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Stress-testing the ALFRED design – Part I: Impact of nuclear data uncertainties on Design Extension Conditions transients

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ABSTRACT

The advancement of the design of ALFRED beyond the conceptual phase, passes through the analysis of the impact of uncertainties, notably to what concerns safety-related conditions. Compliancy of plant safety to Design Extension Conditions is, according to IAEA and in line with the meaning itself of these beyond-design conditions, usually investigated by best estimates only. Due however to the demonstration nature of ALFRED, it was decided to assess the actual safety performances of this system even in ultimate conditions. To this regard, the emphasis was put on unprotected events like the UTOP (unprotected transient of over-power) and ULOOP (unprotected loss of offsite power, resulting from the combination of a loss of flow and loss of heat sink under unprotected conditions), pinpointed as the most challenging situations sought for the plant. The purpose of the present work, which has been divided in three parts, was then to assess the ultimate ALFRED safety margins against failure of the key core components and systems (Part III). To target this objective, the evaluation of uncertainties coming, on one hand, from nuclear data was performed at first, to retrieve their impact on the reactivity coefficients, thereby on the transient behavior driven by the latter (Part I); then, uncertainties from material properties, fabrication procedures, operation and computational tools were propagated to assess their influence on the thermal-hydraulics of the system (Part II). In this paper, presenting the first part of the work, the focus is on nuclear data. As such, a sensitivity/uncertainty analysis of the ALFRED core on key elementary reactivity effects, forming the basis for computing the feedback coefficients, has been performed. The sensitivity analysis allowed pointing out firstly the most relevant cross-sections for every response function. Uncertainty analysis allowed then establishing a possible range of confidence for the reactivity effects. The adjoint-based technique implemented in the TSUNAMI-3D module of the SCALE6 system was used. The confidence intervals identified for each reactivity effect have been combined then into confidence intervals for the feedback coefficients. Finally, the most conservative, yet physically sound (i.e., where correlations among coefficients, stemming from dependencies on common nuclear data, are taken into account), set of reactivity coefficients has been picked out from the confidence intervals, enabling transient calculations to propagate uncertainties into transient behavior. Using this off-nominal set, and comparing results with the reference ones, exploiting the system codes SIM-LFR and RELAP, a negligible effect of nuclear data uncertainties has been found for the ULOOP, while an increase of the maximum achieved power of around 6% has been computed for the UTOP. Overall, a modest contribution of nuclear data uncertainties for these transients has been found which, however, must be combined with the thermal-hydraulics one so to finally assess safety margins.

1. Introduction

In the 7th Framework Program the European Commission co-founded the Lead-cooled European Advanced DEmonstration Reactor (LEADER) project (De Bruyn et al., 2013) which had, as one of the main objectives, the preliminary design of the Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED), resulting in the configuration reported in Fig. 1.

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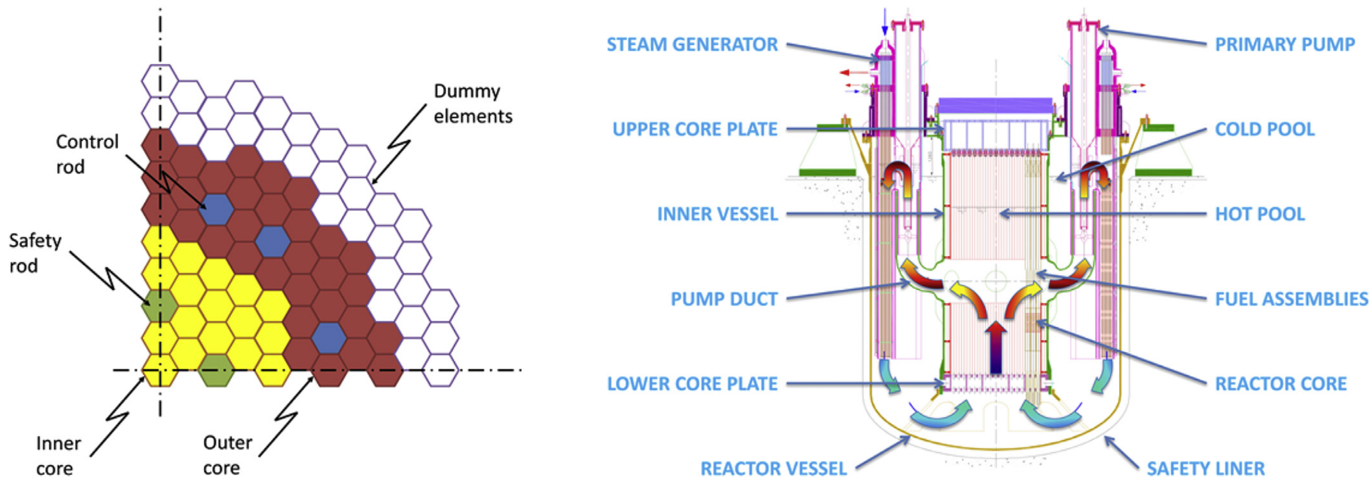

Fig. 1. ALFRED core (left) and primary system (right) layouts.

One of the most crucial objectives of ALFRED, as a demonstration reactor, is proving the commercial viability of the Lead-cooled Fast Reactor (LFR) technology being developed in Europe. This proof passes through the successful operation of ALFRED, demonstrating that the design assumptions, harmonized all together, provide not only the foreseen performances, but also the aimed reliability. Concerning the latter, a key objective is the verification that the assumed safety margins – included to safely account for the uncertainties affecting the design – cope with the abovementioned uncertainties to the aimed confidence, practically substantiating that they are well suited, then reducible in future designs. This verification, necessary before stepping-up the design beyond its conceptual phase and anticipating the certification of the design, was the aim of the task "ALFRED core safety parameters and influence of model uncertainties on transients" in the collaborative project "Preparing ESNII for Horizon 2020" (ESNII Plus), co-funded by the European Commission within the 7th EURATOM Framework Programme. The task focused notably on the transient behavior of the system in accident conditions ((Pasichnyk et al., 2013), (Jaeseok and Bae, 2017) and (Morris and Nutt, 2011)), pinpointed as the most challenging situation sought for the plant, so as to provide an extensively persuasive demonstration of the outstanding safety envisaged for ALFRED.

In particular, taking as reference unprotected scenarios, and aiming at a system allowing the safety authorities to consider the reduction (or even the elimination) of the emergency preparedness zones, ensuring extremely long grace times is a mandatory target. Long grace times are achieved via the respect of thermal limits associated with the integrity of the fuel (as the inventory of the radioactivity) and of all containment structures. Since temperatures (notably: those of the fuel, cladding and vessel) are to be checked, the related sources of uncertainties to be investigated are:

- uncertainties resulting in hot channels/hot spots;
- uncertainties affecting the thermal transient, hence the spontaneous equilibrium achieved by the system.

The formers are mainly due to elementary data (materials properties), system configuration (fabrication tolerances), operative conditions (components characteristics, monitoring and control systems sensibilities) and computational tools (models' approximations and numerical errors); all these sources of error must be evaluated and propagated to the observables of interest. The latter – once their effects are separated from those of the former – mainly come from nuclear data and computational tools. Among these, only the effect of uncertainties from nuclear data are considered, leveraging on a preliminary estimation of the errors due to computational models performed within the LEADER project (Petrovich et al., 2013).

The investigation of these uncertainties and the quantification of their effect on transients is therefore the main subject of the work, presented in three companion papers for conciseness reasons: Part I focuses on the effect of nuclear data uncertainties on reactivity coefficients, being the driving factors in establishing the dynamics of the transients; Part II concerns the translation of the various sources of uncertainties in hot channels/hot spots factors so to retrieve the uncertainties-perturbed temperature field; Part III puts the previous results together so to estimate safety margins for the clad and vessel and to estimate the number of pins expected to experience fuel melting.

Focusing on Part I, specific object of this paper, it mainly gravitates around a sensitivity/uncertainty (S/U) analysis on the reactivity effect of some elementary perturbations. This analysis allows to point out the most relevant cross-sections for every response function, and key regions where reactivity effects need to be evaluated, along with establishing the confidence interval for each reactivity effect. These effects are defined so that they can be directly combined into reactivity coefficients, as well as for the associated confidence intervals, so to provide a physically sound (i.e. where correlations among coefficients, stemming from dependencies on in common nuclear data, are taken into account), yet conservative, set of off-nominal (perturbed) values for the transients of interest.

The paper starts with the presentation of the used calculation methodology in Section 2 and its subsequent verification for the ALFRED case in Section 3; results of the evaluation of the confidence intervals for the key reactivity effects due to the uncertainties affecting nuclear data are presented in Section 4, which are, in Section 5, propagated to the integral reactivity coefficients driving the transients. The effect of the off-nominal reactivity coefficients on the ALFRED transient behavior is considered in Section 6; preliminary conclusions are finally drawn in Section 7.

2. SCALE6 system: calculation methodology

The SCALE6.1 code system (ORNL, 2011) was applied in this work for reactor physics calculations and sensitivity and uncertainty analysis (S/U) to nuclear data, using both SCALE6.1 and SCALE6.2 neutron covariance libraries. The applied methodology can be divided in three steps, as shown in Fig. 2.

2.1. Step 1: criticality analysis using the Monte Carlo KENO-VI code

Criticality calculations were performed using KENO-VI with a multi-group (MG) energy treatment, since the employed perturbation theory-based S/U approach requires forward and adjoint transport fluxes in