Доклади на Българската академия на науките
Comptes rendus de l'Académie bulgare des Sciences
Tome 71, No 3, 2018

PHYSICS
Nuclear reactor physics

SIMULATION OF A ROD EJECTION TRANSIENT IN VVER-1000 WITH COBAYA4-CTF

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(*Submitted by Corresponding Member Ch. Stoyanov on December 21, 2017*)

Abstract

The objective of this work is to analyze hypothetical reactivity insertion accidents including control rod ejection in a VVER-1000 V320 reactor using the coupled best-estimate COBAYA4/CTF neutronics/thermal hydraulics codes. A specific objective is to explore the adequacy of two-group diffusion cross-section libraries of different detail for such simulations. A full-core COBAYA4/CTF nodal calculation model was verified in code-to-code comparison with COBAYA3/FLICA4 results in steady state and control rod withdrawal simulation. This model was used to analyze a rod ejection transient from hot zero power. The results show that the computed local parameters do not exceed the safety limits. The impact of cross-section libraries on the inserted reactivity and transient peak power was assessed by comparing calculations with two variants of the table-interpolation cross-section library. The first one is simplified, as used in some OECD/NEA benchmarks, and the second is full-scope in general multi-group format. The results show that a library in general multi-group format with feedback-dependent kinetic parameters is the reasonable choice for the analysis of such transients.

Key words: rod ejection transient, 3D core simulation, coupled neutronics-thermal hydraulics, nodal level, VVER-1000

1. Introduction. The simulation of control rod ejection accidents (REA) is an important part of the nuclear reactor safety analysis which requires advanced 3D core neutronic and thermal-hydraulic (N-TH) models, as well as accurate cross-section libraries. Modelling of such transients from hot zero power (HZP) with super prompt-critical reactivity insertion is particularly challenging

DOI:10.7546/CRABS.2018.03.03

because of the following features: (a) Strong and fast 3D flux redistribution; (b) Delays in the actuation of the thermal feedbacks depending on the modelling of fuel heat capacity, conductivity, fuel-clad gap conductance and the clad to coolant heat transfer; (c) Sensitivity to the neutron flux calculation model and mesh refinement, as well as to the scope and quality of the cross-section library. Currently, most of the REA analyses rely on assembly averaged power from whole-core nodal neutronics models coupled with 3D or multi-channel thermal-hydraulic models. The associated uncertainties (other than those in evaluated nuclear data) can be reduced by using advanced models, node subdivision and accurate cross-section libraries.

Previous publications on 3D VVER-1000 REA simulation such as [1–6] have focused mainly on code testing for methodological purposes. Most of them [1–4] consider REA from hot power where the ejected rod worth is relatively small and the fuel is initially very hot, so that the Doppler feedback actuates immediately. In the present study we will concentrate on the simulation of REA from HZP. Before proceeding with the discussion of our results we will briefly recall some of the related works on REA from HZP.

Knight et al. [5] reported a comparison of computed results for a VVER-1000 hexagonal-z geometry version of the NEACRP rod ejection benchmark. The nuclear and thermal properties of the fuel, reflector and control rod (CR) assemblies were taken to be the same as in the NEACRP PWR benchmark, and the geometry, coolant flow rate, rated power and other parameters were taken as in VVER-1000. Two-group calculation results from the PANTHER, DYN3D-HEXNEM1 and HEXTIME nodal diffusion models [5], coupled with coarse-mesh core TH models were compared. This study illustrated the order of magnitude of the power excursion for different reactivity insertions and the impact of the various neutron flux models on the results, when using the same cross-section library. For a peripheral CR ejection from HZP the inserted reactivity was in the range of 1.48–1.52 β and the computed peak transient power was in the range of 10500–14640 MW.

Avvakumov et al. [6] reported a 4-group pin-by-pin rod ejection calculation from HZP with the BARS code coupled to RELAP5 for a South Ukrainian VVER-1000 V320 core. The BARS code used a Green's function based expansion to model the pin-cell flux. The reference core was at the end of Cycle 3, with 21 MWd/kg average burn-up. A peripheral CR was ejected. The computed inserted reactivity was 1.21 β and the peak power reached 113.5% of the rated power.

Sanchez-Cervera et al. [7] published a study of a cross-section library related aspect of the PWR rod ejection simulation from HZP. The authors explored the impact of the parameter grid optimization [8] on the computed results for the OECD/NEA PWR MOX rod ejection benchmark. They compared COBAYA3/COBRA-TF simulation results at the nodal level obtained with two versions of a multi-parameter XS library. These versions were generated using the same calculation scheme, with or without optimization of the parameter grid for branch calculations. The grid was optimized so that to reduce the linear interpolation error and keep the error in k-eff below a desired value throughout the whole parameter range. The effect of this optimization on the results in [7] was a 7% lower power excursion compared to that without optimization.

In this work we consider two hypothetical transient scenarios:

- uncontrolled withdrawal of Rod bank #9 at 2 cm/s from fully inserted position at HZP;
- ejection of two peripheral CR in assemblies #91 and #117 from fully inserted position at HZP, in 0.1 s, with a nearby peripheral CR in assembly #140 which remains stuck out of the core during the reactor shutdown (scram).

The task is to solve core boundary condition N-TH problems. The associated small break LOCA aspect and thermo-mechanical aspects of CR ejection are beyond the scope of this analysis. The core specifications are taken from the OECD VVER-1000 MSLB benchmark [9]. The reference core [9,15] is a three-batch loading for Kozloduy-6, Cycle 8 at 270.4 EFPD, near the end of life and contains once, twice and three times burnt assemblies of 4.23 w/o and 4.4 w/o initial enrichment. Assemblies #117 and #91 with the ejected CR are at 31 and 38 MWd/kgHM, respectively. The initial HZP state corresponds to core inlet temperature of 278.25 °C, mass flow rate 17 215 kg/s, core exit pressure 15.7 MPa and reactor power of 3000 W. Rod banks #1–5 are out of the core, #7–10 are fully inserted and #6 is 81% withdrawn. Criticality is achieved through a correction of the production operator.

For the purposes of this study COBAYA4/CTF coupled code solutions were verified in code-to-code comparison vs. COBAYA3/FLICA4 calculations for steady state and CR bank withdrawal at operating speed of 2 cm/s. The utilized codes, couplings and core models are discussed in Section 2 below. The tested COBAYA4/CTF nodal core model was used to simulate a postulated rod ejection transient from HZP in VVER-1000. A specific objective was to compare the performance of two diffusion cross-section libraries of different detail in REA simulation. The libraries are in two energy groups and have been generated with APOLLO2 [11]. Both libraries use linear table interpolation and the same parameter grid, optimized as described in [8] to keep the error in k-eff below a desired threshold. The first library is in compact format, as used in some OECD/NEA benchmarks: with up-scattering correction, node-specific composition-dependent kinetic parameters and interface discontinuity factors (IDF) implicitly included through the cross-sections. The second one is in general multi-group (MG) format, with full matrix of scattering cross-sections and explicit IDF, and node-specific kinetic parameters which are both composition dependent and thermal feedback dependent.

In the sequel, Section 2 summarizes the methodology. The scenarios and the results are described in Section 3. Conclusions are given in Section 4.

2. Methodology. ***2.1. Calculation models.*** COBAYA4 [13] is a recent version of a multi-scale multi-group 3D core simulator code developed by the Universidad Politecnica de Madrid (UPM). It uses diffusion approximation at the nodal and pin scale in rectangular and hexagonal geometry. The hexagonal nodal flux solver [14] is based on the Analytical Coarse-Mesh Finite-Difference Method (ACMFD). The code is capable of radial nodal mesh refinement to 6 and 24 triangles per hexagon and parallelization. The nodal flux solver has been extensively tested vs. transport and diffusion reference solutions and other code results in refs [10, 14–17].

CTF [18] is an improved version of the COBRA-TF thermal-hydraulic code with sub-channel capabilities. COBRA-TF was originally developed by the Pacific Northwest Laboratory in 1980 and later modified by several organizations. It was improved, updated and consequently rebranded as CTF at PSU, USA. The code is currently maintained by the North Carolina State University and the Oak Ridge National Laboratory, USA. The current version uses a two-fluid, three-field modelling approach. Both sub-channel and 3D Cartesian forms of 9 conservation equations are available for LWR modelling. This code has been used and tested for PWR, VVER and BWR in the EU NURESAFE project [19] and recent US R&D projects.

FLICA4 [20] is a fully 3D core thermal-hydraulic code of CEA with fine-mesh capabilities. The two-phase mixture is modelled by a set of four balance equations: mass, momentum and energy of mixture, and mass of steam. The velocity disequilibrium is taken into account by a drift flux correlation. In the present analysis special care was taken to use modelling assumptions in FLICA4 as close to those in the CTF model as possible.

The COBAYA4/CTF coupling method [21] for VVER-1000 is based on the MED Coupling libraries in Salome 6 (`http://www.salome.com`). The Python script which governs the coupled calculation includes a damping scheme to smooth the power profile and accelerate the convergence in transients. The COBAYA3/FLICA4 coupling for VVER-1000 uses the coupling functions in Salome 5 supplemented by FLICA4 routines.

The VVER-1000 core models and code couplings using these tools have already been numerically validated [10, 14–17] for steady state and reactivity transient calculations in the OECD/NEA VVER-1000 MSLB benchmark project [9] and in the EU NURESAFE project [19]. The main modelling assumptions for the present study are summarized below:

- Full-core two-group nodal neutronics;

- Reflector nodes of assembly size considered as diffusive media;
- Spatial mesh of the 3D neutron kinetics with:
  - 30 axial nodes in the heated region and two nodes in each axial reflector,
  - six triangles per hexagon;
- Transport corrections (interface discontinuity factors) for the fuel nodes;
- Zero flux boundary condition on the outer reflector boundary;
- Coarse-mesh thermal-hydraulics with one channel per assembly;
- The spacer grids were not explicitly modelled and were taken into account by the vertical pressure loss coefficients (different in CTF and FLICA);
- Unified fuel models in CTF and FLICA4 with 30 axial nodes and radial discretization with nine radial rings in the fuel, one for the gas gap and one for the cladding;
- Table look-up for the temperature-dependent fuel pin thermal properties;
- Constant fuel gap conductance coefficient of 3070 W/m^2K;
- CTF heat transfer models: Dittus–Boelter for single-phase liquid, Chen's model of nucleate boiling, W-3 general purpose correlation for the critical heat flux (CHF);
- FLICA4 heat transfer models: Dittus–Boelter/Jens–Lottes model of convective heat transfer/nucleate boiling and Groeneveld CHF model in FLICA4.

2.2. Cross-section libraries. A multi-parameter two-group diffusion cross-section library [12] at the nodal level was used. It was generated with APOLLO 2.8 [11] using 281 g for fuel depletion and a two-level 281 g/37 g cross-section calculation scheme with the Linear Surface Method of Characteristics (LS MOC) for branch calculations. The library features:

- JEFF3.1.1 nuclear data;
- Table interpolation format;
- Wide parameter range;
- Optimized parameter grid of five fuel temperature points, three moderator temperature points and 13 moderator density points, such that the error in k-eff is less than 120 pcm in case of linear interpolation [8];

- Generated for 840 fuel compositions in 1/6 3D core (28 assemblies × 30 axial nodes, each with given target exposure);
- Uniform radial, top and bottom reflector compositions;
- Transport corrections (interface discontinuity factors) for the fuel nodes;
- No transport corrections for the reflector nodes.

The two-group cross-section library is available in two formats:

A. Compact two-group format with down-scattering cross-sections corrected for up-scattering; with IDF implicitly included in the cross-sections and node specific composition-dependent kinetic parameters;

B. General multi-group format with full matrix of scattering cross-sections, explicit IDF and nodal kinetic parameters dependent on the composition and the TH feedbacks.

Format B is being used with the COBAYA4 MG flux solver which iterates on the scattering with energy increase in the two-group diffusion calculations.

COBAYA4/CTF calculations of a MSLB transient [17] with the two library options displayed a difference of 2.4% in the total power and 0.9% in Fxyz at the moment of maximum return to power. In REA simulation this difference is more significant as will be seen in the following.

3. Computation results. For verification purposes a COBAYA4/CTF solution for uncontrolled rod bank withdrawal at operating speed of 2 cm/s was compared vs. COBAYA3/FLICA4 results. The tested COBAYA4/CTF calculation model was used to analyze a super prompt-critical rod ejection accident from HZP in VVER-1000. All results were obtained with a time step of 1 ms, after a convergence study. COBAYA4/CTF calculations with the two options of the cross-section library were compared to assess the impact of the library scope and detail on the results.

Figure 1 illustrates the comparison of the COBAYA4/CTF vs. COBAYA3/FLICA4 solutions for the transient core power in case of complete withdrawal of Rod Bank 9 at 2 cm/s, starting from fully inserted position at HZP. The solutions have been obtained with the simplified cross-section library in compact format. The steady state results agree well and the computed rod bank worth is nearly the same, 1071 pcm vs. 1080 pcm. The transient total power results are in good agreement, with only small bias due to differences in the thermal hydraulic models.

Figure 2 shows the computed time history of the total power and reactivity in case of rod ejection from HZP in 0.1 s, in two peripheral assemblies #91 and #117. The results have been obtained with the cross-section library B in

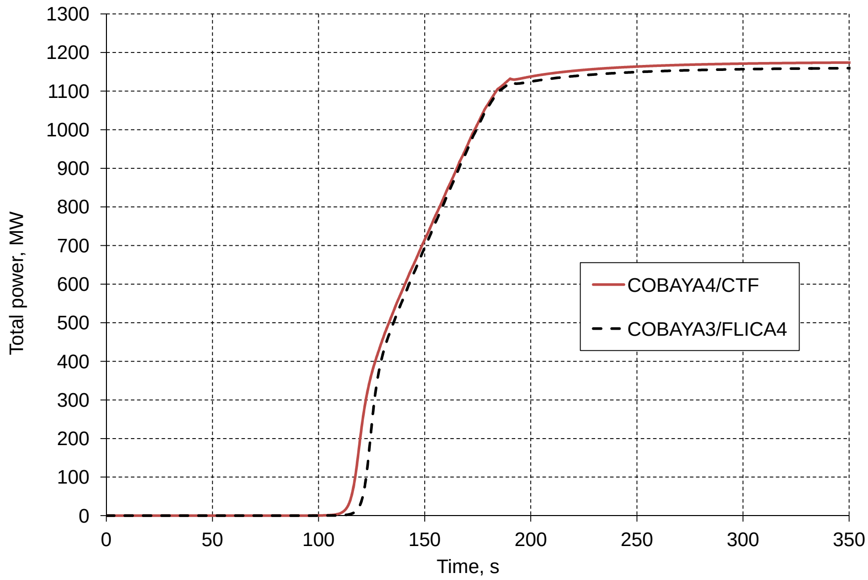

Fig. 1. Withdrawal of CR Group 9 at 2 cm/s: Time history of the total core power (using the XS library A)

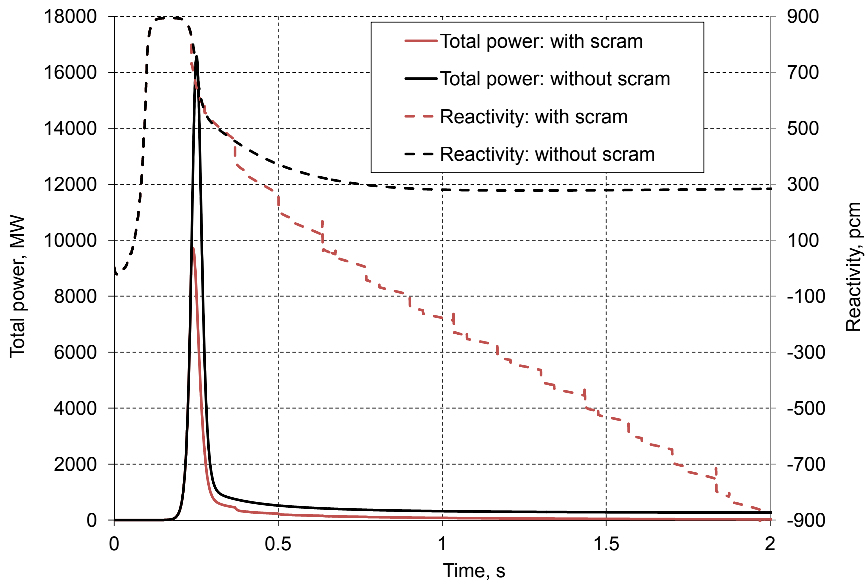

Fig. 2. Ejection of two CR with/without scram: Time history of the core power and reactivity (stretched) obtained with the XS library B

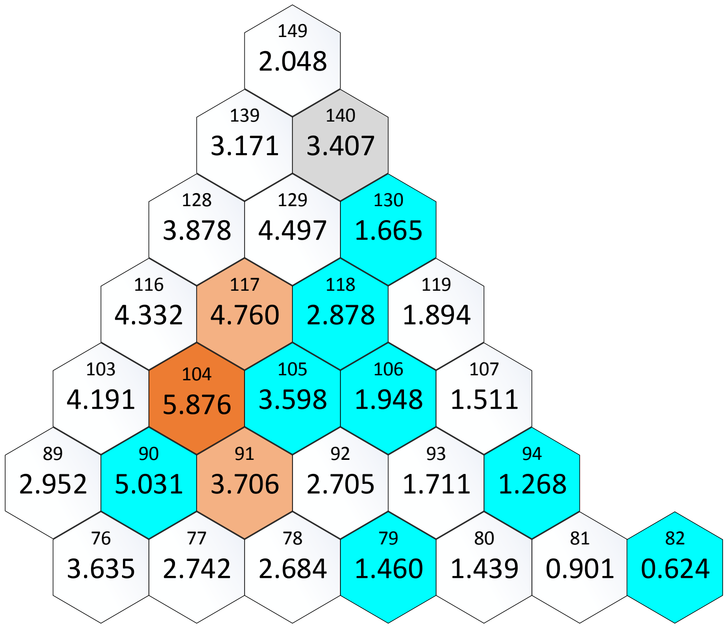

Fig. 3. Ejection of two CR with scram: Relative assembly powers at peak power (using the XS library B)

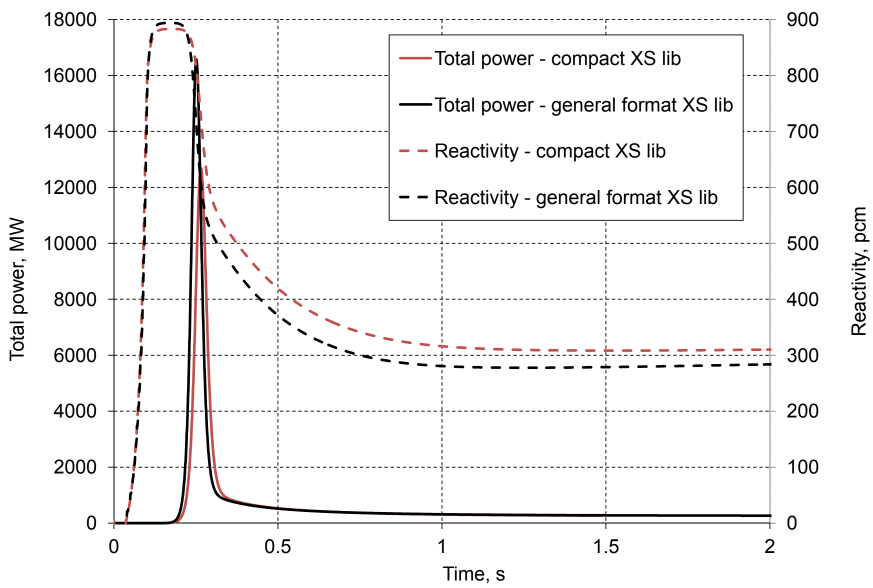

Fig. 4. Ejection of two CR without scram: Time history of the reactivity and total power. Comparison of results with the compact vs. general format XS library

general format, assuming that the scram signal on Period < 10 s occurs at 0.1 s and the delay in scram activation is 0.025 s. The maximum inserted reactivity is 1.58 β (β = 563 pcm at $t = 0$ s). Variants of transients with and without reactor shutdown (scram) are shown. The power peak is 16567 MW without scram and 9720 MW with scram.

Figure 3 shows the core radial power distribution at time of peak power (0.24 s) in case of rod ejection transient with scram. At that time scram is activated and the radial location of the gravitationally dropping CR is marked in blue. The CR in assembly #140 does not trip and remains stuck out of the core. The peak power is 9720 MW and most of the corresponding energy is released in a small number of assemblies around the two ejected rods in assemblies #91 and #117. The hottest assembly is #104 (with a radial peaking factor Fxy of 5.876 and burn-up 15.4 MWd/kgHM) in which the local parameters need to be analyzed. The nodal calculation results show that because of the narrow power peak, the energy release is moderate and the fuel temperature remains within acceptable limits. The integrated core power (energy release) is approx. 600 MJ at 2 s of the transient and tends to stabilize. The coolant flow is single phase and the maximum fuel centre temperature reached in the transient is 520 °C at 0.74 s. This is far from the UO_2 melting point which is approx. 2740 °C for the hot assembly. With such conditions in the hot assembly, sub-channel TH analysis was not performed.

Figure 4 shows the rod ejection results without scram obtained with the two options of the cross-section library. Substantial difference of approx. 24% in the peak power can be seen: 12661 MW with library A vs. 16567 MW with library B. The results suggest that the better choice between the two cross-section library formats for REA analysis is the full-scope multi-group format B which contains more physical information. In this study, with the software tools available to the authors it was not possible to precisely separate all partial effects of the use of library B vs. A, and an approximate analysis was performed. Based on the considerations below the difference in the peak power was attributed to the combined impact of the rigorous scattering treatment and the feedback-dependent inverse neutron velocities ($1/v$).

Concerning the scattering treatment, the COBAYA4 calculation of the initial steady-state at HZP with the two library options shows a difference (A-B) in k-eff of +35 pcm and −1.5% in Fxy, and DYN3D-MG results for the same state [17] give a bias of +10 pcm in k-eff and −1.4% in Fxy.

Regarding the feedback-dependent kinetic parameters, the variation of β during the transient is small and the main contribution to the difference in peak power comes from the $1/v$ variation. In the point kinetics approximation when applied to super prompt critical transients, the derivative of the total neutron power quickly becomes

$$dp/dt \approx ((\rho - \beta)/\Lambda)p(t) \tag{1}$$

and for constant β and Λ the power evolution can be expressed as

$$p(t) = p_0 \exp(\omega_p t), \tag{2}$$

where $\omega_p = (\rho - \beta)/\Lambda$ is the dominant eigenfrequency and Λ is the mean neutron generation time calculated by the lattice code for each fuel node from the inverse neutron velocity weighted in energy and space with the neutron flux and divided by the integral of the fission source. In this study the dynamic reactivity is computed at every time step from the evolution of the total 3D neutron power by inverse point kinetics as the reactivity which in the point kinetics equation would yield the same total power frequency ω_p. When using library B, the approximate assessment of the reactivity from point-kinetics-like equations requires the update of $\Lambda(t)$ and $\beta(t)$ at every time step. Since in the present simulation Λ is not updated, the reactivity obtained with this library is not used for quantitative conclusions.

With library B, when the fuel and the moderator are heating up (including the energy directly released in the moderator) the $1/v$, respectively $\Lambda(t)$ decrease. This gives rise to a higher power excursion compared to that with feedback-independent $1/v$ (and Λ). The $1/v$ decrease is most pronounced in the assemblies around the ejected rods and the local effects strongly influence the 3D core dynamics. The higher peak power causes stronger feedback effects. When the reactivity drops below β it decreases more slowly, with the larger half-life of the delayed neutron precursor group. Table 1 shows the reactivity trend without scram in about 52 s as computed with library A.

Table 1

Time history of reactivity in REA without scram (using library A)

Time, s	0	0.15	1	7	10	22	52
Reactivity, pcm	0	881	316	320	151	96	0

4. Summary and conclusions. A nodal core model of VVER-1000 with the coupled COBAYA4/CTF best-estimate codes was tested and used to analyze control rod withdrawal and REA transients. In this analysis thermo-mechanical aspects were not taken into account, except for data preparation.

The comparison of the COBAYA4/CTF vs. COBAYA3/FLICA4 results for the rod bank withdrawal transient shows good agreement.

For the considered REA scenario a super prompt-critical power excursion is predicted, which is quenched by the fuel Doppler feedback and later by the action of the reactor shutdown system. According to the coarse-mesh results the safety parameters remain well within the admissible limits.

The comparison of the performance of two cross-section libraries with identical optimization of the parameter grid but of different detail shows that the reasonable choice for REA analysis is a general MG format, full-scope cross-section library with feedback-dependent kinetic parameters.

Acknowledgements. This work was partially funded by the EC in the frame of the EU NURESAFE project, contract No 323263. The results were obtained using the COBAYA code developed at UPM Madrid and the CTF code provided by the NCSU, USA.

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