Low-void-effect sodium-cooled core: Uncertainty of local sodium void reactivity as a result of nuclear data uncertainties

Sandro Pelloni[1], Anne-Laurène Panadero[1], Konstantin Mikityuk[1], Andreas Pautz[1]

[1]Paul Scherrer Institute, Nuclear Energy and Safety Department, Laboratory for Reactor Physics and Systems Behaviour, 5232 Villigen PSI, Switzerland

E-mail contact of main author: sandro.pelloni@psi.ch

Abstract. For one of the French design Na-cooled Fast Reactor (SFR) low void effect cores, regional reactivity effects due to coolant density reductions have been analyzed primarily in terms of their uncertainties due to nuclear data uncertainties. In particular it turns out that the uncertainty of the void effect may result large especially in relative terms. These circumstances occur in hypothetical scenarios in which Na boiling would simultaneously take place in lower portions of the plenum region in addition to subjacent active fuel. In the specific case of fresh fuel compositions the main contributors to the uncertainties are ^{238}U, especially resulting from the inelastic scattering cross-section, as well as ^{23}Na elastic and inelastic scattering cross-sections.

Key Words: SFR low void effect core; void reactivity; void effect uncertainty due to nuclear data uncertainties.

1. Introduction

Safety robustness by all means is still an open issue for Generation IV Sodium-cooled Fast Reactors (SFR) which needs to be addressed and demonstrated with respect to the prevention and mitigation of severe accidents. For reliable safety-related investigations it is not only important determining accurate 3D maps of reactivity coefficients [1], [2] which can then be used in transient analyzes within a point kinetics code; it is equally paramount assessing associated uncertainties resulting from nuclear data uncertainties. In view of the final design these uncertainties would need to be propagated to the overall transient behavior together with uncertainties from other sources. It is in this framework, since it looks like that Na boiling a priori cannot be completely excluded by means of safety measures alone, that for one of the promising French SFR low void effect cores [3], [4] with the preliminary assumption of fresh fuel compositions, the uncertainty of regional reactivity effects resulting from coolant density reductions has been determined, including the void effect in these regions. In the analysis ERANOS (Edition 2.2-N) [5] has been used in conjunction with JEFF-3.1 cross-sections and COMMARA-2.0 variance/covariance data in 33 neutron energy groups which is an ENDF/B-VII.0 based library. More specifically, the COMMARA-2.0 covariance library has been jointly developed over a period of three years from 2008 to 2010 between the Brookhaven and Los Alamos National Laboratory in the framework of applications of the so called Advanced Fuel Cycle Initiative. It contains variance/covariance data for 110 materials relevant to fast reactor research and development, including 12 light nuclides, 78 structural materials and fission products, and 20 actinides. The multi-group data has been obtained by means of the processing code NJOY on the basis of the $1/E$ weighting function [6].

The current simulations were based on standard options for typical sodium-cooled configurations including blankets and reflectors. Among other things they have included cell calculations at the individual pin and subassembly lattice level carried out with the ECCO cell code by means of collision probabilities using 1968 fine groups in conjunction with probability tables suited for the subgroup method. The energy collapsing to the 33 broad group structure occurred by means of the B_1 fundamental mode method in conjunction with the consistent option for fuel lattices, and using an external source from the fuel region otherwise. Effective multiplication factors determining reactivity effects, as well as the sensitivity coefficients required for uncertainty assessments were obtained by means of the nodal code VARIANT [5] applied to a detailed 3D model of the full heterogeneous core, by using Equivalent Generalized Perturbation Theory (EGPT) [7] on the basis of diffusion-theory and the previously mentioned 33 broad group energy structure. In EGPT [7] the Generalized Perturbation Theory (GPT) forward and adjoint solutions [7] are replaced by solutions of homogeneous equations applying Standard Perturbation Theory (SPT) which is perturbation theory for the effective multiplication factor at the two states determining the reactivity effect. The uncertainties due to nuclear data uncertainties were then computed by means of the so called “sandwich” rule resulting from the assumption of normal distributions of the nuclear data in conjunction with linear approximations [8]. It is worthwhile emphasizing that properly using a nodal code in conjunction with sensitivity analysis modules being developed in view of finite difference, discrete-ordinates methods requires particularly care in reconstructing direct and adjoint fluxes in both the radial and axial directions. Namely only under the condition of detailed reconstructions of the directional fluxes from the nodal fluxes the reactivity effects obtained on the basis of perturbed and reference multiplication factors stemming from the nodal code could be reproduced by corresponding values resulting from using exact SPT [7], thus indicating that in this case the nodal method based EGPT sensitivity coefficients are more reliable.

Section 2 illustrates the different current scenarios. Section 3 summarizes the most important results and some conclusions are drawn in Section 4. The specific use of a well-established deterministic code system such as ERANOS finds its main justification in supporting parallel and broader investigations carried out with newly developed stochastic techniques and different variance/covariance data [9].

2. Core configuration and coolant density reduction scenarios

The specifications of the investigated 1500MW$_{th}$ oxide-fuel core correspond to the recent European benchmark exercise which has been launched within the Work Package on Core Safety of the EU FP7 cross-cutting project supporting the European Sustainable Industrial Initiative (ESNII) named ESNII+ [10]. More precisely, the ASTRID-like core assumed as the reference for this study presents a highly heterogeneous design: it features several driver zones combined with an inner fertile zone and a large sodium plenum above aimed at maximizing neutron leakage effects so as to achieve a negative overall coolant void worth. It is here recalled that the benchmark was thus set up for a typical SFR low void effect core primarily in view of estimating the current calculation spread of static parameters resulting from the use of modern methods and nuclear data libraries, with particular emphasis on reactivity coefficients. More precisely the investigated core is composed of 291 hexagonal wrapped fuel subassemblies grouped into two radial zones, *FIG. 1*, which differ for both plutonium content and axial zoning. The inner zone is composed of 177 subassemblies with a total active height of 1.1m including a 0.2m thick internal axial blanket in order to limit the power peaking; the latter is not foreseen in the outer zone, which counts 114 subassemblies with a total active height of 1.2m. A 30cm thick axial blanket is incorporated below the entire fissile region comprising inner and outer zone. Each subassembly contains 217 pins in a triangular arrangement. Regulation, compensation and safety functions are respectively handled by 12 control rod (CSD) and 6 dedicated safety rod (DSD) subassemblies all assumed at their parking positions, in addition to 4 diluent subassemblies located in the inner core region. Three rings of radial reflector subassemblies and four rows of radial shielding subassemblies surround the active core. In order to enable easier understanding, the nine original benchmark scenarios forming the basis of the current investigations, Section 3, are shown in *FIG. 2* with layouts corresponding to the core indicated in *FIG. 1*, more precisely including the upper sodium plenum and excluding fission gas plenum at the bottom, reflector, shielding and control rods. Correspondingly, all these hypothetical scenarios, in which the coolant density has not been modified in the channels of the control rod and diluent subassemblies are thereafter clearly characterized by TABLE I.

The sodium void worth is defined by the reactivity change between the sodium voided and nominal states; for converting the analytical pcm[1] values in dollars ($), a fixed effective delayed neutron fraction of 1$ corresponding to 350pcm in compliance with average benchmark values, Table VI of [10], has been assumed without accounting for uncertainties.

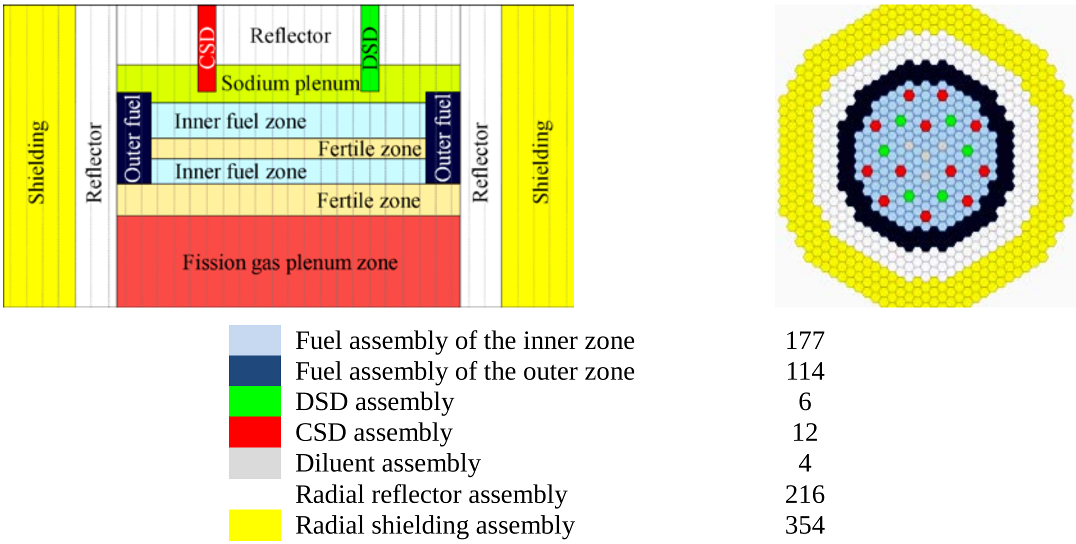


FIG. 1.ASTRID-like core map; figure taken from [10].

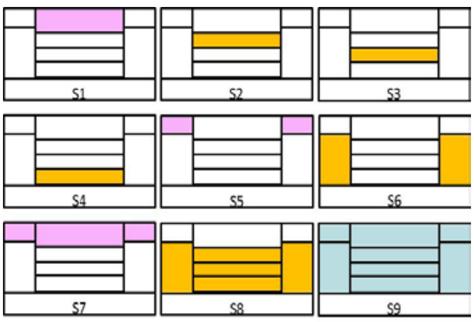


FIG. 2. Schematic representation of various scenarios: light violet refers to sodium plenum zones, amber to fuel zones; figure taken from [10].

[1] 1pcm = 0.00001

3. Results

This section summarizes key data for each of the nine cases in addition to an illustrative scenario resulting in particularly large uncertainties of the void reactivity in both absolute and relative terms.

TABLE II and TABLE III respectively give for the six independent scenarios (S1-S6) and for the three lumped scenarios (S7-S9) the void reactivity in terms of $\Delta\rho = \rho_{void} - \rho_{ref}$ along with the associated one standard deviation (1σ-) uncertainty due to nuclear data uncertainties, expressed in relative as well as in absolute terms. Thereby, ρ_{void} and ρ_{ref} are respectively the reactivity for the voided and reference situation.

TABLE I: SCENARIOS FOR COOLANT DENSITY REDUCTIONS.

Case	Affected regions, see *FIG. 1*
S1	All zones above the inner fissile zone
S2	Upper fissile region of the inner zone
S3	Inner fertile region of the inner zone
S4	Lower fissile region of the inner zone
S5	All zones above the outer fissile zone
S6	Fissile region of the outer zone
S7	All zones above the inner and outer fissile zones, thus corresponding to S1 + S5
S8	Lower and upper fissile and inner fertile region of the inner zone; fissile region of the outer zone, thus corresponding to S2 +S3 + S4 + S6
S9	All fuel regions and all zones above them, thus corresponding to S7 + S8 or to S1 + S2 + S3 + S4 + S5 + S6

3.1.Reactivity effects along with uncertainties due to nuclear data uncertainties

The reactivity effect resulting from voiding the plenum regions (S1, S5 and the lumped scenario S7) is negative; while voiding just the fuel regions (S2, S3, S4, S6 and S8) systematically results in reactivity increases. It is also confirmed that the total voiding scenario (S9) involving both fuel and plenum regions, results in a negative effect. Due the diffusion-theory approximation and consistently with previous independent similar studies [11], however, it is acknowledged that the predicted effect is stronger as compared to the originally provided benchmark solutions which were obtained on the basis of either deterministic transport-theory or stochastic methods [10]; the overall void reactivity is characterized in absolute terms by a larger uncertainty as compared to the partial voiding scenarios, which is of the order of half a dollar corresponding to nearly 10% of the effect. Keeping in mind the different variance/covariance data in use, consistence is shown with independent similar studies [9].

TABLE II: VOID REACTIVITY ALONG WITH 1σ-UNCERTAINTIES DUE TO NUCLEAR DATA UNCERTAINTIES.

Scenario	S1	S2	S3	S4	S5	S6
$\Delta\rho$ (%$)	-508.2	105.6	70.9	42.6	-211.0	43.4
Relative uncertainty (%)	2.4	9.8	8.8	10.2	2.7	19.4
Absolute uncertainty (%$)	12.2[a]	10.3	6.2	4.3	5.6	8.4

[a] $12.2 = 508.2 \times 0.024$

TABLE III: VOID REACTIVITY ALONG WITH 1σ-UNCERTAINTIES DUE TO NUCLEAR DATA UNCERTAINTIES.

Scenario	S7	S8	S9
$\Delta\rho$ (%$)	-604.0	268.2	-383.5
Relative uncertainty (%)	2.4	10.7	11.3
Absolute uncertainty (%$)	14.6	28.8	43.4

TABLE IV correspondingly gives the coolant density reactivity defined in an analogous manner along with its uncertainty.

One interesting remark is that in contrast to $\Delta\rho$, the absolute uncertainty of the coolant density effect which is smaller as compared to the void effect, shows an almost linear dependence with respect to the Na mass removed; whereas the uncertainty in relative terms is seen to slightly decrease with increased mass removal.

By looking back on the whole at the previous two tables, TABLE II and TABLE III, one may infer a similar adding trend for the void reactivity uncertainty as regards space: e.g. the summed-up absolute uncertainty of the six independent scenarios, TABLE I, is 47.0 cents which is comparable to the S9 value of 43.4 cents (%$); while the corresponding void reactivity values are respectively -456.7 and -383.5 cents.

TABLE IV: S9 COOLANT DENSITY REACTIVITY ALONG WITH 1σ-UNCERTAINTIES DUE TO NUCLEAR DATA UNCERTAINTIES.

Na density reduction (%)	5	10	20	100 (void)
$\Delta\rho$ (%$)	-8.7	-18.2	-39.3	-383.5
Relative uncertainty (%)	20.7	20.0	18.8	11.3
Absolute uncertainty (%$)	1.8	3.6	7.4	43.4

3.2.Nuclide dependent contributions to the total uncertainty

TABLE V gives for the nine scenarios the decomposition of the total uncertainty of the void reactivity expressed in terms of important contributions from individual nuclides.

TABLE V: ISOTOPIC CONTRIBUTIONS (%$) TO THE 1σ-UNCERTAINTY OF THE VOID REACTIVITY.

Scenario	S1	S2	S3	S4	S5	S6	S7	S8	S9
^{238}U	10.8	7.1	4.1	3.2	5.2	5.3	12.9	19.5	34.0
^{23}Na	4.9	5.9	4.2	2.3	1.3	5.6	5.8	17.9	23.5
^{16}O	1.4	1.6	1.0	0.8	0.6	1.1	1.6	4.6	5.8
^{56}Fe	1.3	1.9	1.1	0.7	0.8	0.8	1.5	4.5	5.5
^{239}Pu	1.2	2.6	0.9	1.0	0.6	2.2	1.4	6.7	7.3
^{240}Pu	1.5	1.9	0.3	0.6	0.8	1.3	1.8	4.0	5.7

It can be observed that in this specific case of fresh fuel compositions ^{238}U and ^{23}Na play systematically the most important role as regards uncertainties, independently of whether sodium boiling just occurs in either the fuel or plenum region or in both regions. The thumb rule of additivity of the absolute uncertainty in space, previous subsection, is largely confirmed at the level of the individual nuclides as it can be verified e.g. by summing-up the S7 and S8 data and comparing the results with corresponding S9 data. Similar qualitative considerations would also apply to the coolant density reactivity for which the uncertainties are certainly weaker.

While for ^{238}U the uncertainty of the void reactivity is found to be largely contributed by the inelastic scattering cross-section, both the elastic and inelastic scattering cross-sections appear equally important for Na, TABLE VI. More precisely, elastic scattering is dominant in the plenum voiding scenarios in which leakage effects are extremely important, whereas the two components are more balanced when fuel regions are voided.

The energy decomposition of the void effect uncertainty due to the two nuclides is displayed in *FIGS 3-4* for the three lumped scenarios.

TABLE VI: REACTION CONTRIBUTIONS (%$) TO THE
1σ-UNCERTAINTY OF THE VOID REACTIVITY DUE TO SODIUM.

Scenario	Na-plenum			Active fuel					Both
	S1	S5	S7	S2	S3	S4	S6	S8	S9
Elastic	4.5	1.2	5.3	3.5	3.6	1.4	3.5	11.7	16.7
Inelastic	2.0	0.6	2.3	4.8	2.2	1.8	4.4	13.5	16.5

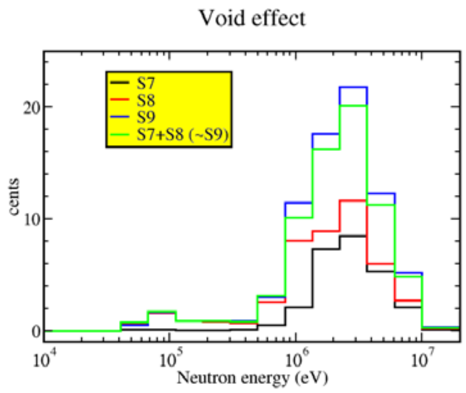
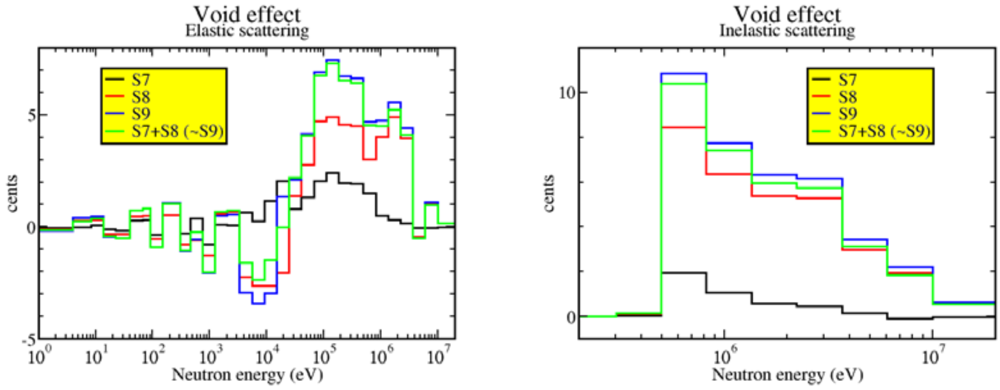


FIG. 3. Energy decomposition of the void effect uncertainty due to ^{238}U inelastic scattering.

As regards ^{238}U, *FIG. 3*, larger uncertainties are resulting in the fission source range, however with a distribution peaked at higher energies as compared to a typical fission spectrum corresponding to stronger sensitivity coefficients of the void effect to the ^{238}U inelastic scattering cross-section. As one may expect, these uncertainties are more pronounced in the two scenarios in which the active fuel containing this nuclide is voided; compare S8 with S7; whereas the overall energy dependency of the uncertainty appears largely independent of the scenario consistently with the rather constant COMMARA-2.0 standard deviation of the ^{238}U inelastic scattering data of the order of 20%-30% in the energy range of interest.

But also in the case of ^{23}Na the uncertainty appears larger for the two scenarios involving active fuel voiding i.e. S8 and S9. In the case of elastic scattering, *FIG. 4 (a)*, the most important energy region lies above 100keV where the COMMARA-2.0 standard deviation reaches 10%. The negative values showing up below this limit and referring to negative variances, which correspondingly lead to a slight reduction of the total uncertainty, are the result of specific cross-correlations available in the COMMARA-2.0 library which are particularly large between ^{238}U reactions. In the case of inelastic scattering, *FIG. 4 (b)*, the most important contributions to the total uncertainty are resulting in the fission source range with a fission spectrum like distribution. At these energies the COMMARA-2.0 standard deviation reaches 15% and the sensitivity coefficients of the void effect to the ^{23}Na inelastic scattering cross-section become larger.



(a) *(b)*

FIG. 4. Energy decomposition of the void effect uncertainty due to ^{23}Na (a) elastic, (b) inelastic scattering.

The additivity with respect to space, which is reflected in the summation plots "S7+S8" reproducing well the curves referring to S9, is largely confirmed.

3.3.Further investigations

An additional, particularly interesting scenario based upon S9 has been analysed. Instead of the entire Na plenum zone, *FIG. 1*, just a variably thick portion located above the active fuel is voided, while the remaining part of the scenario is consistent with S9 except that the regions above the plenum are not voided. Correspondingly, the given non-voided upper part length of the Na plenum referring to the axial direction, TABLE VII, is measured downwards from the top to the bottom of the plenum zone, the top height being the same for the inner and outer plenum region, *FIG. 1*. On the one hand the limiting length of zero, in which case the whole plenum zone is voided together with all fuel regions, would correspond to S9 except that the zones above the plenum are not voided; on the other hand the length of 30cm indicates that the voided volume of the inner core plenum corresponds to the first 10cm above the fuel since the inner core plenum region has a total thickness of 40cm [10], while the outer core plenum region starting 10cm higher as well as the regions above remain unperturbed, *FIG. 1*.

TABLE VII: VOID REACIVITY FOR S9 BASED SCENARIO ALONG WITH 1σ-UNCERTAINTIES DUE TO NUCLEAR DATA UNCERTAINTIES.

Non-voided upper part length of the Na plenum (cm)	$\Delta\rho$ (%$)	Relative uncertainty (%)	Absolute uncertainty (%$)
20	-134.6	27.9	37.5
28	-6.1[a]	568.7	34.7
29	12.6	271.9	34.3
30	32.4	104.8	34.0

[a] $\Delta\rho = (-6.1 \pm 34.7)\%\$$

It can easily be deduced that (1) as one may a priori expect the void reactivity dependency is nonlinear with respect to this length; (2) on the contrary, the absolute uncertainty of the void reactivity shows basically a linear dependency with a slight slope of approximately -0.4cents/cm; (3) despite this just weak dependency of the absolute uncertainty on the length, meaning that the absolute uncertainty in any case is of the order of one third of a dollar, the relative uncertainty grows fast with decreasing strength of the central value of the reactivity effect. This circumstance is not too surprising since the smaller is the achieved net void reactivity depending upon the specific length, the larger are compensations resulting from a positive contribution due to voiding the fuel region alone, a negative contribution due to voiding the plenum regions, and an additional unknown interference effect getting stronger, with the former two effects having individually considerable absolute uncertainties, TABLE III; (4) as regards voiding the outer plenum region within this scenario, when sodium boiling occurs simultaneously in a layer of just a few cm thickness above the active fuel, the sign of the analytical value of the void effect becomes questionable since the relative uncertainty may largely and easily exceed 100%. For example in order to be more specific, the second row of TABLE VII indicates according to footnote [a], of course by assuming normal distributions which is the main criterion for the validity of the sandwich rule for estimating uncertainties, an uncertainty >>100% corresponding to a 1σ interval covering negative as well as positive values between -40.8 (= -6.1 - 34.7) and 28.6 (= -6.1 + 34.7) cents (%$) with a negative expectation or central value of -6.1 cents.

Of course additional scenarios with markedly large relative uncertainties of the void reactivity could easily be conceived in a similar manner. From the previous thoughts it is sufficient that the reactivity effect characterizing the scenario would result from voiding a combination of fuel and plenum regions in such a way that the net analytical reactivity effect is sufficiently small whatever it means.

4. Conclusions

The Na void reactivity together with its uncertainty due to nuclear data uncertainties has been investigated for a 1500MW$_{th}$ oxide-fuel low void effect, ASTRID-like core corresponding to the recent European benchmark exercise within the Work Package on Core Safety of the EU FP7 cross-cutting project. The described work making use of a well-established deterministic code system which is ERANOS [5], was primarily undertaken with the aim of supporting parallel and broader investigations on the basis of newly developed stochastic techniques and different variance/covariance data [9], confirming some independent findings as regards complete voiding scenarios.

Moreover the new study in particular shows based on the specific use of COMMARA-2.0 variance/covariance data that in absolute terms the uncertainty of the coolant density reactivity due to nuclear data uncertainties is almost linear with respect to the Na mass removed as a consequence of temperature variations; while the uncertainty of both the coolant density and void reactivity is approximately adding in space when considering standard scenarios. In fact this property would allow reconstructing more global values from detailed uncertainty maps without the need of performing extra calculations.

It has also been pointed out, Section 3, that it is possible to conceive scenarios suffering from particularly large void reactivity uncertainties especially in relative terms. Such scenarios envisaging simultaneous local boiling in the upper part of some fuel pins and in the corresponding lower plenum region just above these pins, currently form an important topic in view of detailed analyzes and verifications of the coupling between neutronics, thermal-hydraulics and fuel behavior.

5. Acknowledgments

The authors are thankful to A. Vasile and N. Dévictor from CEA for their support in providing the necessary specifications.

6. References

[1] PELLONI, S., et al., “Sodium Void Map Preparation for the Safety Analysis of Sodium-cooled Fast Reactors by Using the Monte Carlo Code Serpent” (Proc. ICAPP 2015), Nice (2015); Contributed papers (CD-ROM).

[2] KREPEL, J., et al., “Mapping of Sodium Void Worth and Doppler Effect for Sodium-cooled Fast Reactor” (Proc. ICAPP 2015), Nice (2015); Contributed papers (CD-ROM).

[3] VARAINE, F., et al., “Pre-conceptual design study of ASTRID core” (Proc. ICAPP 2012), Chicago (2012); Contributed papers (CD-ROM).

[4] SCIORA, P., et al., “Low void effect core design applied on 2400 MWth SFR reactor” (Proc. ICAPP 2011), Nice (2011); Contributed papers (CD-ROM).

[5] RIMPAULT, G., et al., “The ERANOS data and code system for fast reactor neutronic analyses” (Proc. PHYSOR 2002), Seoul (2002); Contributed papers (CD-ROM).

[6] HERMANN, M., et al., “COMMARA-2.0 Neutron Cross Section Covariance Library”, BNL-94830-2011, U.S. Department of Energy, (2011).

[7] WILLIAMS, M. L., “Sensitivity and Uncertainty Analysis for Eigenvalue-Difference Responses”, Nuclear Science and Engineering **155** (2007) 18.

[8] Handbook of Nuclear Engineering, edited by CACUCI, D.G., ISBN 978-0-387-98149-9 (Online), 1913.

[9] GARCIA-HERRANZ, N., et al., “Nuclear data sensitivity and uncertainty assessment of sodium voiding reactivity coefficients of an ASTRID-like Sodium Fast Reactor” (Proc. ND 2016), Bruges (2016); Contributed papers (CD-ROM).

[10] BORTOT, S., et al., “European benchmark on the ASTRID-like low-void-effect core”, (Proc. ICAPP 2015), Nice (2015); Contributed papers (CD-ROM).

[11] BORTOT, S., et al., “Methodology assessment for the evaluation of the coolant void worth in sodium fast reactors with a low void effect core design”, (PROC. PHYSOR 2014), Kyoto (2014); Contributed papers (CD-ROM).