Feasibility Analysis of Representative Configurations of Core Degradation Situation During Severe Accidents in Nuclear Reactors

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Extended summary:

Climate change is probably the most challenging issue humanity is currently facing and will face in the coming years. To meet the global warming targets, set by the Paris agreement of 2015, nuclear energy must play a key role in the decarbonization of the energy sector. However, the currently operating nuclear power reactors fleet is aging and will eventually fail to meet future safety requirements. Therefore, new types of reactors are developed. These future reactor designs are known as Generation IV. The lead- and sodium-cooled fast reactors are among these future designs. These reactor designs are attracting a great deal of attention among the nuclear community, as not all challenges related to their safe operation during accidental conditions are resolved.

The thesis focuses on the development of a methodology for experiment design aimed to model severe core accident that may occur in power reactors, but in the safe environment of a zero-power reactor. The thesis revisits past experimental programs and introduces a full analysis of the SNEAK-12 program. However, the lessons learned from the revisited experimental programs reveal that most of them lacked representativity of an actual power system. Thus, the information gathered in these experiments, although of high value in understanding the physical phenomenon related to severe core accidents, was limited to the specific system examined.

Thus, the new methodology proposed in the thesis for the design of an innovative experimental program is based on the representativity method. This method is used to quantify the relationship between particular integral parameter (in the case of this thesis, the reactivity variation between different nominal and degraded configurations) of an experimental mock-up and the same response in an examined power system. The development of an innovative

methodology was divided into two parts, namely the assembly and the core levels. In the two parts, the main focus is on the representation of temperature effects, occurring at high temperatures (about 900°C at nominal state to about 3000°C during an accident) in a power reactor, in an environment of a zero-power reactor (at about 20°C). The latter is achieved through design optimization of the zero-power core loading in search of a maximal representativity factor.

The methodology developed in the thesis is based on sensitivity calculations performed using the Serpent 2 Monte Carlo code and on representativity analysis performed using Cadarache's covariance data (COMAC) and the publicly available covariance data for the ENDF evaluation. The methodology outline -

Step 1 – Calculation of a known Fast Reactor reference configuration and associated nuclear data based sensitivity coefficients at hot full power conditions (typical operation temperature of the fuel was set to 900°C).

Step 2 – First optimization step to determine the representative configuration of the Zero-Power Reactor, by modification of the core size and material loading (available from the stock-pile of the CEA).

Step 3 – Calculation of the known Fast Reactor degraded configuration and their associated nuclear data based sensitivity coefficients with the associated temperature of the accidental scenario (typically accidental temperature associated to fuel melt are above 2500°C).

Step 4 – Second optimization step to identify the representative second configuration of the Zero-Power Reactor, in such way that the representativity factor associated to the reactivity variation between the two configurations would be as close as possible to unity. This is achieved through the modification of the geometry and/or material balance in the test zone of the Zero-Power Reactor.

Step 5 – This stage could be split into two; the first possible result of Step 4 could be manufacturing of a dedicated fuel for the experimental program. However, this would require additional fund and time for fabrication of the fuel. The second possibility could be the utilization of a vast fuel stockpile available at CEA Cadarache to simulate the fuel characteristics identified in Step 4.

Step 6 – In case the second possibility is preferred, a re-computation of the second configuration of the Zero-Power Reactor is required to ensure that the representativity factor does not drop by much, so the experimental measurement would continue to stay valid.

The first stage of the research was to identify the relationship between temperature and density on a single fuel assembly basis. The two fast reference systems for this stage and the entire research are the ASTRID sodium-cooled fast reactor and the ELSY lead-cooled fast reactor. According to the presented about methodology, the two types of fuel assemblies of the power reactors are degraded, and the sensitivity vectors are obtained. Second, the reference fuel assemblies of the ZEPHYR reactor are constructed to ensure representativity of the reference assemblies in the power system. The next step is to identify the second degraded fuel assembly that ensures high representativity of the reactivity variation between the two reference fuel assemblies. In order to achieve this goal, the representativity factor is treated as target function for an optimization process based on the particle swarm optimization (PSO), where the optimization factor is the PuO_2 content inside the MOX fuel in the degraded zone.

The results obtained from the single fuel assembly optimization reached the minimal representativity value that was set as a goal (0.85) in the two reference systems. The assembly studies examined the impact of the temperature effects on the zero-power system, and the results clearly show that temperature effects can be considered in the experiment design by adapting the amount of the fissile material in the degraded zones. Furthermore, the utilization of PSO as the optimization process enabled a detailed examination of the search space. In all cases, the structure of the search space contained three zones, i.e., low representativity values, a zone where the results are not feasible due to the structure of the sensitivity vectors, and a region (or a point) where the results reach the target value. Finally, a sensitivity analysis of the representativity method is performed with respect to the choice of covariance data. The results show that although the method is sensitive to the data, it is still possible to reach a minimal level of representativity. It should be noted, that the proposed process in this thesis should be treated with care, i.e., nuclear data should be consistently treated (no mixing of cross-section and covariance matrix should be made). With respect to the latter, more research related to nuclear data is needed to ensure more coherent and accurate cross-sections and covariances.

Considering the positive results on the assembly calculations, the final stage of the feasibility studies was to extend the methodology to a full core level and to introduce the actual MASURCA fuel into the simulations. However, a change in the optimization method is required, as PSO is a population-based optimization, which requires a large population to converge and is unrealistic for a full core level. Therefore, the assembly optimization scheme is modified by replacing the PSO with the Nelder-Mead (NM) Simplex method. This optimization method requires a much smaller number of Serpent runs to achieve convergence to the good representativity value. As demonstrated in the assembly level optimization, it is possible to obtain a region of feasible solutions in the search space. However, one result is sufficient to ensure feasibility, as is the case in full core optimization.

The thesis results showed that utilizing the proposed optimization methodology enables to reach high representativity values with respect to the given degraded configuration and the reference system. However, construction of such an experimental program will require modifications to the ZEPHYR core, going from the preliminary coupled fast/thermal core concept to a full fast core. The advantage of this modification is twofold; constructing an experimental mock-up for a fast reactor (sodium or lead) and facilitating the future experimental program design.

The implementation of experimental programs in the ZEPHYR facility related to severe accident studies, as described in this thesis, is of high value for the design of supplementary experimental programs. Such programs can be related to, e.g., development of instrumentation (in-core and ex-core) for severe accident progression monitoring and experiments dedicated to nuclear data, which will target the needs for severe core accident modeling. Finally, these programs can provide experimental information for verification of codes utilized in severe core accident modeling, can provide experimental feedback for new reactor concept designs.