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Development of Steady-State Thermal-Hydraulic Analysis and Bowing Reactivity Evaluation Methods Based on Neutron and Gamma Transport Calculations

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ABSTRACT:

Most of the sodium-cooled fast reactor (SFR) design and analysis codes were initially developed based on the broad-group diffusion theory approximation. Since then, considerable advances in methods and codes have been made along with the advancement of computer technology. Now the VARIANT transport code is routinely used for whole core neutronics analyses based on homogenized assembly models and for fuel cycle analyses with the REBUS code. Significant improvements have also been made in the area of nuclear data and multi-group cross section generation, resulting in the state-of-the-art code MC^2-3. For reactivity and sensitivity coefficient calculations, the generalized perturbation theory code VARI3D based on the diffusion theory code has been extended to the PERSENT code based on the transport theory. In addition, as part of the DOE-NE's NEAMS program, a discrete ordinates unstructured finite element transport code PROTEUS-SN has been developed, which allows the explicit representation of detailed geometry for reactor simulations. However, the SE2-ANL code for steady state thermal-hydraulics analysis and the NUBOW-3D code to evaluate the reactivity feedback due to assembly bowing still use the neutron flux, gamma flux and/or adjoint neutron flux from diffusion calculations.

The objectives of the proposed work are to develop accurate methods for heat distribution calculation and bowing reactivity evaluation based on the core solutions of VARIANT and the assembly solutions of PROTEUS-SN and to implement the methods into the SE2-ANL and NUBOW-3D codes. Specifically, an innovative method to determine the heat generation rates in fuel pins and duct walls will be developed by combining the neutron and gamma fluxes obtained from the VARIANT transport calculations for homogenized-assembly core models and the PROTEUS solutions for heterogeneous single-assembly models. A perturbation theory formulation will be developed to calculate the reactivity worths for the planar displacements of every axial segment of all the moveable assemblies in the core using the reconstructed flux distributions. For the coupled neutron and gamma heating calculation, the gamma production and reaction cross section libraries and heating factor libraries will be generated using the latest ENDF/B-VII.1 data.

By providing enhanced computational capabilities and improved prediction accuracies for the steady-state temperature distributions and the reactivity feedback due to assembly bowing, the developed computer codes will be beneficial to the on-going and future advanced reactor concept projects of the DOE-NE's ART campaign. The enhanced computational capabilities will allow to explore a broader range of design space and to incorporate innovative design features, and the improved prediction accuracies will contribute to reducing the economic penalties due to the conservative design margins to accommodate the prediction uncertainties. At the same time, the required method and code development activities will be helpful in developing expertise in the fast reactor design and analysis at a university and training the next generation of fast reactor designers and analysts.