TRANSPORT-BURNUP CODE SYSTEMS AND THEIR APPLICATION TO IAEA ADS BENCHMARK

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ABSTRACT

Recently for transmutation of spent nuclear fuel, the Accelerator Driven System (ADS) is introduced. Because of anisotropic flux distribution caused by external source, the nuclear calculation of ADS is significantly different with the traditional nuclear reactor. In this paper, two transport and depletion code systems — MCNT-ORIGEN2 and ANISN -DOT4.2-ORIGEN2 are developed and are applied to IAEA ADS benchmark. Initial enrichments of ^{233}U which correspond to given initial K_{eff} values are calculated, and the spatial distributions of power density at BOL, the void reactivity effects, the external source effectiveness and the evolution curves of K_{eff} and external source as function of time are also calculated for each given K_{eff} (BOL). The numerical results are in a good agreement with those of other participants.

1. INTRODUCTION

The accelerator driven subcritical system (ADS) is possibly to be an advanced clean and safety nuclear system in the future. It has some special characteristics compared with the traditional reactor:(1) It is a subcritical system driven by an external source, which leads to the significant flux gradient and the serious anisotropic flux distribution; (2) The neutron energy varies from 0 to 20 Mev, which is more wider than the traditional reactor's. For validation of the nuclear libraries and the calculation methods used in ADS, IAEA launched the Co-ordinated Research Program (CRP) and many countries participated in it. In 1997, CRP issued an ADS benchmark. The benchmark was divided into several stages, and the 1st stage devoted to analyzing neutron performances of ADS and validating the nuclear databases and codes used for computation of ADS[1].

Because of existence of the external source, ADS's neutronics is distinctly different with the traditional nuclear reactor. For the latter, the external source must be excluded when the system is in equilibrium, and the neutron transport equation is homogeneous. The system's critical characteristic can be expressed by the equation's eigenvalue $1/k_{eff}$, and the flux distribution is relevant to the fundamental mode of the eigenfunction. But for ADS, the flux distribution is connected with the external source. The multiplication factor K_S is introduced, which can be calculated as follows:

$$\frac{1}{K_S} = 1 + \frac{< S >}{< M\Phi >} = 1 + \frac{S_0}{w\bar{\upsilon}} \tag{1}$$

where S_0 is total neutron production rate, w is the blanket power (fissions per time unit). $\bar{\upsilon}$ is the averaged secondary neutrons number. Obviously, K_S is different with K_{eff}. K_{eff} is determined by multiplication property of reactor. But K_S depends on both the multiplication properties and the external source quality (neutron spectrum, spatial location, etc.) So, there are two versions to evaluate ADS neutronics: " K_{eff} based version" and " K_s based version". K_{eff} is involved in the nuclear safe analysis, and K_S is concerned with the effectiveness of external source which can be expressed:

$$\varphi^* = \frac{\Phi_s^*}{\overline{\Phi}_F^*} = \frac{1 - 1/K_{eff}}{1 - 1/K_s} \tag{2}$$

where $\overline{\Phi}_S^*$ is the averaged importance of external neutron source, and $\overline{\Phi}_F^*$ is the averaged importance of neutrons produced by fissions. K_{eff}, K_S and φ^* are taken as the principal parameters for ADS study.

2. COMPUTING METHOD AND CODE SYSTEMS

Based on the requirements of ADS nuclear calculation, two code systems are developed in this paper: Monte Carlo-burnup code system MCNT-ORIGEN2 and deterministic transport-burnup code system ANISN-DOT4.2-ORIGEN2.

(1) Monte Carlo-burnup code system MCNT-ORIGEN2

In the code system MCNT performs Monte-Carlo simulation of neutron transport. The fission product and actinide generation in ADS is simulated using the burnup code ORIGEN2. [2]

MCNT is a Monte Carlo neutron transport code, which has a large set of functional and geometry options like MCNP. It can be run in two modes. The K_{eff}-value of the system is determined by running it in the KCODE mode, and K_S and spatial power distributions can be obtained by running it in the standard mode.

MCNT contains a nuclear data library of continuous energy cross section sets for about 200 nuclides including actinides, fission products and light elements. The cross sections are processed with NJOY from ENDF/B-VI data flies. The several sets of cross sections for the relevant temperature (300K, 500K, 900K, 1200K) for U233, Pu239 and Th232, etc., fissile materials are calculated with NJOY.

OROGEN2 has been used extensively for fuel management studies for fission reactor. It is

one-group, point-depletion code. One-group cross sections are acquired from predetermined reactor specific libraries. Current version of ORIGEN2 contains libraries for US PWRs, BWRs, liquid metal fast breeder reactor(LMFBR), etc.. But it has not yet the data of ADS. In this paper it is proposed that effective one-group cross sections in the core regions should be calculated with MCNT for 31actinides (U233, U235, Pu239, Np239, Am242, etc.) and some fission products at BOL and given enrichments. After MC simulation, one-group cross sections are generated by collapsing reaction rates from MCNT. Thus separate sets are obtained for the different region of the core. The BOL values are assumed to be valid throughout the entire burnup. ORIGEN2 makes use of one-group cross sections of the rest isotopes, fission product yields and decay constants from the specified LMFBR library of ORIGEN2.

MCNT is linked to ORIGEN2 to form an integrated code system for Monte-Carlo and burnup simulation. The neutron flux or power distribution in each spatial sub-region was obtained as MCNT simulation was executed. The isotopic composition of the burned fuel was calculated by ORIGEN2 based on burnup provided by MCNT. The ORIGEN2 output was automatically transcript into the material specification part of MCNT input by a script written.

(2) Transport-burnup code system ANISN-DOT4.2-ORIGEN2

This code system is based on the deterministic method. Two-dimensional neutron transport simulation is performed by a multi-group finite difference two-dimensional S_N code DOT4.2[6]. In order to save computing time, few-group approximation is used in the code system. A few-group cross section library as function of burnup is generated by ANISN[3], which is a one-dimensional discrete ordinates transport code with anisotropic scattering. ORIGEN2 simulates the fission product and actinide generation.

For generating the few-group cross sections library as function of burnup, ANISN performs one-dimensional transport calculations and obtains multi-group neutron flux distribution in each region. Then, separate sets of few-group cross sections for different region are obtained by collapsing the multi-group cross sections weighted with spectrum from ANISN. At the same time the new burnup of each fuel region are also updated at current step and transited to the depletion code ORIGEN2, which calculates the new isotopic compositions of each fissile region. For simply linking between ANISN and ORIGEN2, a script that automatically transits the ORIGEN2 output into the material specification part of ANISN input is written. After such repeated calculations, separate sets of few-group cross sections libraries as function of burnup are obtained.

At each burnup step, DOT4.2 performs two dimensional few-group transport calculation and the spatial neutron flux distribution are obtained. According to the burnup in each sub-region, the new few-group constants can be generated from the few-group cross sections library mentioned above. These procedure are similar to the usual depletion calculation in PWR

reactor.

The transport nuclear library for ANISN code can be obtained using NJOY nuclear data processing code with JENDL and ENDF data files. In this paper, the expanded BISON-C[7] code's neutron transport library is used. The effective one-group cross sections used in ORIGEN2 is generated in the same way as in MCNT-ORIGEN2 code system.

3.CALCULATION OF IAEA ADS BENCHMARK

The detail description of IAEA benchmark is given in reference [1]. The geometry is a cylindrically symmetric r-z representation. The U233 and Th232 are the fissile and fertile materials respectively. The dimension of target, lead reflector and the volume ratios of materials are also given in reference [1].

In MCNT-ORIGEN2 code system, the values of K_{eff} are obtained by running MCNT in the KCODE mode. The number of KCODE cycles was 100, and 5,000 source neutrons are tracked per cycle. Experience indicated that K_{eff} values of the first 10-15 cycles were significantly deferent with the convergence value, and were slowly converged after some cycles. In order to improve the precision, K_{eff} values of the first 20 cycles are not used for calculating the averaged K_{eff}.

K_S, spatial neutron flux distribution and effectiveness of source are obtained by running MCNT in the standard mode. For the external neutron source, its position is sampled uniformly in source region (dia.20, hgt.50), located in the center of the reactor, and its energy is sampled from the given spallation neutron spectrum. 60,000 source neutrons are sufficient to achieve a small statistical error in K_S (about 0.0001).

In ANISN-DOT4.2-ORIGEN2 code system, the number of few-group is determined according to the multi-group neutron spectra and the experience from the fast reactor calculation. In this paper 18 energy groups and S_8/P_1 approximation is adopted in two-dimensional transport calculations.

Because of geometrical symmetry of problem, two dimensional transport calculation is only applied to semi-core by using reflective boundary condition. Total number of spatial meshes is 111 in radial direction and 97 in axial direction. The partition of three fissile regions is the same: 50 sub-regions for region 1, 60 sub-regions for region 2, and 30 sub-regions for region 3.

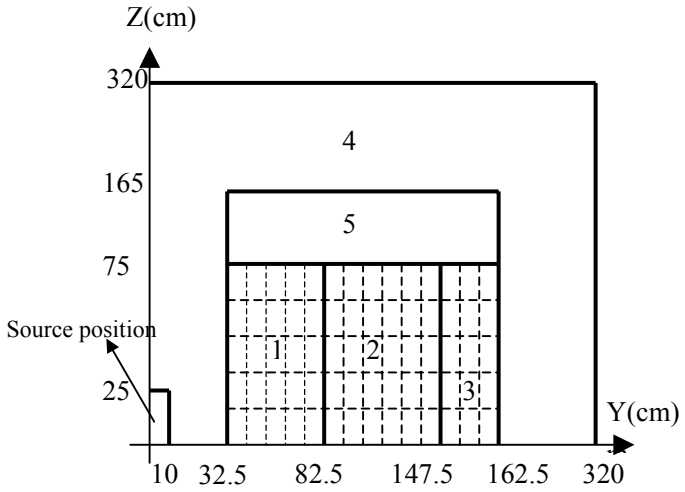

Figure1. partition of three fissile regions for burnup calculations

3.1 Initial ^{233}U enrichment in zones 1 , 2 corresponding to given BOL K_{eff}

Results of the enrichment calculation for given K_{eff} (0.98, 0.96, 0.94) are presented in Table I. "$E_{averaged}$" is the averaged value of all participants[1]. The difference of the results is provoked by the following factors: different neutron cross-section libraries and different calculation methods, such as Monte Carlo method, S_N method and diffusion approximation method.

Table Ⅰ. ^{233}U enrichment (E in %) in zones 1 , 2

Participants	K_{eff}=0.98	K_{eff}=0.96	K_{eff}=0.94
RUSSIA (Diffusion)	10.01	9.69	9.38
SWITZERLAND	9.88	9.57	9.25
ITALY	10.29	9.96	9.63
FRANCE	10.27	9.94	9.61
HETHERLANDS	10.13	9.81	9.49
JAPAN	9.7	9.4	9.1
BELARUS	10.50	10.17	9.85
KOREA	9.85	9.48	9.2
$E_{averaged}$	10.17	9.85	9.53
MCNT -ORIGEN2	9.99	9.672	9.953
ANISN-DOT4.2 -ORIGEN2	10.265	9.948	9.629

3.2 Multiplication factor K_S

The neutronics parameter K_S expresses the multiplication characteristic in the subcritical system. The results are listed in the Table II.

Table II. Multiplication factors K_S of given K_{eff}(BOL)

K_{eff}	ANISN-DOT4.2 -ORIGEN2		MCNT -ORIGEN2		SWEDEN[5]	
	C(^{233}U)	K_s	C(^{233}U)	K_s	C(^{233}U)	K_s
0.94	9.629%	0.9534	9.353%	0.954	9.771%	0.9504
0.96	9.948%	0.9699	9.672%	0.968	10.095%	0.9679
0.98	10.265%	0.9855	9.990%	0.984	10.419%	0.9854

3.3 Spallation source effectiveness

External neutron source effectiveness φ^* is rather high in ADS, and will decrease with burnup. Its' value at BOL is presented in Table III.

Table III. Spallation source importance of given K_{eff}(BOL)

Participants	$K_{eff,\ BOL}$=0.98	$K_{eff,\ BOL}$=0.96	$K_{eff,\ BOL}$=0.94
RUSSIA(Diffusion)	1.35	1.33	1.31
SWITZERLAND （JEF-2.2）	1.45	1.42	1.40
ITALY	1.31	1.31	1.29
FRANCE	1.30	1.27	1.25
NETHERLANDS	1.18	1.26	1.23
$E_{averaged}$	1.32	1.31	1.30
MCNT -ORIGEN2	1.28	1.27	1.34
ANISN-DOT4.2 -ORIGEN2	1.39	1.34	1.31

3.4 Void reactivity effects

The void reactivity effects play very important roles in accessing void power effects for ADS. It can be calculated as follow:

$$\delta K_{eff} / K_{eff} = \{K_{eff}(BOL, void) - K_{eff}(BOL)\} / K_{eff}(BOL)$$

Table IV. Void reactivity effects of given K_{eff}(BOL) in pcm

Participants	$K_{eff,BOL}$=0.98		$K_{eff,BOL}$=0.96		$K_{eff,BOL}$=0.94	
	Region 1 voided	Region 1+2 voided	Region 1 voided	Region 1+2 voided	Region 1 voided	Region 1+2 voided
RUSSIA (Diffusion)	+1183	+262	+1211	+302	+1242	+347
SWITZERLAND	+786	-530	+812	-500	+840	-457

（JEF2.2）						
ITALY	+863	-596	+892	-554	+924	-509
FRANCE	+998	-413	+1072	-386	+1152	-354
HETHERLAND	+625	-932	+811	-667	+637	-785
JAPAN	+3290	+4730	+3364	+4820	+3744	+4986
KOREA	+517	-364	+704	-1048	+1202	+178
$E_{averaged}$	+959	-220	+994	-250	+1023	-140
MCNT-ORIGEN2	+952	-653	+879	-551	+777	-592
ANISN-DOT4.2-ORIGEN2	+935	+94	+961	+129	+969	+180

3.5 Spatial distribution of power density (BOL)

Spatial distributions of power show a large deviation from “cos” or “Bessel” type distribution in a critical system due to the presence of sufficiently strong external source. Power peaking factor in ADS is also larger than a critical reactor’s.

(A). Radial distribution

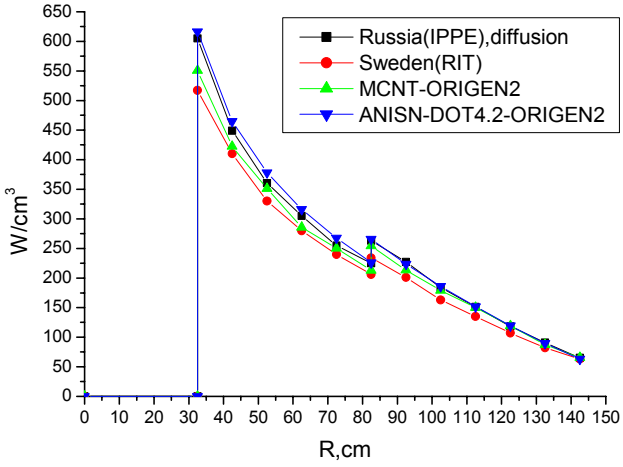

Figure 2. Radial core power distribution at z=0cm (K_{eff}(t=0)~0.94)

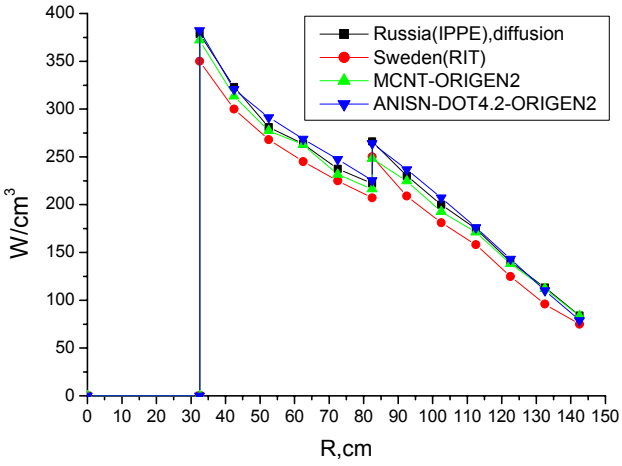

Figure3. Radial core power distribution at z=0cm(K_{eff}(t=0)~0.98)

(B). Axial distribution

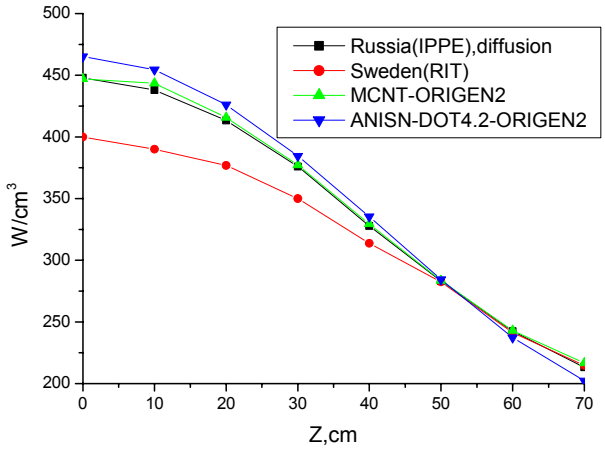

Figure 4. Axial power distribution at r=42.5cm(K_{eff}(t=0)~0.94)

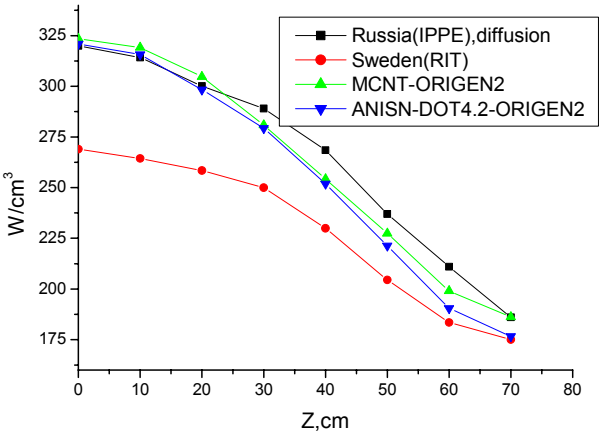

Figure 5. Axial power distribution at r=42.5cm(K_{eff}(t=0)~0.98)

3.6. Reactivity burnup swing

ADS is a reactor driven by a external neutron source, so its power depends strongly on the external neutron source density and K_{eff}. In the all lifetime, K_{eff} varies due to depletion of the fissile nuclides and accumulation of fission products, and external source intensity should be adjusted to maintain the given total power (1500GWt). The general trend of K_{eff} is on the contrary to external source intensity's (see figure 6,7,8 and 9).

(A). K_{eff} as function of time

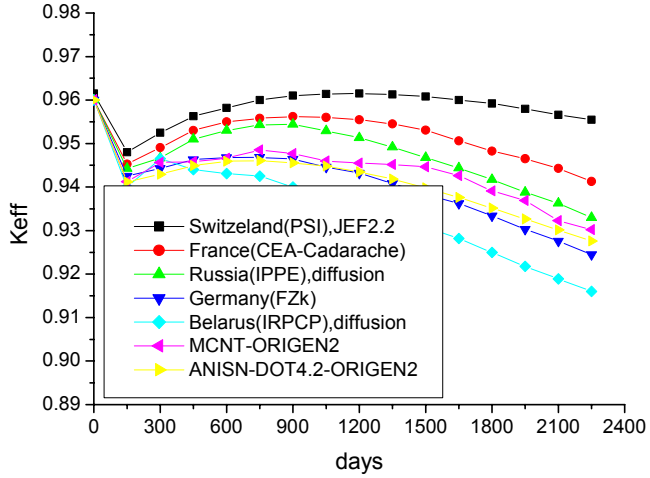

Figure 6. The evolution of K_{eff} as function of time (K_{eff}(t=0)~0.94)

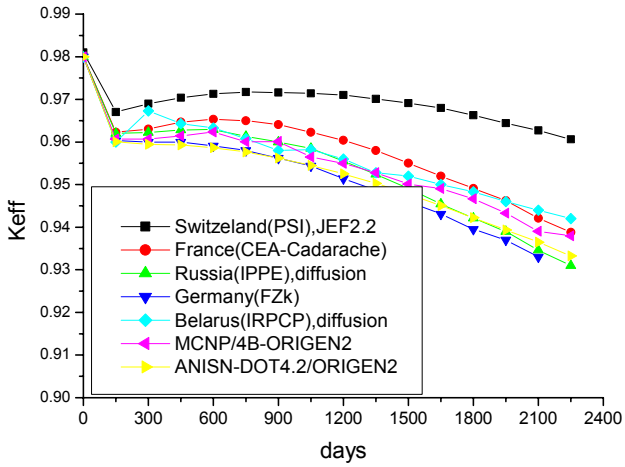

Figure 7. The evolution of K_{eff} as function of time (K_{eff}(t=0)~0.98)

(B). External source density as function of time

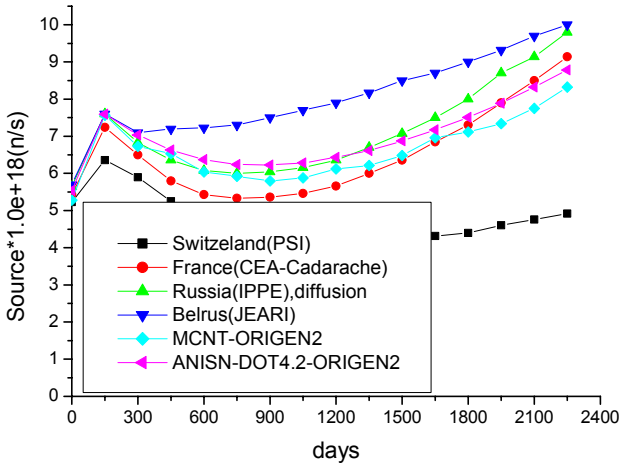

Figure 8. The evolution of Source as function of time(K_{eff}(t=0) ~0.94)

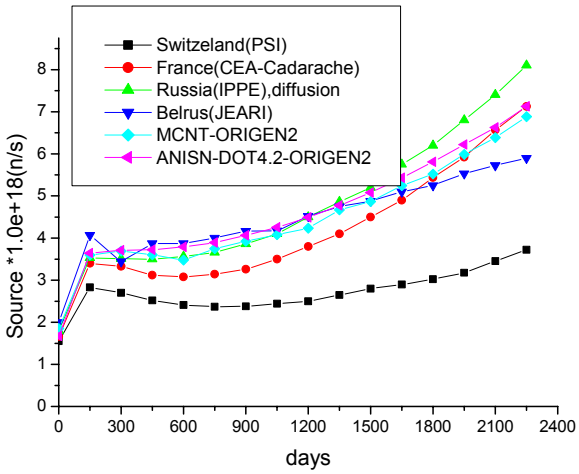

Figure 9. The evolution of Source as function of time(K_{eff}(t=0) ~0.98)

4. CONCLUSIONS

In this paper, MCNT-ORIGEN2 code system and ANISN-DOT4.2-ORIGEN2 code system are developed. They are applied to IAEA ADS benchmark. The numerical results are in a good agreement with those of other participants.

However, the rather large discrepancies in some results are displayed between some participants'. For example, the dispersion of initial enrichments for given K_{eff} can reach 2-4.5%. The main reasons for these discrepancies are considered due to the following factors: different neutron cross sections libraries (for example, JEF, JENDL, ENDF/B) and different calculation methods (Monte-Carlo method, deterministic S_N method and diffusion method) are used. This dispersion is acceptable for concept design stage and obviously should be largely reduced for engineering purposes.

So further study for ADS neutronics is necessary in many respects. Firstly, modification of existing nuclear data libraries, especially in high energy range (10-20Mev) is necessary and database for more actinides and fission products should be supplemented and reviewed. secondly, development of new methods and modification of existing code systems are indispensable for the improving the calculation precision and the further requirement of the development of ADS.

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