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VAREX, A Code for Variational Analysis of Reactivity Effects: Description and Examples

Jan Leen Kloosterman
Interfaculty Reactor Institute
Delft University of Technology
Mekelweg 15, NL-2629 JB Delft, Netherlands
J.L.Kloosterman@IRI.TUDelft.nl, www.iri.tudelft.nl

Jim C. Kuijper
NRG
P.O. Box 25, NL 1755 ZG Petten, Netherlands
Kuijper@NRG-nl.com, www.nrg-nl.com

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ABSTRACT

The code VAREX is described that calculates by first-order perturbation theory the contributions of individual nuclides to reactivity changes induced by, for example, a change of temperature or nuclide densities. Furthermore, VAREX calculates the contributions that make up the well-known four-factor formula in reactor physics, which has proven to be a very useful tool to characterize the neutron spectrum. The (effective) delayed neutron fractions per nuclide and for the whole geometry are calculated by use of two nuclear data libraries that are based on JEF2.2. Finally, the mean neutron generation time can be calculated. Results of calculations are given to show the reader the type of information that VAREX can provide.

1. INTRODUCTION

The fuel temperature coefficient (FTC), and the moderator temperature coefficient (MTC) are important parameters that determine to a large extend the time-dependent behavior of a nuclear reactor. Often one is interested only in the sign and the absolute value of the two-mentioned reactivity coefficients, but sometimes more information is needed to get insight into the physics of the underlying problem. Extra information can be obtained by first-order perturbation theory, which can be used to calculate the contribution of the individual isotopes to the reactivity change upon a variation of the temperature and/or the nuclide densities.

This paper describes the code VAREX, which originally was the acronym for Variational Analysis of Reactivity Effect with XSDRNPM. It calculates by first-order perturbation theory the isotopic contributions to reactivity changes. Later, modifications to the code were made to include the calculation of the (effective) delayed neutron fractions and accompanied decay constants of individual nuclides and mixtures, as well as the mean neutron generation time. Furthermore, the code can now be used not only in connection with the SCALE code system (SCALE4.2, 1994), but also with WIMS (WIMS8, 1999).

Chapter two of this paper reviews first-order perturbation theory, and shows the equations that are calculated by VAREX. Chapter three explains the data that are needed by the code, as well as how VAREX should be used in connection with either the SCALE code system or WIMS. Results of calculations with VAREX are shown in chapter four. This should give the reader some feeling of the type of extra information VAREX can provide. The last section provides some suggestions for further modifications.

2. CALCULATIONAL METHODS

2.1 Reactivity effects

In operator notation, the transport equation reads (Ott and Neuhold, 1985):

$$M\Phi = \frac{1}{k} F\Phi, \tag{2.1}$$

where M is the migration operator for leakage and absorption processes, F the fission operator, Φ the neutron flux and k the eigenvalue. The equation mathematically adjoint to the neutron transport equation reads:

$$M^*\Phi^* = \frac{1}{k} F^*\Phi^* \tag{2.2}$$

with M^* and F^* the operators adjoint to M and F, respectively. The solution of this equation is the so-called adjoint function or importance proportional to the reactivity increase per neutron introduced in the system at the phase-space coordinates considered. Note that the eigenvalue of Eq. (2.2) is equal to the eigenvalue of the forward transport equation.

If the microscopic cross section of a nuclide or the nuclide's atomic density is perturbed, the transport operator M and the fission operator F will change accordingly:

$$M' = M + \delta M, \quad \text{and} \quad F' = F + \delta F. \tag{2.3}$$

Inserting these into the transport equation will lead to a perturbed eigenvalue k' and a perturbed neutron flux Φ':

$$M'\Phi' = \frac{1}{k'} F'\Phi'. \tag{2.4}$$

Multiplying all terms in Eq. (2.4) with the adjoint function Φ^* (the solution of Eq. (2.2)), substituting Eq. (2.3), and taking the inner product gives:

$$\left\langle \Phi^*, M\Phi' \right\rangle + \left\langle \Phi^*, \delta M\Phi' \right\rangle = \left\langle \Phi^*, \frac{1}{k'} F'\Phi' \right\rangle, \tag{2.5}$$

which, when making use of the following properties of adjoint operators:

$$\left\langle \Phi^*, M\Phi' \right\rangle = \left\langle M^*\Phi^*, \Phi' \right\rangle = \frac{1}{k}\left\langle F^*\Phi^*, \Phi' \right\rangle = \frac{1}{k}\left\langle \Phi^*, (F' - \delta F)\Phi' \right\rangle, \tag{2.6}$$

can be used to find for the reactivity change $\delta\rho \equiv \frac{1}{k} - \frac{1}{k'}$:

$$\delta\rho = \frac{\left\langle \Phi^*, \frac{1}{k}\delta F\Phi' \right\rangle - \left\langle \Phi^*, \delta M\Phi' \right\rangle}{\left\langle \Phi^*, F'\Phi' \right\rangle}. \tag{2.7}$$

This equation according to exact perturbation theory contains no approximation, and can be used to calculate the reactivity effect due to a change of the *M* and *F* operators. However, if the reactivity effects due to changes in the microscopic cross sections or the atomic density of each individual isotope have to be calculated, Eq. (2.4) has to be solved for each perturbation, which is very impractical, and in many cases even impossible. The CSAS drivers of the SCALE code system and the WIMS code system, for example, do not accept a fuel composition containing isotopes with different temperatures. Therefore, a simplified procedure is followed to expand the perturbed operators and to neglect all second-order effects in Eq. (2.7):

$$\delta\rho = \frac{\left\langle \Phi^*, \frac{1}{k}\delta F\Phi \right\rangle - \left\langle \Phi^*, \delta M\Phi \right\rangle}{\left\langle \Phi^*, F\Phi \right\rangle}. \tag{2.8}$$

In multi-group notation, the first term of the numerator reads:

$$\left\langle \Phi^*, \frac{1}{k}\delta F\Phi \right\rangle = \frac{1}{k}\int_V dV \sum_{g=1}^{G} \delta\chi_g \Phi_g^* \sum_{g'=1}^{G} \nu_{g'} \Sigma_{g'}^f \Phi_{g'} + \frac{1}{k}\int_V dV \sum_{g=1}^{G} \chi_g \Phi_g^* \sum_{g'=1}^{G} \delta\nu_{g'} \Sigma_{g'}^f \Phi_{g'} + \frac{1}{k}\int_V dV \sum_{g=1}^{G} \chi_g \Phi_g^* \sum_{g'=1}^{G} \nu_{g'} \delta\Sigma_{g'}^f \Phi_{g'} \tag{2.9}$$

where *V* is the total volume of the problem under consideration, *g* is the index for the energy group and *G* the total number of energy groups considered. The macroscopic cross section for fission is Σ^f. The average number of neutrons released per fission equals ν, which are distributed in energy according to the fission spectrum χ. In Eq. (2.9), a perturbation in the total fission operator is due to a perturbation of the fission spectrum, a perturbation of the average number of neutrons released, or a perturbation of the macroscopic fission cross section. The latter can be due to perturbations of the microscopic cross sections and/or the atomic densities:

$$\delta\Sigma^f_{g'} = N\delta\sigma^f_{g'} + \delta N\sigma^f_{g'} \tag{2.10}$$

The perturbations in the transport operator *M* include the effects of absorption and scattering. In multi-group notation, the second term in the numerator of Eq. (2.8) reads:

$$\left\langle \Phi^*, \delta M\Phi \right\rangle = \int_V dV \sum_{g=1}^{G} \Phi^*_g \delta\Sigma^a_g \Phi_g + \int_V dV \sum_{g=1}^{G} \Phi_g \sum_{g'=1}^{G} \delta\Sigma^s_{g,g'} \left(\Phi^*_g - \Phi^*_{g'} \right). \tag{2.11}$$

Where Σ^a and Σ^s are the macroscopic cross section for absorption and scattering, respectively. Similar to the fission cross section in Eq. (2.10), the perturbations of the macroscopic absorption and scattering cross sections may be due to changes of the microscopic cross sections and/or the atomic densities.

Note that VAREX uses always first-order perturbation theory for the calculation of the reactivity effects of the individual nuclides (Eq. (2.8)), because the perturbed flux calculated by, for example, the SCALE code system is the flux representing the situation where the temperature of *all* nuclides in the same geometrical zone (e.g. fuel or moderator) is changed, which in general is not a good representation of the situation where the temperature of *only one* nuclide has been modified.

2.2 Four-factor formula

As will be shown in Chapter 4, it is quite instructive to calculate the components of the four-factor formula (Duderstadt and Hamilton, 1976) during burnup, possibly as a function of some other parameter like the fuel temperature. The program VAREX calculates the fast-fission factor ε as the ratio of the total neutron production rate and the thermal neutron induced neutron production rate:

$$\varepsilon = \frac{\int_{V_f} dV \left(\sum_{g=1}^{G} \nu_g \Sigma^f_g \Phi_g \right)}{\int_{V_f} dV \left(\sum_{g=1}^{T} \nu_g \Sigma^f_g \Phi_g \right)}. \tag{2.12}$$

The integrals in the numerator and the denominator extend over the fuel volume, which is implicitly taken as any volume containing at least one fissile nuclide. *G* is the total number of neutron groups, while *T* is the number of thermal neutron groups, with an upper boundary set by default to the first thermal group in the AMPX working library that receives an up-scatter source (parameter IFTG in record number 1). However, if needed, the user can modify this parameter. From the definition above, it is clear that the fast-fission factor could better be called the non-thermal fission factor, because resonance-neutron induced fission is included in the numerator of Eq. (2.12).

The resonance escape probability *p* is calculated as the ratio of the thermal neutron absorption rate and the total neutron absorption rate:

$$p = \frac{\int_V dV \left(\sum_{g=1}^{T} \Sigma_g^a \Phi_g \right)}{\int_V dV \left(\sum_{g=1}^{G} \Sigma_g^a \Phi_g \right)}, \tag{2.13}$$

where the integrals in the numerator and denominator extend over the whole volume of the problem under consideration. Note that for the calculation of the four factor formula, the absorption cross section is a modified one corrected for the (n,2n) and the (n,3n) neutron production:

$$\Sigma^{a^*} = \Sigma^a - \Sigma^{n,2n} - 2\Sigma^{n,3n} \tag{2.14}$$

The reproduction factor η is calculated as the ratio of the thermal neutron induced neutron production rate and the thermal neutron absorption rate in the fuel zones:

$$\eta = \frac{\int_{V_f} dV \left(\sum_{g=1}^{T} \nu_g \Sigma_g^f \Phi_g \right)}{\int_{V_f} dV \left(\sum_{g=1}^{T} \Sigma_g^a \Phi_g \right)}, \tag{2.15}$$

where, similarly to the definition of ε, the integrals extend over all the fuel zones of the problem (all zones containing any fissile nuclide).

Finally, the thermal utilization factor is calculated by the ratio of the thermal neutron absorption rate in the fuel zones and the total thermal neutron absorption rate:

$$f = \frac{\int_{V_f} dV \left(\sum_{g=1}^{T} \Sigma_g^a \Phi_g \right)}{\int_V dV \left(\sum_{g=1}^{T} \Sigma_g^a \Phi_g \right)}, \tag{2.16}$$

It can easily be verified that the product of the four factors calculated in VAREX give a correct expression for the infinite multiplication factor of the system.

If VAREX is run for both an unperturbed case and a perturbed case, it calculates the contribution of ε, p, f, and η to the reactivity effect of the perturbation. This means that $\delta\varepsilon / \varepsilon$, $\delta p / p$, $\delta f / f$, and $\delta\eta / \eta$ are calculated such that (Kruijf and Janssen, 1993):

$$\frac{\delta k}{k} = \frac{\delta\varepsilon}{\varepsilon} + \frac{\delta p}{p} + \frac{\delta f}{f} + \frac{\delta\eta}{\eta}. \tag{2.17}$$

Note that the exact unperturbed and perturbed fluxes are used for this calculation.

2.3 Kinetic parameters

Two data libraries are linked to VAREX that contain data for the production and energy distribution of delayed neutrons based on nuclear data from JEF2.2. With these data, the delayed neutron fraction for a single nuclide β^i, and the associated adjoint-weighted delayed neutron fraction β^i_{eff} can be calculated, as well as the β and β_{eff} for a mixture of nuclides. The delayed neutron fraction β^i for nuclide i is calculated by (Saphier, 1986):

$$\beta^i_l = \frac{\int_V dV \sum_{g'=1}^{G} W_{g'} \chi^i_{g',l} \sum_{g=1}^{G} \nu^i_d \Sigma^{f,i}_g \Phi_g}{\int_V dV \sum_{g'=1}^{G} W_{g'} \chi^i_{g'} \sum_{g=1}^{G} \nu^i_t \Sigma^{f,i}_g \Phi_g} = \frac{D^i_l}{T^i}, \tag{2.18}$$

where the weighting function $W_{g'}$ is flat for the calculation of β^i_l, and equal to the adjoint function for the calculation of the $\beta^i_{eff,l}$. In Eq. (2.18), the index l is for the delayed group number ranging from 1 to L (usually 6). Note that the only differences between the numerator and the denominator are the number of neutrons released per fission and the associated neutron fission spectra. In simplified form, the delayed neutron fraction equals the ratio of the delayed neutron production rate (D^i_l) and the total neutron production rate (T^i). The one-group delayed neutron fraction for a nuclide i is just the sum over the delayed neutron groups:

$$\beta^i = \sum_{l=1}^{L} \beta^i_l \quad \text{and} \quad \beta^i_{eff} = \sum_{l=1}^{L} \beta^i_{eff,l} \tag{2.19}$$

with associated decay constants:

$$\lambda^i = \frac{\sum_{l=1}^{L} \beta^i_l}{\sum_{l=1}^{L} \frac{\beta^i_l}{\lambda^i_l}} \quad \text{and} \quad \lambda^i_{eff} = \frac{\sum_{l=1}^{L} \beta^i_{eff,l}}{\sum_{l=1}^{L} \frac{\beta^i_{eff,l}}{\lambda^i_l}}. \tag{2.20}$$

Besides the one-group delayed neutron fraction of each nuclide (Eq. (2.19)), also the fractional contribution of each nuclide to the total delayed neutron production is calculated (the numerator of Eq. (2.18) D^i_l summed over all delayed groups divided by the total delayed neutron production in the fuel). Similarly, the (effective) delayed neutron fraction per delayed group l is calculated by summing the delayed neutron production in group l over all nuclides i with a non-zero fission cross section:

$$\beta_l = \frac{\sum_{i=1}^{N} D_l^i}{\sum_{i=1}^{N} T^i} \quad \text{and} \quad \beta_{eff,l} = \frac{\sum_{i=1}^{N} D_{eff,l}^i}{\sum_{i=1}^{N} T^i} \tag{2.21}$$

with:

$$\lambda_l = \frac{\sum_{i=1}^{N} D_l^i}{\sum_{i=1}^{N} \frac{D_l^i}{\lambda_l^i}} \quad \text{and} \quad \lambda_{eff,l} = \frac{\sum_{i=1}^{N} D_l^i}{\sum_{i=1}^{N} \frac{D_l^i}{\lambda_l^i}}. \tag{2.22}$$

The total one-group delayed neutron fraction is calculated by summing the β_l over all delayed neutron groups.

In addition to the delayed neutron data, also the mean neutron generation is needed by codes that solve the point-kinetics equation. This parameter is calculated by:

$$\Lambda = \frac{\left\langle \Phi^*, \frac{1}{v}\Phi \right\rangle}{\left\langle \Phi^*, F\Phi \right\rangle} = \frac{\int_V dV \sum_{g=1}^{G} \frac{\Phi_g^* \Phi_g}{v_g}}{\int_V dV \sum_{g'=1}^{G} \Phi_{g'}^* \chi_{g'} \sum_{g=1}^{G} \nu_g \Sigma_g^f \Phi_g}, \tag{2.23}$$

where the importance Φ^* is the solution of the adjoint transport equation (Eq. (2.2)), and v_g in the numerator is the group-averaged neutron velocity.

3. CONNECTION TO OTHER CODES

3.1 SCALE

The CSAS drivers of the SCALE code system run a sequence of codes to produces resonance-shielded and cell-weighted cross sections that can be used in shielding analyses or reactor calculations. At the Interfaculty Reactor Institute, BONAMI-S and NITAWL-S are used for the resonance shielding calculations in the unresolved and resolved energy regions, respectively, and XSDRNPM for the one-dimensional cell/zone-weighting calculations.

When VAREX is used with the SCALE code system, typically two CSAS runs are needed: one for the perturbed situation, for example with the cross sections of all the fuel nuclides at an elevated temperature, and one for the unperturbed situation. Use is made of the option in XSDRNPM to write the group fluxes into a binary file that is later on read by VAREX. Besides the perturbed and unperturbed CSAS runs, an adjoint transport calculation for the unperturbed situation needs to be done. To this purpose, a special program has been written that changes the forward/adjoint option in the binary input file for XSDRNPM produced by the CSAS driver in the unperturbed forward run, and converts the total input to ASCII format, which can subsequently used by XSDRNPM. In this case, we avoid the resonance shielding calculations for the adjoint calculation, which usually take the longest CPU time. The geometrical information, like the number of intervals and the volume of each interval, the composition of the mixture (nuclide densities) in each zone, and the calculated eigenvalue are read from the XSDRNPM output files. In conclusion, VAREX uses eight files produced by the CSAS calculations: three XSDRNPM output files (two forward runs, and one adjoint run), three XSDRNPM binary flux files, and two AMPX-working libraries (perturbed and unperturbed forward runs). Furthermore, VAREX uses two files containing nuclear data to calculate the (effective) delayed neutron fractions and the associated decay constants.

3.2 WIMS

The original VAREX code was designed for use with the one-dimensional neutron transport code XSDRNPM of the SCALE code system. To investigate multi-dimensional problems, NRG has interfaced the VAREX code with the well-known modular WIMS lattice code (WIMS8, 1999). WIMS performs resonance shielding and multi-group neutron transport calculations in general two-dimensional geometry (WIMS CACTUS module) to obtain the unperturbed and perturbed forward fluxes, required by VAREX. For the data transfer to VAREX, the group fluxes are given per volume element of the geometry under investigation. The same holds for the microscopic group cross sections for each of the nuclides present in the problem.

Contrary to XSDRNPM, the calculation of the required adjoint function (WIMS CRITIC module) is of a more approximate nature. Starting from the forward multi-group flux, the multi-group problem is flux-volume weighted into a single volume. Then the scatter matrix is transposed, the fission cross sections and the fission spectrum are interchanged and a "homogenized" adjoint spectrum is calculated with the same eigenvalue as the forward solution. Subsequently, this adjoint spectrum is expanded into an approximative multi-region adjoint function employing the original forward multigroup flux as a weighting factor. The VAREX calculational scheme has been tested quite succesfully in combination with the WIMS code. However, testing, verification and validation is still in progress.

4. EXAMPLES

In this section, some examples of results calculated with VAREX are presented to show the reader the type of information that can be obtained by VAREX.

4.1 Burnup calculations on (U,Pu) MOX fuel

Here, some results of burnup calculations on (U,Pu) MOX fuel in a standard PWR fuel lattice with a moderator-to-fuel (MF) volume ratio of 2 will be compared with similar calculations on UO_2 fuel with an initial U-235 enrichment of 4% (Kloosterman and Bende, 2000). The plutonium composition in the MOX fuel corresponds with the plutonium produced in the UO_2 fuel after a decay time of 5 years prior to reprocessing. The plutonium density in the MOX fuel equals 10%, which enabled us to reach for both fuels a discharge burnup value of 47.5 GWD/THM. At regular times during burnup, branching calculations have been performed with an elevated fuel temperature to calculate the Fuel Temperature Coefficient (FTC) as a function of burnup. VAREX has been applied to calculate at each branching the contribution of each fuel nuclide to the FTC, the four-factor formula, as well as the (effective) delayed neutron fraction and the contribution of each nuclide to the total delayed neutron production.

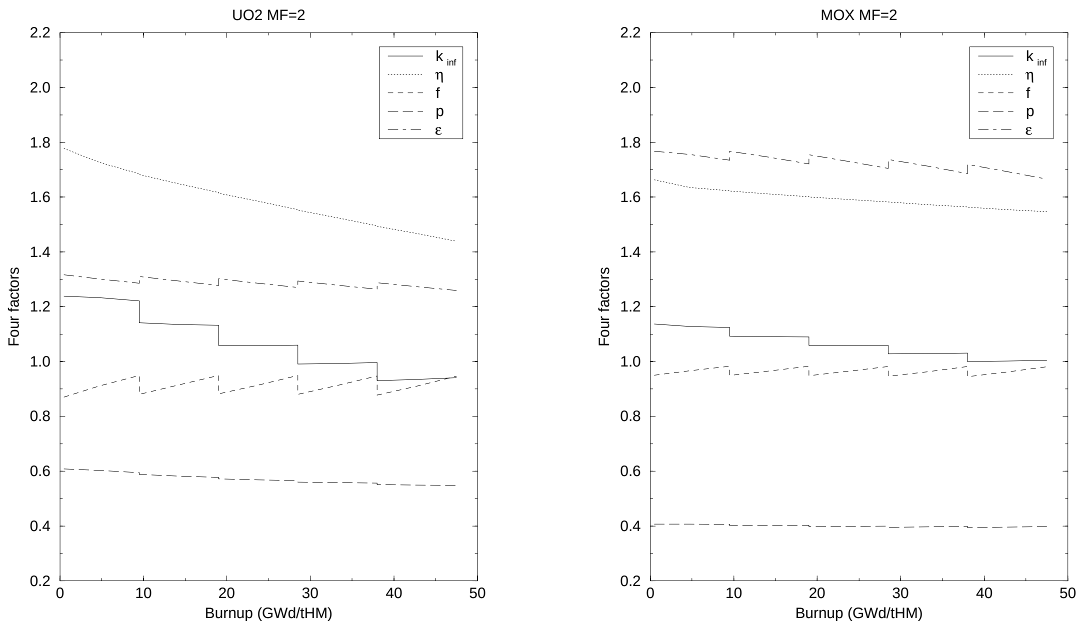

Fig. 1 The four factors as a function of burnup for UO_2 and MOX fuels. During each fuel cycle of 9.5 GWD/THM the boron concentration decreased linearly from either 875 ppm (UOX) or 1190 ppm (MOX) to zero (Kloosterman and Bende, 2000).

Figure 1 shows the components that make up the four-factor formula as a function of burnup. Clearly, the fast fission factor (or better non-thermal fission factor) ε is much larger for the MOX fuel than for the UO_2, which is mainly due to the harder neutron spectrum in the MOX. The reproduction factor η decreases more strongly for the UO_2 fuel than for the MOX, due to the buildup of plutonium isotopes, which in general has a very pronounced effect on the neutron spectrum and the absorption rates in the UO_2 fuel.

In Figure 2, the contributions of the individual isotopes to the effective delayed neutron production are shown as a function of burnup. In the left plot, the U-235 contribution decreases strongly, but remains dominant for all burnup values, while for the MOX fuel, the contribution of the fissile plutonium isotopes and of U-238 change only very gradually with burnup. For both fuels, the contribution of the U-238 is quite large, despite its rather small contribution to the fission power production rate. This is mainly due to the rather large delayed neutron production of U-238 (about 0.046 neutrons per fission versus 0.017 neutrons for U-235 and 0.0065 neutrons for Pu-239 (Duderstadt and Hamilton, 1976)). For both fuel types, the U-238 isotopic contribution to the FTC dominates (-2.5 pcm/K for UO_2 and -2.2 pcm/K for MOX), although for the MOX fuel there is an extra contribution of about -1 pcm/K due to the even plutonium isotopes.

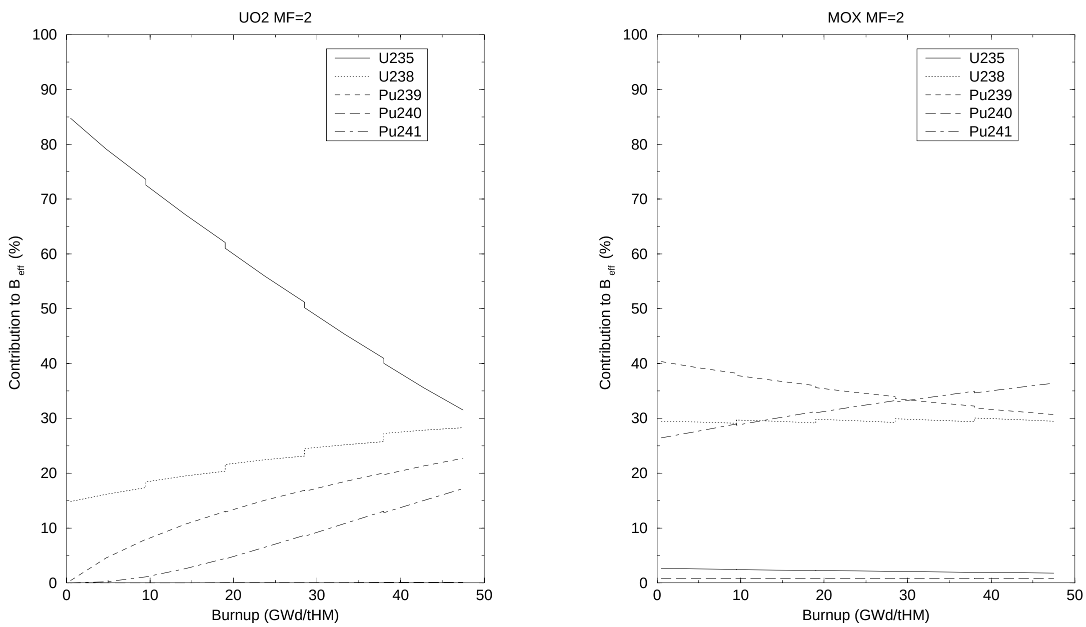

Fig. 2 The isotopic contributions to the effective delayed neutron production for UO_2 and MOX fuel. During each fuel cycle of 9.5 GWD/THM the boron concentration decreased linearly from either 875 ppm (UOX) or 1190 ppm (MOX) to zero (Kloosterman and Bende, 2000).

4.2 Burnup calculations on (Th,U) MOX fuel

In this section, some results of burnup calculations with the WIMS-7 code system on (Th,U) MOX fuel in a standard PWR fuel lattice will be compared with similar calculations on UO_2 fuel with an initial enrichment of 4.45% and a fuel discharge burnup of 60 GWD/THM (Kloosterman and Gruppelaar, 1999). The (Th,U) MOX fuel contained either Medium Enriched UO_2 fuel (MEU) with U-235 enrichment of 20% mixed in ThO_2, or Highly Enriched UO_2 fuel (HEU) with enrichment of 93% mixed in ThO_2.

Table 1 shows the values of the four factors at Begin of Life. It is seen that the differences between the UO_2 fuel at the one hand, and the (Th,U) MOX fuels at the other are much smaller than the differences between UO_2 fuel and (U,Pu) MOX fuels as shown in Fig. 1. When no U-238 is present (HEU in ThO_2), the resonance-escape probability p is larger by about 10% due to the relatively small resonance integral of Th-232 (85 barn compared with 278 barn for U-238).

Table 2 shows the contribution of the main isotopes to the FTC at Begin of Life. Clearly, the FTC is strongest negative for the MEU fuel mixed in ThO_2 due to the strong contributions of both the U-238 and Th-232 isotopes in the fuel.

Table 1 Components the make-up the four factor formula at Begin of Life (Kloosterman and Gruppelaar, 1999).

Fuel	ε	p	f	η
UO_2	1.26	0.64	0.86	1.89
MEU in ThO_2	1.24	0.63	0.88	1.73
HEU in ThO_2	1.21	0.69	0.87	1.68

Table 2 Isotopic contribution to the FTC (pcm/K) at Begin of Life according to first-order perturbation theory (Kloosterman and Gruppelaar, 1999).

Fuel	Th-232	U-238
UO_2	0	-1.9
MEU in ThO_2	-2.0	-1.3
HEU in ThO_2	-2.3	-0.2

4.3 Fuel particle design for a fluidized bed nuclear reactor

This last example (Kloosterman, *et al*, 2001) shows some results of calculations with VAREX coupled to the SCALE code system. The fuel considered here consists of TRISO coated fuel particles with a fixed outer diameter of 1 mm that contains a UO_2 fuel kernel. By varying the diameter of the fuel kernel, the moderator-to-fuel atomic ratio is varied. Note that the moderator in this case consists of carbon and that the moderator-to-fuel ratio equals the atomic ratio of carbon and uranium. The fuel particles should be designed such that the Fuel Temperature Coefficient (FTC) is strongly negative. Because the fuel kernel contains 17% enriched UO_2 fuel, the FTC is totally determined by the U-238. Therefore, it is of no use to plot the contribution of each individual nuclide. However, it is of interest to show the contributions to the FTC of the components that make up the four factor formula. These contributions are shown in Fig. 3 as a function of the moderator-to-fuel ratio.

The total FTC is made up of two large contributions with opposite sign. The resonance escape probability gives a large negative contribution, while the non-thermal fission factor gives a large positive contribution. The latter effect is due to the increase of the reaction rates by the fission resonances when these widen upon a temperature increase. When the moderator-to-fuel ratio increases, both contributions decrease in magnitude due to the better thermalization of the neutrons. The total FTC remains strongly negative.

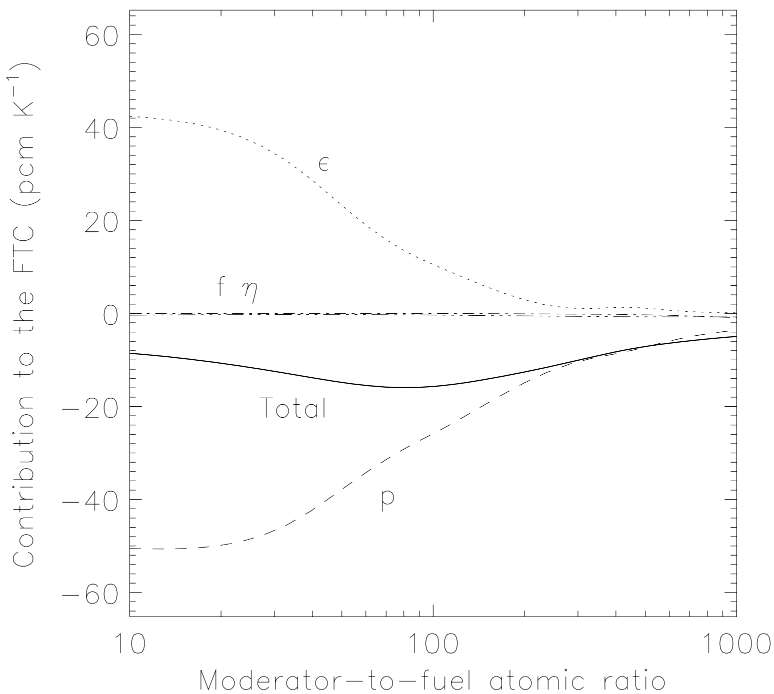

Fig. 3 Contributions to the FTC (exact calculation) of the components that make up the four-factor formula for a TRISO coated UO_2 fuel particle (Kloosterman, *et al*, 2001).

CONCLUSIONS and SUGGESTIONS

The VAREX code has been successfully applied to a large number of problems that needed more insight than could be provided by the standard tools. The main application of the code is in the field of the analysis of reactivity effects by means of first-order perturbation theory. To this purpose, it is common practice by the authors to use VAREX during the sequence of burnup calculations to calculate at several time steps the contributions of the individual isotopes to the various temperature coefficients of reactivity. Furthermore, the characterization of the neutron spectrum by means of the four-factor formula has been shown to give much insight into the reactor physics trends during the burnup of (exotic) fuels. The kinetic parameters calculated by VAREX, like the (effective) delayed neutron fraction and the mean neutron generation time, can be applied to further studies on the dynamics of fuel types and reactors (e.g. see Kloosterman, 2000). The code can be used in connection with the SCALE code system and WIMS.

At the moment, the coupling of VAREX to either SCALE or WIMS is not made in a general sense. For example, exchange of information between the SCALE code XSDRNPM and VAREX is done by reading the ASCII output file of XSDRNPM. This can be improved by making use of the CCCC output option in XSDRNPM, which contains the geometrical data and the neutron fluxes. In fact this output option is already in use at the Interfaculty Reactor Institute to plot fluxes and reaction rates calculated with XSDRNPM.

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