SENSITIVITY ANALYSIS OF ALLEGRO MOX CORE

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1. Introduction

ALLEGRO, a 75 MW_{th} reactor unit plays a vital role in the development of the electricity producing Gas-cooled Fast Reactor (GFR) prototype. As a demonstrator of the unique technology, never built before, it will serve to demonstrate the viability of the GFRs system. ALLEGRO design in corporates all the architecture and main materials and components foreseen for the GFR, except the power conversion systems [1]. The starting ALLEGRO configuration (ALLEGRO MOX) is a qualified technology core, characterized by standard mixed oxide (MOX) subassemblies consisting of fuel pins with stainless steel cladding operating at an average coolant temperature around 400°C [2].

Within the reactor analysis and design calculation, sensitivity analysis offers a nuclear engineer a unique insight into the investigated system. Estimation of the change of the system response, due to change in some input parameter, can identify important processes and evaluate the influence of variation in this parameter. The sensitivity coefficients can be further used for calculation of k_{eff} uncertainty coming from cross section data, sensitivity coefficients of reactivity response, cross section adjustment and integral experiment similarity assessment [3].

The calculation of sensitivity coefficients was carried out by the SCALE6 system which has capability to perform flux calculation in heterogeneous configurations by using Monte Carlo methodology. In combination with Standard Perturbation Theory (SPT) implemented in TSUNAMI module, the SCALE6 system provides the unique results not only for further design issues, but also for system validation and licensing [4]. The Sensitivity coefficients derived based on SPT can be written in a simple form as follows:

$$S_{k,\sigma} = \frac{\sigma}{k}\frac{\Delta k}{\Delta\sigma} \cong \frac{\Delta\sigma}{\sigma}\frac{\langle\Phi^*\left(\frac{1}{k}\frac{\partial P}{\partial\sigma}-\frac{\partial L}{\partial\sigma}\right)\Phi\rangle}{\frac{1}{k}\langle\Phi^* P\Phi\rangle} \qquad (1)$$

The uncertainty of the $k_{eff}$is then given approximately by

$$\sigma^2 = S_{k,\sigma} C_\sigma S_{k,\sigma}^T \qquad (2)$$

where $S_{k,\sigma}$ is the sensitivity coefficient of k_{eff} with respect to σ, which represents nuclear data like cross sections, fission spectrum or nubar. Symbols L and P in Eq.(1) are net loss and production Boltzmann operators; Φ^* and Φ are adjoint and forward fluxes respectively; σ^2 in Eq. 2 is the variance of the k_{eff} and C_σ is a covariance matrix of nuclear data used in the calculation. All input information necessary to determine the sensitivity coefficients and uncertainties by Eq.(1,2)can completely characterize the investigated system, therefore the sensitivity coefficients can be considered as unique fingerprint of the system.

2. ALLEGRO MOX core characterization

The 120° symmetric core includes 81 fuel subassemblies, the volumetric content of PuO_2in heavy metal is 25.5%. In addition, the ALLEGRO MOX core features 6 in-core dummy subassemblies of dedicated stainless steel 15-15Ti (AIM1), so far assumed homogeneous in geometry and composition. The horizontal cross-sectional view at the middle active core height is shown in Fig.1. It displays the reference core with reflector and shielding subassemblies. The control rod system is composed of 4 Diverse Shutdown Devices (DSD) and 6 Control and Shutdown Devices (CSD). The rod follower and absorber materials are respectively AIM1 and B_4C while the absorber part consists of the same boron content for both types of device. From center to periphery, the active core is surrounded by 4 additional rings of subassemblies consisting of AIM1 and by 3 rings of AIM1/B4C which serve as reflector and shielding subassemblies respectively. Both are assumed to be homogeneous in geometry and in composition [2].

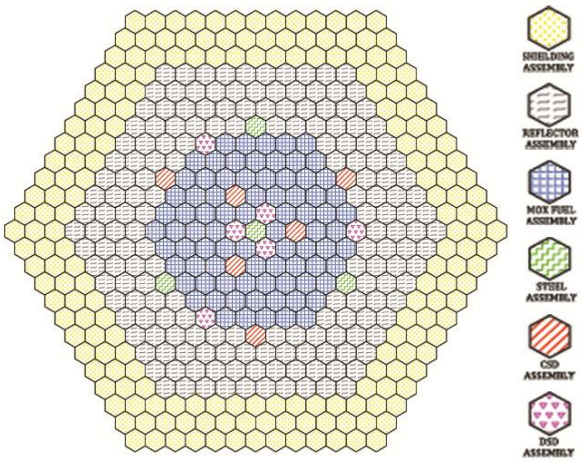


Fig.1: *Cross section of the ALLEGRO MOX core in the level of fuel area*

Heterogeneous geometry definition was considered only for the fuel part of fuel assembly and for the absorber part of CSD/DSD assemblies. For the neutron flux calculations square mesh was placed through the core with a uniform step of 1.5 cm in the fuel region. In other parts of the core, the size of the mesh was directly proportional to the distance from the core centre. The axial and radial dimensions of individual core regions were modified compared to the reference description to account for the thermal expansion of the structural materials under hot condition.

3. Discussion and results

Sensitivity and uncertainty analysis of the ALLEGRO MOX core was performed using two computational tools. In the first case, the TSUNAMI-3D code was utilized using ENDF/B-VII 238 group cross section data and 44groupcov covariances. Forward and adjoint transport calculations were carried out with KENO6 and the sensitivity coefficients were computed by the SAMS module [4]. In the second case, self-developed perturbation PORK code was used which is interconnected with the diffusion flux solver DIF3D [5] and ZZ-KAFAX-E70 [6] basedENDF/BVII nuclear data library collapsed from 150 to 25 groups. In the sensitivity analysis it is common to ensure correctness of the sensitivity coefficients using Direct Perturbation (DP) calculation for the most important nuclides. Therefore DP calculation was carried out for a group of the most sensitive nuclides, which includes majority of the fissile nuclides and main nuclides of the structural materials. The results and nuclide affiliation to the material of the core are presented in Tab.1.The sensitivity coefficients calculated by TSUNAMI-3D contain also a statistical uncertainty, but this uncertainty is not significant in most cases and therefore is not listed in this paper.

Tab.1.Energy and reaction integrated sensitivity coefficient for the most sensitive nuclides.

Material	Nuclide	Atom Density (atom/barn*cm)	Sensitivity (no-dim)			Error (%)
			DP	TSUNAMI-3D	PORK	
Fuel	Pu239	3.38250E-03	4.25266E-01	4.12350E-01	4.12812E-01	3.04
Fuel	Pu241	4.43250E-04	7.90059E-02	7.32350E-02	7.36110E-02	7.30
Fuel	U238	1.67980E-02	-6.69112E-02	-6.61660E-02	-9.52177E-02	1.11
Fuel	Pu240	1.55790E-03	3.88069E-02	3.64980E-02	3.50291E-02	5.95
Reflector *	Fe56	4.05280E-02	1.37285E-02	1.37110E-02	-1.71042E-02	0.13
Fuel	Pu238	1.63770E-04	1.20460E-02	1.23260E-02	1.21718E-02	-2.32
Fuel	U235	1.21830E-04	9.90339E-03	1.05700E-02	1.05833E-02	-6.73
Fuel	Pu242	4.35450E-04	5.82790E-03	6.48720E-03	6.07468E-03	-11.31
Reflector *	Cr52	8.77190E-03	5.98321E-03	5.95130E-03	-3.74535E-03	0.53
Reflector *	Ni58	6.76710E-03	4.71109E-03	4.49750E-03	-7.31611E-03	4.53
Absorber	B10	6.05480E-02	-4.40663E-03	-4.18000E-03	-8.18243E-04	5.14
Reflector **	Fe56	3.79950E-02	3.36164E-03	3.67040E-03	-2.82496E-03	-9.18

* material of the radial reflector, ** material of the axial reflector

The DP calculation was performed using the KENO6 code, where good agreement with the results from TSUNAMI-3D calculation was reached. The last column of Tab.1 demonstrates a difference between DP and TSUNAMI-3D calculation. None of the errors markedly exceeded 10%, which is acceptable. The situation is different in case of comparison of TSUNAMI-3D and PORK sensitivity coefficients. Sensitivity coefficients of the nuclides in structural materials calculated by the PORK code are negative where the sensitivity coefficients calculated by TSUNAMI-3D are strictly positive but with a similar order of magnitude. When the integral sensitivity coefficients for structural materials were decomposed to reactions, scattering reactions were identified as the main contributor, with same behaviour as it was for integral values.

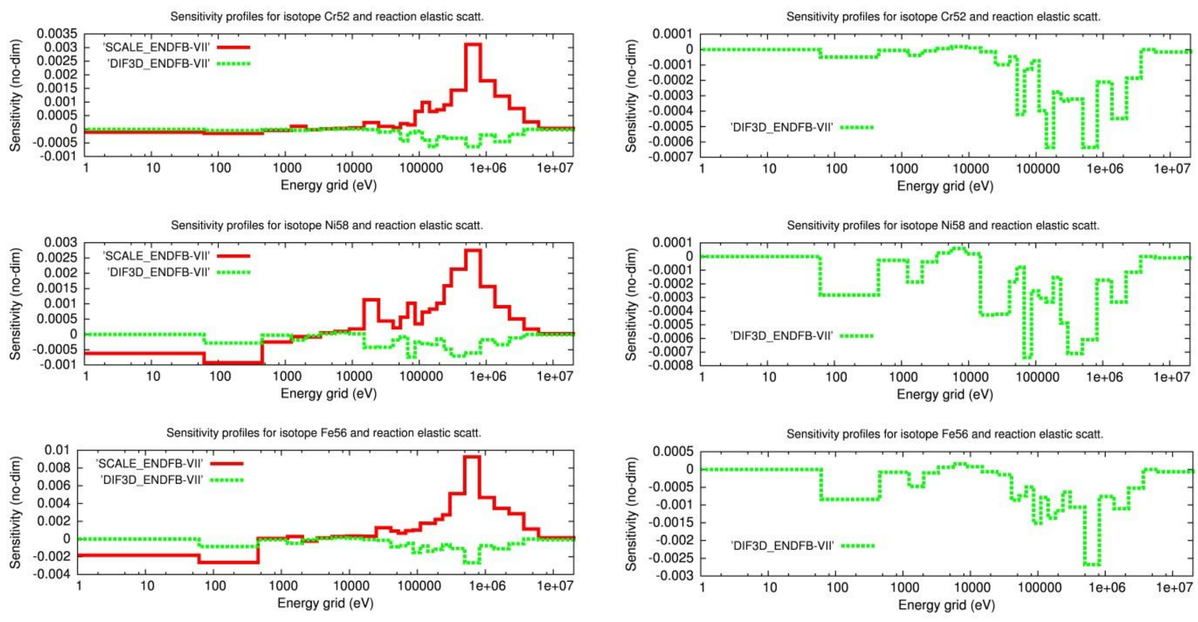


Fig.2: *Sensitivity profiles for scattering reactions and chosen nuclides.*

In Fig.2 the sensitivity profiles for elastic scattering are presented for three nuclides from structural materials (Fe56, Cr52 and Ni58) for which different integral sensitivity coefficient were obtained by calculation in TSUNAMI-3D and PORK codes. On the left side of Fig.2 the comparison of profiles for both codes is shown and on right side of Fig.2the sensitivity profiles from the PORK code are highlighted. Despite the fact that sensitivity profiles lay also in the positive and negative region, the importance of elastic scattering is the highest around incident neutron energy 1 MeV in both cases. This phenomenon has its origin in the core construction, where there is a lack of materials able to significantly slowing the neutrons down. Therefore, the part of the fission neutrons which escaped from the fuel region, is coming back with lower energy and their probability to cause fission is higher. From the point of view of uncertainty calculations, the orientation of a sensitivity profile is not crucial, but in a case of optimization, interface effects between fuel and reflector or reactivity effects play a vital role.

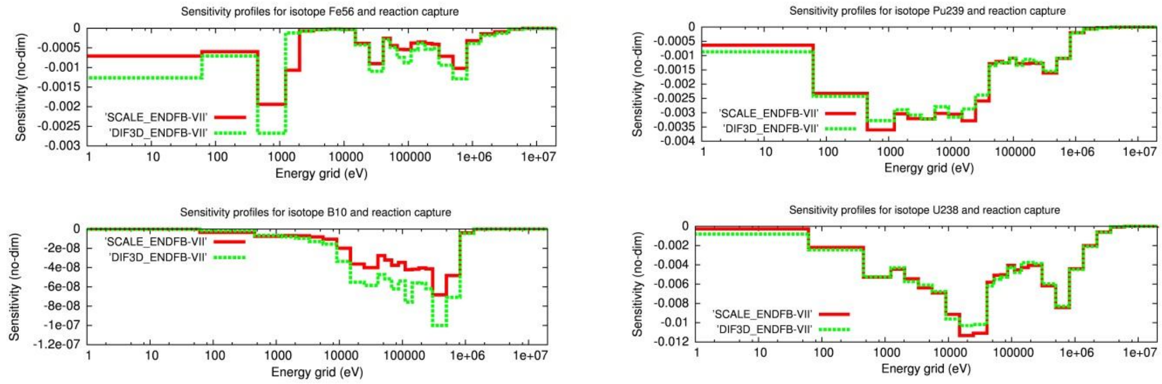
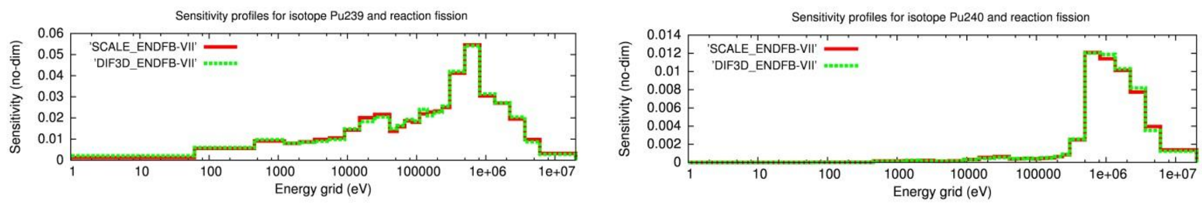


Fig.3: Sensitivity profiles for reaction capture and chosen nuclides

Another effect which is also important in the frame of core design and operation is the leakage due to neutron capture. The sensitivity profiles for capture reaction and fissile nuclides (Pu239 and U238) are presented in the right side of Fig.3. The sensitivity profile for U238 reaches the highest absolute value in the resonance region, between energy 1 to 100 keV, where the highest depression is shifted to the right part. Small depression can be also observed close to energy 1 MeV. This second (reverse) peak effect probably relates to the energy distribution of neutron flux, which has maximum in this energy range. The second fissile nuclide, shown in Fig.3, is Pu239. The absolute values of the sensitivity coefficient are two times lower, but the effective region of the profile is shifted to lower energies, where it is comparable to the sensitivity profile of U238. The sensitivity profiles for capture reaction and nuclides B10 and Fe56, presented in left side of Fig.3, demonstrate an influence of parasitic absorption in structural materials and safety devices. During the flux calculation, the control and safety rods were in the upper position, since the sensitivity coefficient for capture on B10 were small and the effective area of the profile is shifted to the higher energies. For correct evaluation of B10 capture sensitivity, it is necessary to perform sensitivity analyses for a system with partially or fully inserted control and safety rods.



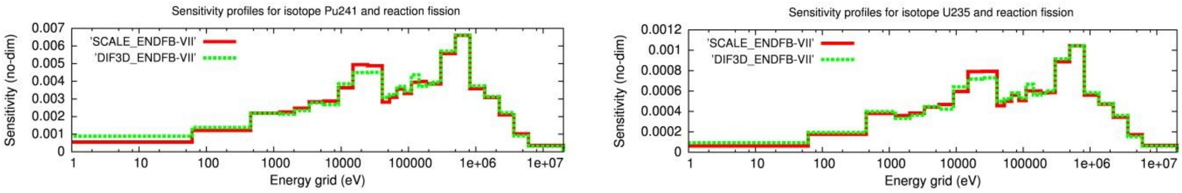


Fig.4: Sensitivity profiles for reaction fission and chosen nuclides

The last part of the sensitivity analyses is aimed on fission reactions. The sensitivity profiles for fission and for four main fissile nuclides (Pu239, Pu240, Pu241 and U235) are shown in Fig.4. A good consistency of sensitivity profiles with neutron flux energy spectra was achieved, except the case of Pu240 sensitivity profile. The presented shape of Pu240 profile is related to the shape of Pu240 cross section, which indicates appropriate application of SPT methodology.

Finally, the sensitivity profiles obtained from TSUNAMI-3D code were used to calculate uncertainty of k_{eff} due to cross section covariance data. Overall k_{eff} uncertainty reaches value of 1.04% which is in a good accordance with previous results presented in [7].The main contributors, in terms of nuclide and reaction via appropriate sensitivity profile, to the total uncertainty are in correlation with the list of most sensitive nuclides.

4. Conclusion

The sensitivity analysis of the ALLEGRO MOX core was performed by using two different computational tools. The correctness of integral sensitivity coefficients was investigated by DP calculation for TSUNAMI-3D sensitivity coefficients, where the satisfactory conformity was achieved. The comparison of integral sensitivity coefficients calculated by TSUNAMI-3D and PORK code identified some discrepancies for elastic scattering reaction which was also confirmed by visual comparison of corresponding sensitivity profiles. Other sensitivity coefficients and profiles show a good agreement for both codes and can serve as a base for following analyses. Finally the overall uncertainty of k_{eff} was calculated consistent with previous results.

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