Final Results and Lessons Learned from EBR-II SHRT-17 Benchmark Simulations

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Abstract. In 2012 the IAEA initiated a 4-year coordinated research project (CRP) “Benchmark Analyses of EBR-II Shutdown Heat Removal Tests”, with Argonne National Laboratory serving as the lead technical institution. Nineteen participants from eleven countries were involved in the project. The overall purpose of the

CRP was to improve validation of state-of-the-art sodium-cooled fast reactor (SFR) computer codes through comparisons of the analytical predictions against whole-plant recorded test data. A secondary purpose was training of the next generation of SFR analysts and designers through participation in international benchmark exercises. Numerical simulations were performed for the two most severe experiments conducted in the 1980s during Argonne's EBR-II Shutdown Heat Removal Tests program. The first test was SHRT-17, where a PLOF (Protected Loss Of Flow) accident scenario was performed, and the second – SHRT-45R, where a ULOF (Unprotected Loss Of Flow) scenario was performed. This paper describes the results (blind and final) of the SHRT-17 experiment simulation, findings of the CRP benchmark exercise associated with the EBR-II SHRT-17 test, improvements proposed by the participants, and the lessons learned within the project.

Key Words: Benchmark, Validation, Sodium thermal-hydraulic, Fast Reactor.

1. Introduction

One of the CRP aim was an improving in the understanding of processes, which occur in a sodium cooled fast reactors under transient conditions and providing an experimental data for system codes validation. The most valuable data in the frame of the validation process is the data obtained during the integral facility operation. With these facilities, we can investigate all thermal-hydraulic and neutron physics processes, which are important from the safety point of view, in a self-consistent approach. One of such process, in terms of sodium cooled fast reactor, is the possibility of the natural circulation using to remove the core decay heat [1]. In the CRP participants were proposed to provide a simulation of the SHRT-17 test, where such phenomena during the PLOF transient was investigated on June 20, 1984. Total number of 18 participants were involved in calculation process of SHRT-17 test with their codes, models and approaches. Another test was carried out under the ULOF conditions. More details about this test CRP results can be found in [2].

2. EBR-II and SHRT-17 test description

Experimental Breeder Reactor-II (EBR-II) was designed, built and operated by ANL in Idaho until the 1964. Nominal thermal power of the reactor was 62.5 MW with an electric output of approximately 20 MW. Some other EBR-II nominal parameters are presented in Table I. EBR-II is a fast sodium cooled reactor with integral layout, meaning that all the main equipment of the primary side (core, IHX, pump) is located in a vessel filled with sodium.

On June 20, 1984 a full power loss-of-flow test in the SHRT series demonstrated the effectiveness of natural circulation in the EBR-II reactor. During this test the plant protection system was used to simultaneously scram the reactor. To initiate the SHRT-17 test, both primary coolant pumps and the intermediate-loop pump were tripped to simulate a protected loss-of-flow accident beginning from full power and flow conditions. In addition, the primary system auxiliary coolant pump that normally had an emergency battery power supply was also turned off. The reduction in coolant flow rate caused reactor temperatures to rise temporarily to high, but acceptable levels as the reactor safely cooled itself down by natural circulation.

As the SHRT-17 test continued, the reactor decay power decreased due to fission product decay. After the start of the test, no automatic or operator action took place until the test had concluded. Table I summarizes the initial conditions and transient initiators for the SHRT-17 test.

TABLE I: EBR-II parameters prior to the SHRT-17 test.

Initial Power	57.3 MW
Initial Primary Coolant Flow Through Core	8500 gpm at 800^{o}F
Initial Intermediate Coolant Flow	5615 gpm at 582^{o}F
Initial Core Inlet Temperature	665^{o}F
Primary and Intermediate Pump Coastdown Conditions	Power to motor-generator sets removed
Control Rods	Full insertion at test initiation
Auxiliary EM Pump Conditions	Power to Auxiliary EM Pump removed

3. The SHRT-17 test benchmark

To provide a calculation of the SHRT-17 test each participant has to prepare a primary heat transport model. In general, all of the participants use the same approach, which consist in a representation of the primary and secondary sides of EBR-II with 1D/0D hydraulic elements and 1D/2D heat structures. Hence, some participants attempted to represent EBR-II elements with 2D or 3D approach. For example, SIMMER code is a two-dimensional, so the primary side was represented in the R-Z geometry and the secondary side by boundary conditions.

Because of the fact that primary side loops are located in a sodium tank asymmetrically, participants from ENEA, with RELAP-3D code, for primary sodium tank use a 3D model, consisting of 19 axial nodes, 2 radial rings, and 8 azimuthal sectors. The number of azimuthal meshes was chosen on the basis of the geometrical position of the pumps, reactor inlet and outlet pipes, and the IHX. This attempt was made to investigate the behaviour of sodium at low velocities in a sodium tank.

During the SHRT-17 test a lot of data, devoted to the core parameters, was recorded. It was the average core outlet temperature, outlet sodium temperature for some SAs, mass flows through the experimental SAs. For this reason, significant part of the participants' time was focused on core models developing. Some differences were observed in a core modelling approach. According to the Table II there are two main options in a core representation. One option is to divide a core channels in accordance to the power to flow ratio (P/F ratio). Another option is to divide the channels in accordance to the subassembly type. Some models assumed to simulate a core in accordance to the core rows.

It would be also noted that participant from POLITO simulated only the core in their analysis with 1D approach and they done it in a 1:1 approach.

TABLE II: Participants' approaches for a EBR-II model.

Participant	Code used for SHRT-17 simulation	Heat transport model approach for SHRT-17 test	Channels dividing for core model
CIAE	SAS4a/SASSYS-1	0D and 1D	P/F ratio, SA type
NCEPU	SAC-CFR	0D and 1D	SA type
XJTU	THACS	0D and 1D	SA type
IRSN	CATHARE	0D and 1D	SA type
KIT/Kyushu	SIMMER-III	2D (R-Z)	P/F ratio, SA type
IGCAR	EBRDYN	0D and 1D	P/F ratio, SA type

ENEA	RELAP5-3D	0D, 1D, sodium tank modelled in a 3D approach	Core 1:1 Blanket 12 azimuthal subdivisions with SA type
NINE	RELAP5-3D	0D, 1D	Rows
POLITO	FRENETIC	Core model in 1D approach	1:1
JAEA	Super-COPD	0D and 1D	1:1
U. Fukui	RELAP5-3D, NETFLOW++	0D and 1D	SA type
KAERI	MARS-LMR	0D and 1D	P/F ratio, SA type
KINS	TRACE	0D and 1D	Flow type
NRG	SPECTRA	0D and 1D	SA type
IBRAE	SOCRAT-BN	0D and 1D	SA type
PSI	TRACE	0D and 1D	P/F ratio, SA type
ANL	SAS4A/SASSYS-1	0D and 1D	P/F ratio, SA type

As a boundary conditions for SHRT-17 test the following parameters were used for a primary side [3]:

- Primary pumps speed as a function of time;
- Total core power (fission and decay) as a function of time.

The boundary conditions for the intermediate sodium loop for the SHRT-17 benchmark were the sodium flow rate and sodium temperature at the inlet to the IHX.

Many measurements were recorded during the SHRT tests. The measurements best suited for comparison with the benchmark calculations are listed below. Several other calculated values are included that were not measured during the SHRT tests but are ideal for direct code-to-code comparisons among benchmark participants.

Those values that benchmark participants calculated during the transients are:

- High-pressure and low-pressure inlet plena temperatures,
- Z-Pipe inlet temperature,
- IHX primary side inlet temperature,
- Sodium mass flow rate at the primary sodium pumps,
- IHX intermediate side outlet temperature,
- XX09 and XX10 temperatures at the thermocouples locations
- XX09 and XX10 sodium mass flow rate,
- Peak cladding and fuel temperatures,
- Peak in-core coolant temperature,
- Minimum margin to coolant boiling.

4. Improvements, lessons learned and final results from EBR-II SHRT-17 test benchmark

The first phase of the CRP was blind calculations of tests without access to the recorded data. After completion of the first phase participants received experimental data for both transients, so the second phase of the CRP begin. In Table III the overall improvements made by participants between phases are summarized.

TABLE III: SUMMARIZE OF THE IMPROVEMENTS MADE BY CRP PARTICIPANTS.

Participant	Improvements
CIAE	Improvements in the core model discretization, which allow to achieve a better prediction of a natural circulation in a transient
XJTU	Implementation of a three-layer model of the sodium pool. Heat transfer between the Z-Pipe and the cold pool, and the inter-wrapper flow model. With such improvements the prediction of a natural circulation flow become closer to experimental data
IRSN	Individual modeling of primary pumps and inlet piping in order to reproduce the dissymetrical behavior and flow inversions in the inlet piping during the natural convection transient. Increase of pressure drop coefficients at low Reynolds number for better mass flow rate prediction.
KIT	Two main modifications were introduced: • The input pump head was modified early in the transient to take into account the slightly different behaviour of the two primary pumps. • IHX definition/location in SIMMER was slightly modified • Radial heat transfer made available for experimental subassembly XX10
IGCAR	Regrouping of the SAs. Correction for steady-state mass flow rates. Taking into account of heat losses from the Z-pipe to the cold pool
ENEA	Implementation of the dependence of energy loss coefficients on the Reynolds number. Imposing of "locked rotor" conditions in the final calculation in a pump model. Modification of the core orientation
NINE	Changing the type of the core channels discretization. Updating in initial boundary conditions. Modification of the sodium pool model. Modification for the leakage flow paths
POLITO	Implementation of the radial thermal conduction model of the pins, substituting for the previous axial model. Implementation of the coolant flow in the thimble of the so-called "box-in-the-box" hexagonal subassemblies. Improvement of the model for the subassembly inlet orifices
JAEA	Modification of the upper plenum model from the single perfect mixing volume used in the blind simulation to two mixing volumes. The two mixing volume result seems to be slightly better than the single mixing volume result

FUKUI	Adding of a discharge line from the reactor tank to the cold pool after the blind calculation in order to check the effect of the leakage on the plant behaviour. It was clarified that the effect was negligible. Adjusting of local loss coefficients in order to reproduce the measured data. Adjusting of heat transfer coefficients from the inlet piping of the lower plena to the cold pool based on the CFD result
KAERI	Improvement of wall-friction model. Addition of sodium gap between SAs. ANS94 decay heat model
KINS	Improvements in core model and the heat structures for the upper plenum, Z-Pipe and IHX shell
NRG	Loss factor for the locked pumps. Decay heat based on SHRT-17 data. Refinement of the upper/lower plenums nodalization. Change the calculation cell where IHX inlet sodium temperature was measured. Introducing an axial power profile. Heat loss through the floor was added
IBRAE RAN	IBRAE did not participate in phase 1. But during the CRP some modifications were made: For better prediction of an average core outlet and Z-pipe inlet sodium temperature the upper plenum is separating into several nodes. Implementation of a natural convection heat transfer model for estimation of heat losses from Z-pipe, IHX and head pipes to the cold pool. Making an assumption for a locked rotor pump.
ANL	Improvements in a locked pump model

Below will be presented and discussed only the results of the final participants calculations.

4.1.Primary side mass flow

During the SHRT-17 test all four flowmeters in the piping following pump 1 had failed, but the flowmeters measuring the flows in a pump 2 were in operate. In the FIG.1 the mass flow rates through the pump 2 obtained by participants are depicted in comparison with experimental data.

All the simulations model the rapid loss of flow following the loss of power to the pumps at time t=0 s. and predict the qualitative behaviour of the flow as it stabilizes upon establishment of natural circulation. The region where the simulations differ from one another the most is from about 75 to 300 seconds, which is the transition to natural circulation. The differences in nodalization, pump models, locked rotor coefficient values, etc. among the models all impact the predicted values in this region.

4.2.Primary side sodium temperatures

FIG.2 illustrates the comparison of the participants sodium high-pressure inlet temperatures with comparison to experimental data (red line). It is obvious that the core inlet temperature is determined by a sodium, which comes via pumps from the sodium tank. This is determined by the fact that thermal inertia is relatively high (total sodium volumetric inventory in primary system is about 340 m^3), and sodium velocity in a tank is relatively low after pumps coast down. Thus, there are no significant sodium temperature changes in the experimental data during the transient. All of the participants predict this quite accurately.

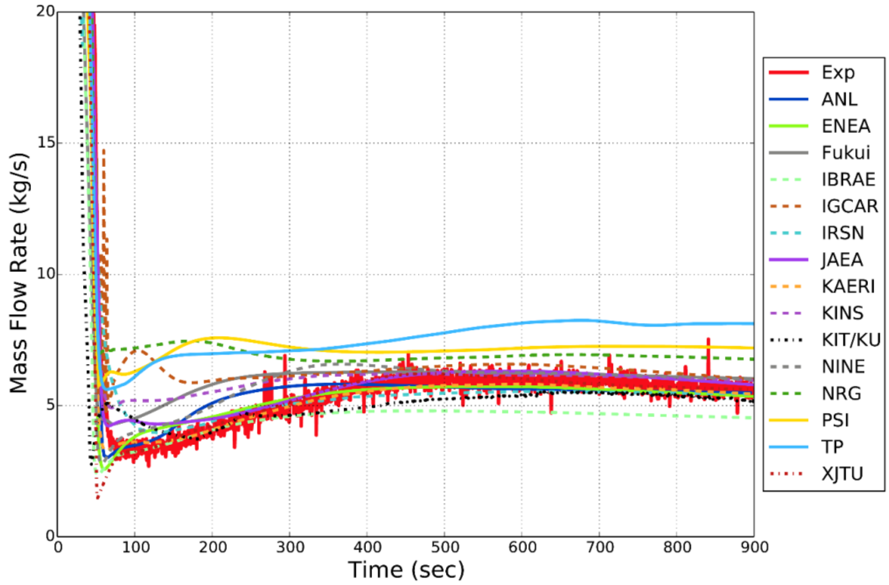

FIG. 1. Time dependencies for the pump 2 mass flow rates.

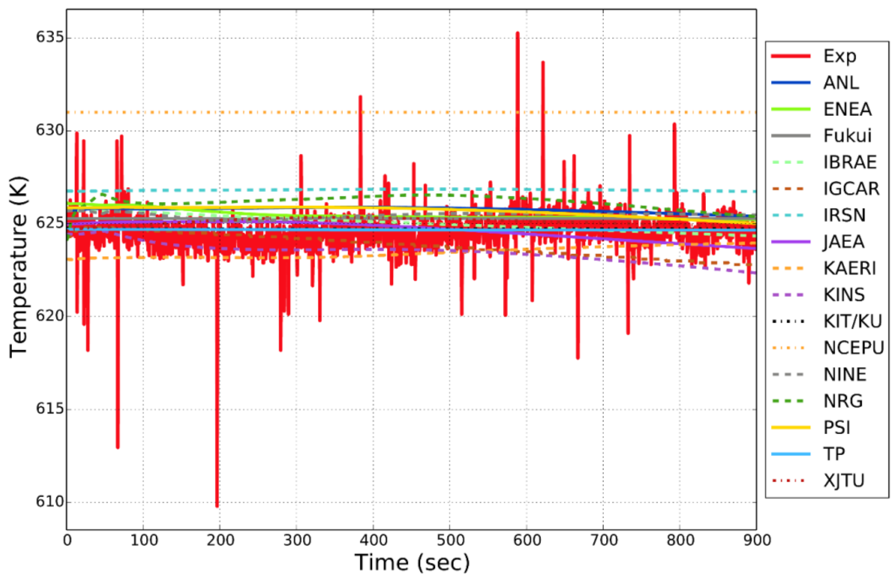

FIG. 2. Time dependencies for the core inlet sodium temperature.

One of the most complicated experimental data gained in the SHRT-17 test was the primary side IHX inlet temperature. The primary-side IHX inlet temperature was measured by a thermocouple installed behind an impact baffle plate in the diffuser region at the top of the IHX. The primary-side outlet temperature is the average of four temperatures measured just outside the IHX outlet window, so we should treat it as a sodium temperature in a sodium tank or the core inlet temperature. As we can see in a FIG.3, during the transient the sodium temperature at the IHX inlet becomes lower than a temperature at the IHX outlet (core inlet). Obviously, the sodium couldn’t cool down to such temperatures in the Z-pipe. One of the explanation may be that the thermocouple measured the IHX inlet temperature was located below the Z-pipe mid line inside the IHX tubes bundle. As a fact, such low temperatures in a thermocouple reading are caused by the cooling down from the secondary side sodium, which is relatively cold. FIG.4 illustrates the participants' calculation results of the primary IHX temperature with comparison to the experimental data. Most of the participants provide a calculation of the temperature exactly at the Z-pipe outlet mid-plane, but two participants make a measurement in neighbour computational cells, what gives them relatively good agreement with experimental data.

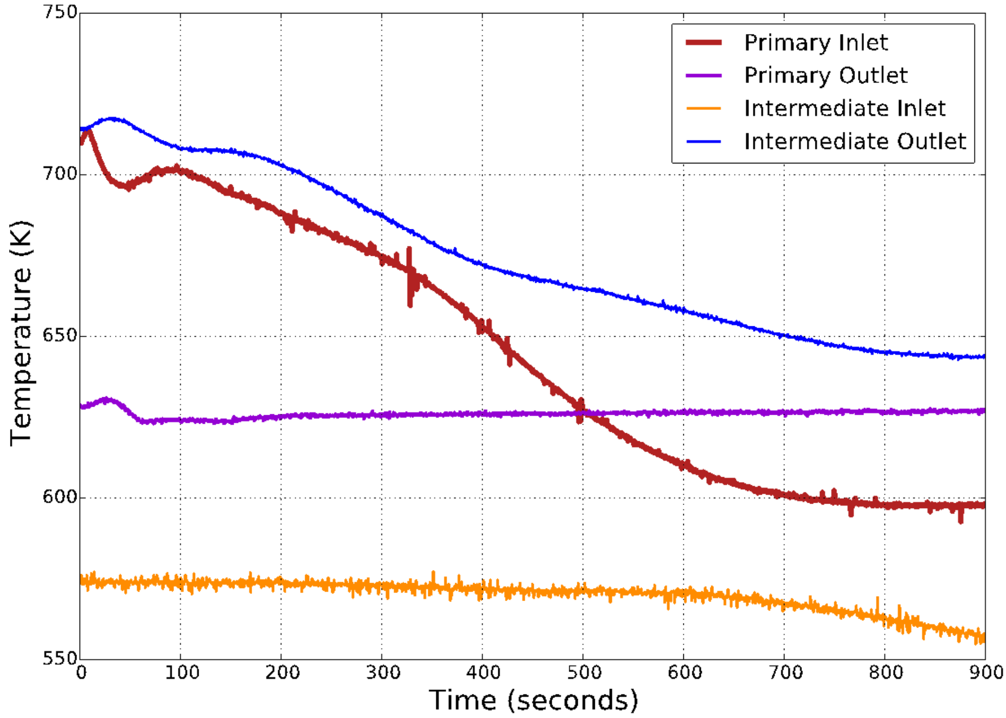

FIG. 3. Experimental time dependencies for the primary sodium temperatures.

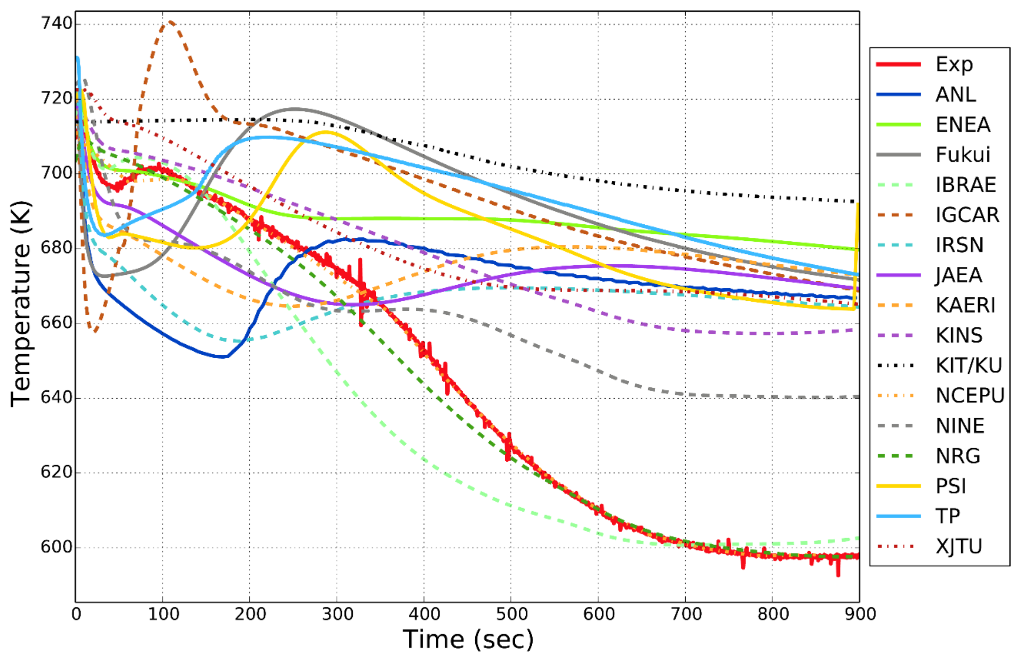

FIG. 4. Time dependencies for the primary side IHX inlet sodium temperature.

As a code-to-code comparison a peak in-core cladding temperature was chosen. This temperature is determined by a power in a corresponded SA and mass flow. As mentioned above, the power was not calculated in this benchmark, and the mass flow rates, gained by participants are quite close to each other. So the same picture we can see for a peak in-core cladding temperature. Deviation between maximum and minimum participants' results depicted in a FIG. 5 is not more than a 125K.

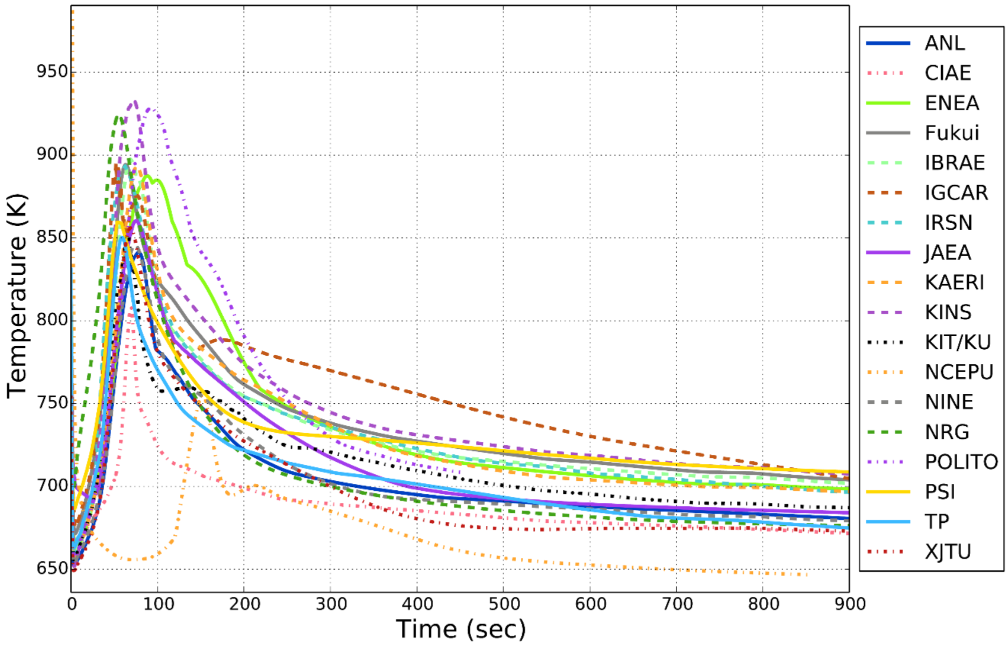

FIG. 5. Time dependencies for the peak in-core cladding temperature.

5. Conclusions

Participation in the IAEA CRP, "Benchmark Analyses of an EBR-II Shutdown Heat Removal Test" allows to participants improve their understanding in the frame of the system codes developing in a natural circulation core cooling phenomena for sodium cooled fast reactors. Many improvements in participants' models were made during the 4 years project. Regarding SHRT-17 test it's possible to get insight into the heat transfer processes both in core and in a Z-pipe. Much attempts were made to predict more accurately the mass flow during the natural circulation, namely, the pump behavior and friction loss factors. A lot of limitations were challenged for the participants. Lack of experimental data for some recorded parameters, uncertainties in a measurement device locations, absence of some EBR-II design specification all this did not prevent the CRP participants collectively to show significant modelling progress over the course of the CRP.

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