FAKIR 6.0: A PC CODE FOR RESIDUAL DECAY HEAT, ACTIVITY AND DOSE RATE CALCULATION

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SUMMARY

The knowledge of the residual activity, the decay heat power of irradiated fuel is needed in the different areas of the nuclear reactor industry. The neutron and gamma equivalent dose rate is also needed before shipping cask loading and transportation. The core physicist, the nuclear plant operator, the spent fuel conveyor and the safety authorities, all of them need data on residual activity and decay heat power. For this reason, the FAKIR calculation code has been developed, it takes into account the contribution of Fission Products, U239 + Np239 and Actinides. FAKIR is easy to use and provides good results of UO2 and MOX fuel calculations, it has been developed by CEA, under EDF and COGEMA contracts as the standard method of calculation for both transport and storage of all PWR fuel assemblies received.

INTRODUCTION

FAKIR has been developed with two aims:

- at the origin, FAKIR was only used for irradiated fuel assembly casks before package loading,
- in a second time, FAKIR has been used by EDF and CEA to calculate the core residual heat after shutdown for safety reasons (core and storage pool cooling during and after the irradiated fuel element discharging).

The purpose of the paper is to describe the FAKIR 6.0 capabilities corresponding to the first aim only. Prior to spent fuel transport, it must be demonstrated that the decay heat, the residual activity of the content and the neutron and gamma equivalent dose rates, all around the package, do not exceed the limits specified in the corresponding package approval. Since spent fuel with high burn-up is currently transported, it becomes more and more important to learn how to keep the radiation equivalent dose rate surrounding the fuel package as low as possible, for the purpose of reducing the exposure of the operators working on the packages. Radiation source is an important physical quantity for the design of systems in which the radiation emission (neutron and gamma-ray) has to be taken into consideration.

The FAKIR 6.0 code contains: the user friendly interface, the PEPEBAC module which calculates the fuel evolution during operation and cooling and the KAHINA module which determines the neutron and gamma-ray sources and deduces the equivalent dose rate. FAKIR uses simple formula and tabulated data libraries which explain its short running time. It can

treat UO2 and MOX fuels from PWR , a PC will be enough for the calculations. The input data is reduced, it contains only the parameters from the power histogram and some informations relative to the fuel assemblies such as the types of reactor and fuel assemblies. The tabulated libraries are obtained using: the cell code APOLLO 1, the evolution code PEPIN (fission products and actinides), the gamma-ray attenuation code MERCURE 5 and for neutron equivalent dose rate calculations the Monte Carlo code TRIPOLI 3. These codes are briefly described below.

DESCRIPTION OF THE USED CODES

APOLLO 1 code

APOLLO solves the Boltzman equation using the collision probability method and determines the isotope concentrations and the neutron related parameters such as flux and cross sections which are functions of the burn-up. The self shielded cross sections are taken into account using the sophistical algorithms from the APOLLO to provide homogeneised cross sections used for the core calculations. The cell code APOLLO uses the exact geometry, moderating ratio and enrichment for depletion calculations. The APOLLO code is largely used within E. D. F. and by FRAMATOME for fuel management and reactor design, and has been approved by the French safety authority.

PEPIN code

PEPIN solves analytically the differential equations describing the evolution and depletion of 699 fission products. The PEPIN is used in France for safety analysis, reprocessing, design of shipping casks and waste storage calculations. The PEPIN code is used to carry out the calculation of residual heat and gamma-ray source due to fission products.

MERCURE 5 code

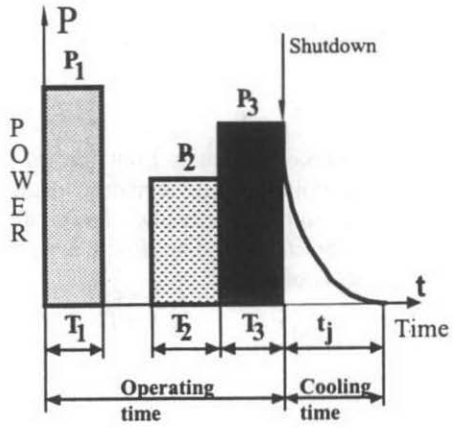
MERCURE 5 calculates equivalent dose rate and heating using the line of sight point attenuation kernel method. The integration of the transfer kernel is performed using a Monte Carlo technique in multigroup approximation. The program itself calculates the importance distribution that is used to sample the source particles. The three dimensional treatment of the geometry allows to describe exactly the shipping casks with their baskets and their shield.

TRIPOLI 3 code

The TRIPOLI 3 code was used for calculating the neutron equivalent dose rate . TRIPOLI is a Monte Carlo simulation code that solves Boltzman's equation for neutral particles transport (neutron and gamma-ray). It is a three dimensional polykinetic code. Cross sections are issued from ENDF/B-IV and ENDF/B-VI evaluated files through NJOY nuclear data system. Cross sections are represented by 315 groups. Sophisticated biasing techniques allow to treat deep penetration in shield with a good accuracy.

DESCRIPTION OF THE PEPABAC MODULE

We examine now this module with more details. PEPABAC can treat all power diagram described by steps:



Power histogram

(The power histogram describes the variation of power versus time, subdivided into intervals of constant power output.)

The notations used in PEPABAC are the following:

f_{iL} = Fractional fission of isotope i during irradiation step L.

Q_i = Thermal energy released from one fission of isotope i.

$Q_{0,L}$ = Thermal energy released from exceeding neutron capture in structural material.

P_{iL} = Fractional heat power released from one fission of isotope i during irradiation step L (MW per ton of heavy metal).

P_L = Total thermal power of the fuel during irradiation step L.

t = Burn-up (MWd per ton of heavy metal).

t_j = Cooling time.

T_L = Duration of irradiation step L.

$F_i(t_j,\infty)$= Decay Heat Power due to Fission Products from the fission of nuclide i, at the cooling time t_j, after the end of an infinite irradiation time referred to one fission per second without capture.

$Q(\tau,t_j)$= Average decay energy (function of Burn-up and cooling time)

$G(\tau,t_j)$= Corrector factor which takes into account the neutron capture by Fission Products.

$P_{FP}(t_j)$: The decay heat power, at cooling time t_j, due to Fission Products.

The number of fissions per second of fissile nuclide i during irradiation step L is determined by division of the heat power released by fission of nuclide i by fission energy of the same nuclide.

Thus:

$$\frac{P_{i,L}P_L}{Q_i} = \frac{P_L f_{i,L}}{\sum_i^n f_{i,L}Q_i + Q_{0,L}}$$

The decay heat power (without capture) released from one fission per second of the fissile isotope i, after the irradiation step L, is given as follows:

$$\left[F_i(t_j,\infty) \;-\; F_i(t_j+T_L,\infty)\right]$$

The decay heat power due to Fission Products from the fissions of nuclide i during the irradiation L was obtained by multiplication of the number of fissions of nuclide i per second by the decay heat power released from the elementary fission of isotope i. The summation of this decay heat power on irradiation step L and on fissile isotopes i gives the total decay heat power due to Fission Products without capture as:

$$\sum_{L}^{N} \frac{P_L}{\sum_{i}^{n} f_{i,L} Q_i \;+\; Q_{O,L}} \sum_{i}^{n} f_{i,L}\left[F_i(t_j,\infty) \;-\; F_i(t_j+T_L,\infty)\right]$$

To take into account the neutron capture of Fission Products, the decay heat power (without capture) was multiplied by the corrector factor $G(\tau,t_j)$. Finally:

$$P_{PF}(t_j) = \left\{ \sum_{L}^{N} \frac{P_L}{\sum_{i}^{n} f_{i,L} Q_i \;+\; Q_{O,L}} \sum_{i}^{n} f_{i,L}\left[F_i(t_j,\infty) \;-\; F_i(t_j+T_L,\infty)\right] \right\} G(\tau,t_j)$$

The Activity due to Fission Products is got by division of the decay heat power $P_{FP}(t_j)$ by the average decay energy $Q(\tau,t_j)$.

The quantities $F_i(t_j,\infty)$, $G(\tau,t_j)$ and $Q(\tau,t_j)$ were pre-tabulated and constitute libraries of FAKIR (PEPABAC module). The fractional fission f_{iL} will be calculated by the Actinides calculation module.

The contribution of Cs 134 to the decay heat power is significant at long cooling time between 10^5 and 10^8 s, thus, it's necessary to treat explicitly the Cs 134 taking into account the operating condition.. From the concentration of Cs 134, its Activity and Decay Heat Power will be calculated and served to rectify the total decay heat power and activity due to Fission Products.

We examine now the contribution of Actinides (with U 239 and Np 239) To take into account the isotopic composition of the nuclear fuel (UO2 or MOX), the concentration of each isotope will be calculated, at time t, during irradiation or cooling.
The evolution of the concentration of the nuclide i, is described by the following equation:

$$\frac{dN_i(t)}{\phi dt} = \sum_j b^{\lambda}_{j\to i} \frac{\lambda_j}{\phi} N_j(t) + \sum_k b^{\sigma}_{k\to i} \sigma_{c,k} N_k(t) - \left(\frac{\lambda_i}{\phi} + \sigma_{a,j}\right) N_i(t)$$

Where :

$N_i(t)$: Concentration of nuclide i at time t.

λ_i : Decay constant of isotope i.

$\sigma_{c,k}$: Capture cross section of nuclide k.

$\sigma_{a,i}$: Absorption cross section of nuclide i.

$b^{\lambda}_{j\to i}$: Radioactive decay branching ratio from radio-nuclide j to nuclide i.

$b^{\sigma}_{k\to i}$: Capture branching ratio from nuclide j to nuclide i.

ϕdt : Fluence.

This isotopic evolution equation will be solved by Runge-Kutta method using the initial conditions (the isotopic concentration at t=0), the fluence and the pre-tabulated cross sections. The last ones are functions of isotopic composition of nuclear fuel, initial enrichment of U235 (for UO2) or Pu (for MOX) and Burn-up.
During cooling, where the flux is equal to zero, the production and the disappearing by neutron capture will be cancelled. The differential equation will be solved analytically. The chains taken into account are U 233, U 235, U 238 , Pu 236 and Th 232 chains.
The radio elements with short half time, will be ignored during irradiation. The concentrations of U 239 and Np 239 will be calculated from the U 238 ones after each cycle.

The Activity of Actinides, at the cooling time t, is determined as follows:

$$A_{Act.}(t) = \sum_{i}^{N} \lambda_i N_i(t)$$

and the Decay heat Power:

$$P_{Act.}(t) = \sum_{i}^{N} \lambda_i N_i(t) \overline{E_i}$$

For the UO2 fuel calculation, the initial enrichment in U 235 is used as interpolation parameter, but for the MOX fuel the Pu 239 equivalent is served to interpolate the different tabulated data. The Pu 239 equivalent represents the reactivity equivalence of all heavy isotopes found in MOX fuel and the neutronic conservation conditions.

DESCRIPTION OF KAHINA MODULE

Before the total equivalent dose rate calculation around the shipping cask, FAKIR needs the neutron and gamma- ray sources for each spent fuel element contained in the different baskets. For this first purpose, the KAHINA module uses the informations determined by the PEPABAC module:

1. the neutron sources of each assembly, deduced from the actinide concentrations and tacking into account:
   the contribution of the neutrons produced by spontaneous fissions;
   the contribution of the neutrons produced by induced fissions;
   the contribution of neutrons resulting from (α,n) reactions.
   The multiplication factor (Keff) is taken into account for the neutron source intensity (induced fission) at a later step during equivalent dose rate calculation step.

2. the fission cross sections versus the burn up of each fissile isotope and for each spent fuel element. KAHINA will determine the spectra emitted by the most important fission products using these cross sections.

GAMMA-RAY SOURCE DETERMINATION BY KAHINA

The chosen solution allows to calculate the gamma-ray spectra with a good accuracy for the cooling times between 30 days and 100 years. Only 35 fission products have been considered. The precursors with short half time are neglected and give directly the first fission product of each chain. The cumulative fission yield is then used for the head of the chain. The differential equation giving the concentration of fission products is solved using a RUNGE KUTTA method which permits to take into account the neutron flux, the nuclear power and the capture and fissile cross section variations versus burn up. From these concentrations, KAHINA deduces the activity of each fission product and finally the gamma-ray spectra at the given cooling time.. The gamma-ray spectra are defined in a multigroup structure with 16 groups from 71.3 keV until 7.5 MeV. The group structure is the same as the one used in MERCURE 5. The nuclear data (yields, branching ratios, fission product capture cross sections and decay gamma-ray energies) are coming from the PEPIN library.

EQUIVALENT DOSE RATE CALCULATION

A set of points are disposed in the mid plan of the shipping cask: at the contact, at 1 and 2 meters of the outer surface of the cask. The casks actually considered are the TN12 and the NTL8/3. MERCURE 5 and TRIPOLI 3 calculations allow to give the gamma-ray dose rate $E_{\gamma,g}(J,P_k)$ for each group g and the neutron dose rate $E_n(J,P_k)$ at each point P_k and for each basket position J. In these calculations, the gamma-ray (per group g) and neutron source densities are unitary (1 particle per unit of spent fuel element volume). From the gamma-ray $S_{\gamma,g}(J)$ and neutron $S_n(J)$ source densities of the spent fuel element located at position J, calculated previously by KAHINA module, we deduce the gamma dose rate $D_\gamma(P_k)$ and the equivalent neutron dose rate $D_n(P_k)$ at point P_k:

$$D_\gamma(P_k) = \sum_J \sum_g S_{\gamma,g}(J) \times E_{\gamma,g}(J,P_k)$$

$$D_n(P_k) = \sum_J S_n(J) \times E_n(J,P_k)$$

Note that the coefficients $E_{\gamma,g}(J,P_k)$ and $E_n(J,P_k)$ are, for a given cask, once for all precalculated in a FAKIR library.

FAKIR VALIDATION

The validation of FAKIR is based on the comparison of FAKIR's results with the other calculation codes as PEPIN (French code), FISPIN (English code), ORIGEN (American code) and KORIGEN (American, German code). The fuel assemblies used for the comparisons are UO2 and two MOX from PWR. We give in the following tables 1 and 2 a comparison for a fuel with initial enrichment 3.7% in U^{235} and final burn up of 60000 MWd/TU.

Table 1

DECAY HEAT POWER (kW/TU)-T. 1					
Cooling time	*FIS-PIN*	*ORI-GEN*	*KORI-GEN*	*PE-PIN*	*FA-KIR*
15 d	84.48	79.47	80.40	82.89	83.26
1 m	62.90	58.03	59.50	61.62	61.92
3 m	37.99	34.54	36.00	37.27	37.49
6 m	26.09	23.78	24.80	25.50	25.67
9 m	20.13	18.52	19.20	19.62	19.75
1 y	16.50	15.36	15.80	16.06	16.17
1,25 y	13.96	12.80	13.50	13.61	13.70
1,50 y	12.06	11.50	11.70	11.76	11.85
2 y	9.35	9.07	9.15	9.14	9.22
3 y	6.24	6.18	6.23	6.13	6.20
5 y	3.78	3.83	3.89	3.77	3.83
10 y	2.41	2.49	2.54	2.45	2.50

Table 2

ACTIVITY (MCi/TU)-T. 1					
Cooling time	*FIS-PIN*	*ORI-GEN*	*KORI-GEN*	*PE-PIN*	*FA-KIR*
15 d	19.05	18.66	18.00	18.58	18.65
1 m	14.36	13.92	13.50	14.03	14.08
3 m	8.46	8.21	7.99	8.29	8.32
6 m	5.51	5.37	5.25	5.40	5.43
9 m	4.22	4.11	4.05	4.13	4.16
1 y	3.49	3.40	3.36	3.41	3.43
1,25 y	2.99	2.83	2.89	2.93	2.95
1,50 y	2.62	2.55	2.54	2.56	2.58
2 y	2.08	2.03	2.03	2.04	2.05
3 y	1.46	1.42	1.45	1.43	1.44
5 y	0.95	0.94	0.98	0.95	0.96
10 y	0.66	0.66	0.69	0.66	0.66

Table 3 Comparison between neutron source calculated by the cell code APOLLO 1 (reference) and by the module KAHINA

Neutron source (n/s/TU)			
Element	APOLLO	KAHINA	K/A
Pu238	2.899E6	2.928E6	1.01
Pu239	2.382E5	2.396E5	1.01
Pu240	2.749E6	2.937E6	1.07
Pu241	1.975E3	2.054E3	1.04
Pu242	9.874E5	9.974E5	1.01
Am241	1.065E5	1.118E5	1.05
Am242	9.764E1	9.918E1	1.02
Am243	1.692E4	1.724E4	1.03
Cm242	3.587E8	3.565E8	0.99
Cm243	3.807E4	3.246E4	1.17
Cm244	3.733E8	3.900E8	1.04
All	7.390E8	7.538E8	1.02

Table 4 Result (10^{-5} Sv/h) of equivalent dose rate measured and calculated using FAKIR on a TN12 cask.

contact			1 meter			2 meters			
γ	n	total	γ	n	total	γ	n	total	
9.4	5.3	14.7	4.9	2.9	7.8	3.3	1.9	5.2	measure
8.4	3.3	11.7	4.6	1.5	6.1	3.1	0.8	3.9	FAKIR

CONCLUSION

Some improvements will be needed for the equivalent neutron dose rate calculation for the TN12 shipping cask. The shield for this one contains 30 cm thick of iron. A recent version of TRIPOLI 3 allows to treat with a better accuracy the iron cross sections using the probability table method. After FAKIR 7, which is devoted to core residual heat calculations, FAKIR 8 is under development: its contains the capabilities of FAKIR 6 and 7 and will be able to calculate total residual heat uncertainties and to treat spent fuel provided after reprocessed Uranium containing U^{234} and U^{236}.

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