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Coupled calculations for the fast reactors safety justification with the EUCLID/V1 integrated computer code

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The EUCLID/V1 integrated computer code is designed for the safety analysis and justification of the new generation NPPs with liquid metal cooled fast reactors under normal operating conditions, design basis accidents and beyond design basis accidents. The EUCLID/V1 code includes the system thermohydraulics module, spatial time-dependent neutronics module, quasi two-dimensional fuel rod module and the module of burnup and decay heat calculations. In the neutronics module the improved quasistatic method is employed to solve the transport equation in the multigroup diffusion or discrete ordinates (Sn) forms.
The extensive V&V of the single modules have been carried out on analytical and numerical tests, experimental results and benchmarks. To validate a coupled modeling of physical processes in a reactor core and its loops, the experimental data on the BN-600 and BOR-60 transient regimes have been used.
Some of the BN-1200 and BREST-OD-300 reactor facilities design basis accidents and beyond design basis accidents have been simulated by means of the EUCLID/V1 code. Particularly, the loss of offsite power accident in the BN-1200 reactor has been modeled. In this design basis accident the reactor shutdown cooling with the emergency heat removal system is considered. It has been shown that after the control rods drop and reactor pumps shutdown the efficiency of two of four channels of the emergency heat removal system is sufficient to prevent the maximum fuel, cladding and sodium temperatures from exceeding the design limits.
In frame of the test calculations the accident caused by the introduction of the total positive reactivity margin via withdrawal of all control rods from the reactor core at the maximum design speed during full power operation without scram operation (UTOP+ULOF) has been simulated for the BREST-OD-300 reactor facility. The obtained results indicate that after the reactor coolant pumps shutdown the total power decreases due to the thermohydraulic reactivity feedbacks and operation of the passive feedback system. By 100 s of the scenario natural circulation of lead progresses in the primary loop, it is equal to 9.6% of the nominal total flow. The core disruption does not happen.

Country/Int. Organization

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