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Benchmark analyses for EBR-II shutdown heat removal tests SHRT-17 and SHRT-45R – (2) subchannel analysis of instrumented fuel subassembly

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ABSTRACT

Two kinds of Loss of Flow experiments conducted in EBR-II are the subject of an IAEA benchmark exercise. The present work discusses the subchannel analysis in an instrumented subassembly, analyzed with the COBRA-IV-I code. Boundary conditions are provided by a 1D thermal–hydraulic system code (NETFLOW+ +) for a protected loss-of-flow and an unprotected-loss-of-flow tests conducted at the EBR-II reactor. The instrumented subassembly with 61 pins is installed in the 5th row of the reactor core in order to measure the temperature profile across the mid-plane and the top region of the fuel subassembly. The pin lattice in the subassembly consists of 126 subchannels. Calculated temperature profiles at three levels are compared with the measured results. Biased temperature profiles with respect to the center line of the subassembly measured in the experiments are appropriately simulated by the code. The presence of the spacer wire by itself introduces a biased temperature profile. The biased temperature profile is also shown by a CFD calculation using a 7-pin partial model of the instrumented subassembly. In order to obtain close agreement with measured temperatures, the power profile within the subassembly must be taken into account. Temperature evolutions during the transients are simulated by the subchannel code.

1. Introduction

Benchmark problems concerning experiments in the Experimental Breeder Reactor II (EBR-II), proposed by Argonne National Laboratory (ANL) and coordinated by the IAEA, were analyzed by the one-dimensional thermal-hydraulic system code NETFLOW+ + and the neutronics analysis code ERANOS (Mochizuki et al., 2014). The same problems were calculated by Yue et al. (2015) in the scope of the same IAEA Coordinated Research Project (CRP), and the calculated results up to now were summarized by Briggs et al. (2015), Bates et al. (2017) and Partha Sarathy et al. (2017). The problems are the shutdown heat removal tests SHRT-17 and SHRT-45R conducted at the experimental fast breeder reactor EBR-II. These sort of tests were conducted on the basis of the rigorous evaluation on the fuel intactness during the transients (Lahm et al., 1987). The test SHRT-17 is a natural circulation test after a loss-of-flow (LOF) event by tripping two main pumps and scramming the reactor, and the test SHRT-45R is a natural circulation test after a supposed unprotected-loss-of-flow (ULOF) event. The temperature distribution in the subchannels was measured in two subassemblies by replacing the stainless wire spacers with thermocouples. The measured data show the local temperature distribution in the instrumented subassemblies. The prediction of temperatures in various subchannels is one of the benchmark tasks. The COBRA-IV-I code (Wheeler et al., 1976) is used to investigate the axial and radial temperature distributions caused by the wire-wrapped spacer in the instrumented subassembly. Especially, since asymmetric radial temperature distribution was observed in a specially instrumented subassembly of the EBR-II, the cause should be clarified. Furthermore, a small-scale pin geometry is calculated by the CFD code to check whether or not the calculated result by COBRA-IV code is reasonable one. Channel inlet flow rate, channel inlet temperature and heat flux from the cladding calculated by the NETFLOW+ + code are imposed on the COBRA-IV-I code as boundary conditions. Results of an analysis of the two tests with the SASSYS-1 code were presented earlier by Dunn (1996) and Dunn et al.

Abbreviations: ANL, Argonne National Laboratory; CFD, Computational Fluid Dynamics; EBR-II, Experimental Breeder Reactor II; EMP, electromagnetic pump; IHX, intermediate heat exchanger; LMFBR, liquid metal cooled fast breeder reactor; LOF, loss-of-flow; MTC, mid-plane thermocouple; SHRT, shutdown heat removal test; TC, thermocouple; TTC, top-of-core thermocouple; ULOF, unprotected loss-of-flow

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Nomenclature

A	flow area (m^2)
D_h	hydraulic equivalent diameter (m)
G	axial mass velocity (kg/m^2s)
$\overline{G}$	average axial mass velocity (kg/m^2s)
Re	Reynolds number
S	gap width (m)
w'	fluctuating cross-flow per unit length (kg/m s)
β	mixing factor
μ	dynamic viscosity (Pa·s)

(2006).

The NETFLOW++ code has been developed by Mochizuki (2010) through verification using various data measured at mockups with water or sodium coolant, and validation using test data measured at liquid-metal-cooled fast reactors "Monju" and "Joyo" (Mochizuki, 2007a). Since these reactors are so called loop-type reactors, every component needed for the analysis is already implemented in the code. An inter-subassembly heat transfer model has also been implemented (Mochizuki, 2007b). As for a pool type FBR, the code was used for the IAEA natural convection benchmark analysis of the PHENIX reactor (Mochizuki et al., 2013).

The COBRA-IV-I code was developed by Pacific Northwest National Laboratory (Wheeler et al., 1976) and modified in order to calculate the wire wrapped fuel assemblies (Donovan et al., 1979). We used the version prepared by the Institute Jozef Stefan (IJS) available from the NEA Data Bank. COBRA-IV-I is a subchannel analysis code which calculates the flow and temperature distributions in fuel bundles and reactor cores for steady state or transient conditions. The COBRA-IV-I code can be used for the analyses of water cooled reactors, liquid metal cooled reactors and gas cooled reactors. For liquid metal cooled fast breeder reactors, the COBRA-IV-I code has been validated for the core and heat exchangers (IAEA, 1999).

Temperature distributions for the above mentioned tests using various codes are introduced by Partha Sarathy et al. (2017). Results using CFD codes and subchannel codes are also introduced in this paper.

2. Benchmark problems

2.1. EBR-II plant overview

The EBR-II reactor is a very unique liquid metal cooled fast breeder reactor (LMFBR) which is classified as a hybrid-type reactor because of its heat transport system. The flow path from a main pump to an intermediate heat exchanger (IHX) through the reactor core is connected by piping, which is a similar configuration as in a conventional loop type reactor. However, the outlet of the IHX is not connected directly to the main pumps but instead discharges into a large cold pool as shown in Fig. 1, and this arrangement is similar to the situation of pool type reactors. All components, two main pumps, an electromagnetic pump (EMP), the reactor core, piping system of the primary heat transport system, and the IHX are submerged in the sodium pool. There are two kinds of lower plena, i.e., a high-pressure plenum and a low-pressure plenum. Piping from the main pump is connected to the high-pressure plenum and a branch pipe of small diameter is connected to the low-pressure plenum. Fuel assemblies are connected to the high-pressure plenum, while reflector and blanket assemblies are connected to the low-pressure plenum. All subassemblies are connected to the upper plenum, and the plenum is connected to the IHX via a reactor outlet piping called Z-pipe. The EMP is provided at the inlet of the IHX. The secondary system of the IHX is connected to steam generators. A fuel element consists of a cladding tube with a single, sodium-bonded metal fuel slug. The fuel slug contacts on the cladding surface during the reactor operation. Therefore, gap conductance or contact conductance is quite large compared to that of oxide fuel and fuel expansion should be taken into account during the transients. A large negative reactivity is applied to the rector by the radial and axial fuel expansions which are intrinsic mechanisms of the reactor. Control rod relative positions to the

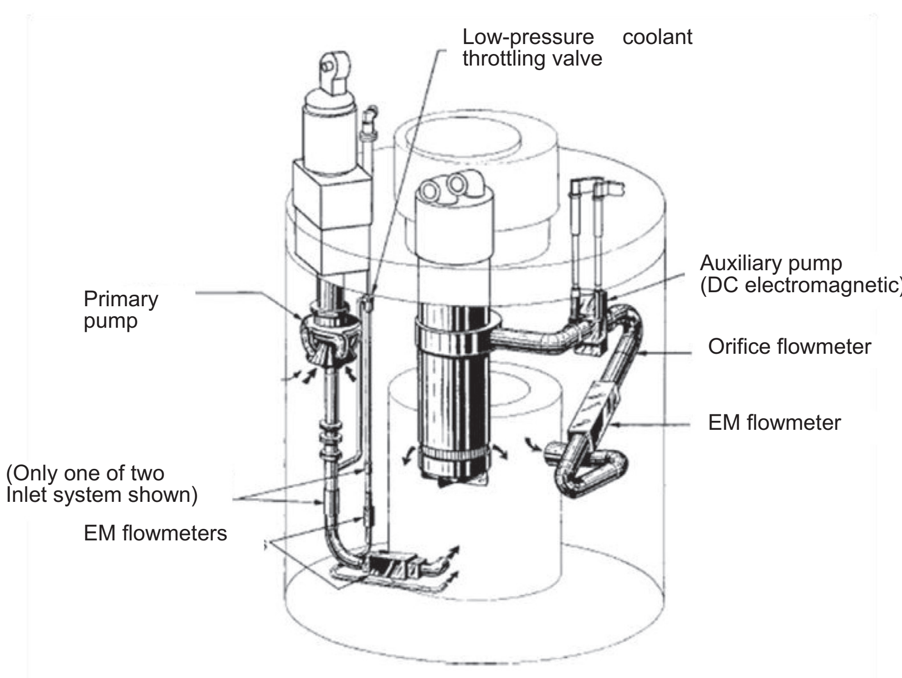

Fig. 1. Schematic of the EBR-II reactor.