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Modelling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Primary System/Containment System Interface Source Term Estimation

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Three Work Packages (WPs) were defined in this Coordinated Research Project (CRP) whose objective was to estimate fission-product-transportation behaviour inside the reference pool-type sodium-cooled fast reactor (SFR) volumes (i.e., in-primary vessel, cover gas system and in-containment building) at different time scales under severe accident conditions. This WP, WP-2, is defined to estimate the primary system/containment system interface source term using improved models and tools for the cover gas, sodium ejection, radionuclide chemical composition and distribution in the containment. After the discussion between the participants of this WP, it was decided to evaluate mass of primary sodium instantaneously ejected into the Reactor Containment Building (RCB) as a common benchmark problem.

The exercises were carried out for a reference pool type SFR of 1250 MWth capacity fuelled with mixed oxide fuel. The accident sequence to be considered is Unprotected Loss of Flow Accident (ULOFA) which is assumed to result in a core damage event with release of radionuclides into the primary coolant and cover gas.

Four organizations, NCEPU (China), IBRAE/RAN (Russia), IGCAR (India) and JAEA (Japan) were finally participated in this WP. NCEPU calculate the leakage of liquid sodium using CFD code, FLUENT. IBRAE/RAN did the simulations with their code EUCLID/V2. IGCAR used FUSTIN/BLVDYN and NETFLOW codes for simulations. JAEA used PLUG code for the calculations of sodium ejection.

Basic and parametric case of the calculation were carried out. The total amount of the ejected sodium onto the roof slab for basic case was in a good agreement between the participants. The results of the parametric analysis revealed that the pressure difference above and below the vessel head, which is the driving force of the sodium ejection, and the resistance coefficient of the bent portion of the plug gap make a relatively large contribution to the amount of ejected sodium.

In WP-2, the amount of sodium ejected onto the roof slab was evaluated as a common benchmark problem. The long-term release of radio nuclide would be due to the release of suspended/dissolved activity from sodium through the leak paths made available in the top shield. Quantitative evaluation of such long-term release is addressed as the future work.

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