANISN and MCNP were run again using a cross-section data consistently generated for both codes. The BUGLE96[4] was used to generate a multi-group library SNXS67M for ANISN and MCNP through CRSRD code. The BUGLE96 was originally developed for the light water reactor shielding and reactor pressure vessel dosimetry applications based on ENDF/B-VI. However, for the purpose of comparing two codes, the source of the library won't be a problem once they are generated consistently. As a result, the neutron flux distributions obtained with BUGLE96 library are in an excellent agreement between two codes. However, the total dose rate of ANISN calculation is slightly higher than that of MCNP calculation as shown in Fig.5. It is now clear that the difference of neutron dose rate between two codes comes from the cross-section data not from the solution method.

This study has shown that the conventional shielding analysis by ANISN with DLC-37D reasonably estimates the total dose of CANDU 6 end shield system. However, in order to have an improved result for individual dose rate, it is recommended to generate the cross-section data consistently with the reference model which is the MCNP calculation with the continuous energy library in this study.

REFERENCES

1. J.S. Lee et al., "Research and Development Program of KAERI for DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors)," Int. Conf. and Tech. Exhibition on Future Nuclear System: Emerging Fuel Cycles and Waste Disposal Options, GLOBAL'93, Seattle, USA, 1993.
2. W.E. Ford, "Coupled 100 Neutrons-21 Gamma Ray Group, P_8 Cross Section Library for EPR," Oak Ridge National Laboratory, ORNL/TM-5249, 1976.
3. Korea Electric Power Cooperation, "Final Safety Analysis Report : Wolsong NPP units no.2/3/4 Vol.3," Korea Electric Power Cooperation, 1995.
4. "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Oak Ridge National Laboratory, RSIC-DLC-185, 1996.

Table I. Region Dimensions and Number of Meshes for End Shield Calculation

Region	Region ID	Number of Meshes	Thickness (cm)	Rad. Dist. from Center (cm)
1	Core	85	297.18	297.18
2	Calandria Side Tube Sheet	10	5.08	302.26
3	Carbon Steel Ball & Water	90	78.74	381.00
4	Fuelling Machine Side Tube Sheet	15	7.62	388.62
5	Concrete Wall(I)	20	30.48	419.10
6	Concrete Wall(II)	50	76.20	495.30
Total		270		

Table II. Atom Densities for Materials used in End Shield Calculation

No.	Region ID	Element	Atomic Density (atoms/b.cm)	Element	Atomic Density (atoms/b.cm)
1	Core	H	1.097E-04	Zr	1.443E-03
		D	5.638E-02	U-235	8.605E-06
		O	2.825E-02	U-238	1.187E-03
2	Stainless Steel 304L (ρ=7.9g/cm^3)	C	1.387E-04	Mn	1.732E-03
		Si	1.271E-03	Fe	5.812E-02
		Cr	1.734E-02	Ni	8.107E-03
3	Carbon Steel Ball/H_2O (60/40 region)	H	2.674E-02	Si	2.525E-04
		C	7.794E-04	Mn	5.010E-04
		O	1.337E-02	Fe	5.002E-02
4	Ordinary Concrete (ρ=3.36g/cm^3)	H	9.583E-03	Al	1.534E-04
		C	1.143E-02	Si	1.783E-03
		O	4.531E-02	Ca	7.498E-03
		Mg	6.018E-03	Fe	1.112E-04

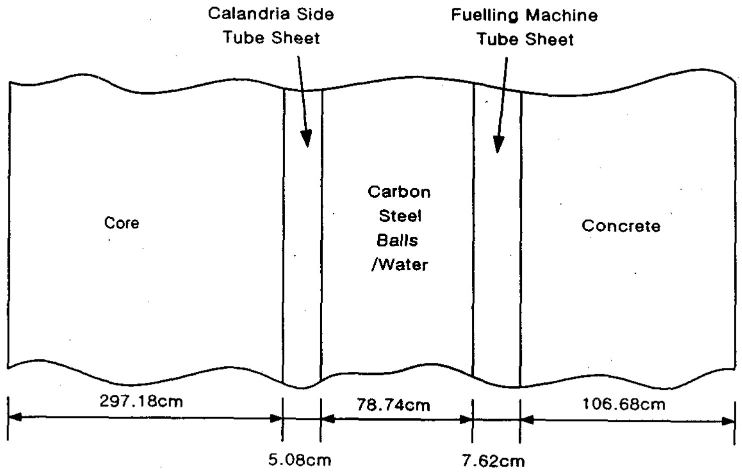

Fig.1. One-dimensional Model of End Shield System (No Scale)

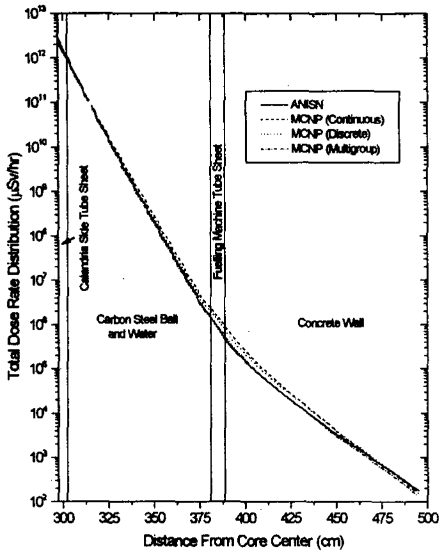

Fig.2 Total dose rate through the End Shield

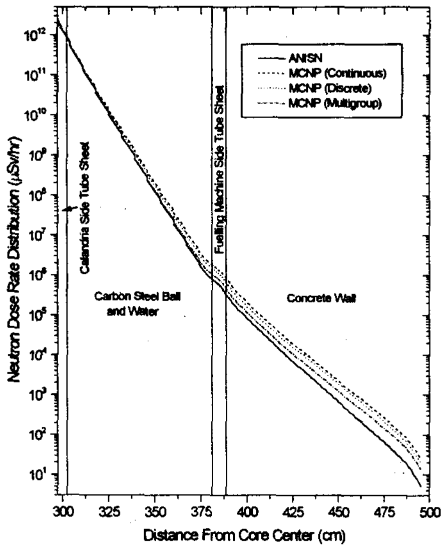

Fig.3 Neutron dose rate through the End Shield

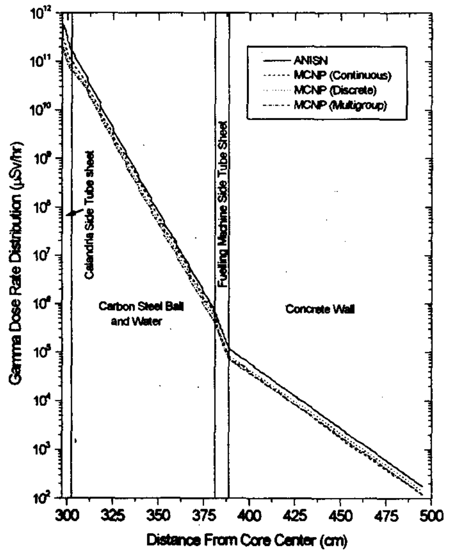

Fig.4 Gamma dose rate through the End Shield

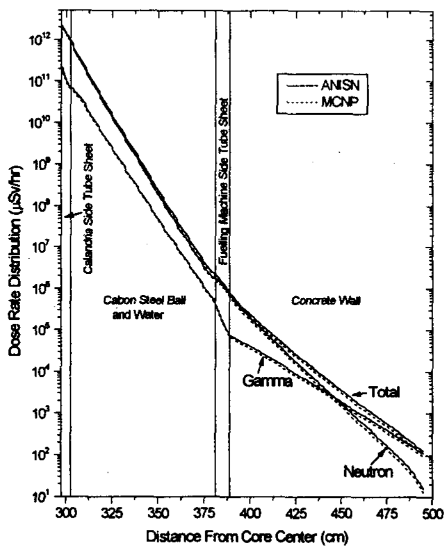

Fig.5 Total dose rates by bugle96