Conclusions of a Benchmark Study on the EBR-II SHRT-45R Experiment

E.Bates[1], D. Zhang[2]., B. Truong[1], D. Sui[3], W. Hu[4], G. H. Su[2], T.Sumner[5] L. Maas[6], B. Vezzoni[7], M. Marchetti[7], R. Zanino[8], D. Caron[8], W.F. Van Rooijen[9], H. Mochizuki[9], K. Morita[10], C. Choi[11], M. Stempniewicz[12], N. Rtischev[13], Y. Zhang[14], K. Mikityuk[14]

[1] TerraPower LLC, Bellevue, WA, USA

[2] Xi'an Jiaotong University, Xi'an, Shaanxi, China

[3] North China Electric Power University, Beijing, China

[4] China Institute of Atomic Energy, China National Nuclear Corp., Beijing, China

[5] Argonne National Laboratory, Argonne, Illinois, USA

[6] Institute for Radiological Protection and Nuclear Safety, Fontenay-aux-Roses, France

[7] Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany

[8] Department of Energy, Politecnico di Torino, Turin, Italy.

[9] Research Institute of Nuclear Engineering, University of Fukui, Fukui, Japan

[10] Kyushu University, Fukuoka, Japan

[11] Korea Atomic Energy Research Institute, Daejeon, Republic of Korea

[12] Nuclear Research and Consultancy Group, Petten, Netherlands

[13] Nuclear Safety Institute of the Russia Academy of Sciences (IBRAE), Moscow, Russia

[14] Paul Scherrer Institute (PSI), Villigen, Switzerland

E-mail contact of main author: ebates@terrapower.com

Abstract. This paper presents the conclusions of a 4 year benchmark study on the simulation of the EBR-II SHRT-45R experiment. The SHRT-45R experiment was an unprotected loss of flow transient where pump dynamics, natural convection, core, and mechanical behavior played a large role in passively and safely limiting the power and temperature rise of the fuel assemblies. Participants from China, Germany, France, Italy, Japan, Korea, Netherlands, Russia, Switzerland, and the U.S.A presented transient reactor system modeling results for a variety of instrumented parameters, including core outlet temperatures, pump flow rates, and fission power. The final coordinated research project (CRP) meeting held April 2016 in Vienna summarized key findings and sensitivity studies completed after the experimental data was released and the benchmark study converted from blind to open.

The fidelity and methodology of core and system models varied greatly between participants. It was found that accurate simulation of the pump coastdown, system pressure drop, and coolant and radial expansion feedbacks strongly influenced the fission power and temperatures in the core during the transient. Relatively simple models for radial expansion were sufficient to capture the behavior during the transient, in part due to the simpler mechanical dynamics of EBR-II's core and the applicability of the point-kinetics model. Reactor core outlet (Z-pipe and IHX) temperatures were somewhat difficult to match due to the high modeling fidelity required to capture the temperature at the specific thermocouple location. Faulty subassembly flow meter data from XX09 and XX10 prevented a more accurate study of the core flow redistribution occurring during the pump coastdown. Uncertainties and variations in heat transfer and subassembly pressure drop correlations, and fuel expansion assumptions were found to have little effect on the prediction of fission power and temperature. Overall, the benchmark of the SHRT-45R was a valuable exercise that facilitated the development of state-of-the-art models for sodium fast reactor system and neutronic reactivity feedbacks.

Key Words: Fast reactor safety, reactivity feedback, benchmark study

1. Introduction

The Experimental Breeder Reactor II (EBR-II) historical record contains a large amount of valuable data on the performance, phenomena, and behavior of metal fueled sodium fast reactors (SFRs). Due to the availability of data and range of conditions experienced, the SHRT-17 and SHRT-45R test results were compiled and formed into an international benchmarking exercise led by Argonne National Laboratory (ANL) and the IAEA from June of 2012 to April of 2016 [1, 2]. The benchmark participants were blind to the experimental results until phase 2, which began mid-2014. Dozens of conference and journal publications were released by individual organizations, and the complete results have been incorporated into an IAEA technical document [3].

In SHRT-17 and SHRT-45R, the reactor safely demonstrated a protected (i.e., w/scram) loss of flow (PLOF) and an unprotected (w/o scram) loss of flow (ULOF), respectively. SHRT-45R was particularly significant because it proved that a metal fueled SFR can achieve enhanced and passive safety goals related to severe accidents [4] [5]. A similar ULOF from full power has not been repeated at any other electricity producing nuclear reactor in the 30 years since the SHRT-45R test was conducted in 1986. Thus, simulating the experiment represents a special test of current SFR system modeling capabilities; specifically, the implementation of reactivity feedbacks occurring during such an extreme transient. This paper covers the lessons learned from the 13 organizations from 10 nations that chose to model the SHRT-45R test in the benchmark period. The focus of this paper is the thermal-hydraulic and neutronic features affecting the accuracy of predicting power and flow in the reactor using system models. Another paper [6] summarizes more specific results pertaining to the behavior of the XX09 and XX10 instrumented assemblies during these transients.

2. EBR-II and the SHRT-45R Test

The benchmark data package compiled by ANL gave detailed information on EBR-II's core loading pattern, compositions, reactivity feedback coefficients, and primary coolant system including intermediate heat exchanger (IHX). No details were given on the layout of the secondary system, so participants had to impose boundary conditions on the IHX (secondary flow rates and sodium temperature). The reactor utilizes a large cold pool with thermal inertia and the pump coastdown occurs rapidly (<2 minutes), so the secondary side assumptions have a relatively small effect on the peak core temperatures and behavior in this experiment. Thus, the main challenge in capturing the peak temperatures is to correctly model the primary pump coastdown, primary circuit frictional losses, temperatures, reactivity feedbacks and power changes (fission, decay) occurring during the loss of flow and transition to natural circulation.

3. Summary of Modeling Approaches

3.1. Primary System Models

Prediction of the transient flow rate through the core requires accounting for the hydraulic (e.g., friction, gravitational, acceleration, pumps) and thermal energy balance (e.g., heat transfer to IHX tubes) of the coolant traveling through the primary system. EBR-II used two centrifugal pumps each with branched, parallel, and throttled pathways to restrict the flow rate provided to the outer core (blankets and reflectors) relative to the inner core (fueled drivers, reflectors, control rods and experimental assemblies). Another unique feature of EBR-II is the "Z-pipe" which directly connects the enclosed reactor outlet plenum volume to the IHX, creating a loop-in-pool design. The Z-pipe includes an auxiliary electromagnetic pump

(EM) that operated on battery power (with an increased in supplied voltage at 10 minutes). More detailed descriptions of the primary system are available in the benchmark summary document [3].

With the exception of ANL, TerraPower, and China Institute of Atomic Energy (CIAE) who all used versions of the SASSYS-1/SAS4A system and core modeling codes, all participants adopted unique codes to simulate the core, primary components, and the transient. Most codes were originally developed for SFR application, while some were created for light water reactor (LWR) system analysis and were extended in application to SFR. TABLE I summarizes the organizations, abbreviations, code developers, original usage, and dimensionality for the various system codes used in the benchmark.

TABLE I: SYSTEM ANALYSIS CODES USED TO SIMULATE SHRT-45R.

Org.	**Abbr.**	**System Code(s)**	**Developer**	**Original use**	**Dimensionality**
China Institute of Atomic Energy	CIAE	SASSYS-1/SAS4A	ANL	SFR	1D volumes, axial flow
North China Electric Power University	NCEPU	SAC-CFR	NCEPU	SFR	3D capability, porous media
Xi'an Jiatong University	XJTU	THACS	XJTU	SFR	1D volumes, axial flow
Institute for Radiological Protection and Nuclear Safety	IRSN	CATHARE (v. V2.5_3)	CEA, EDF, IRSN, AREVA	LWR	1D volumes, axial flow
Karlsruhe Institute of Technology	KIT	SIMMER-III v.3E	JAEA, CEA, KIT	SFR	2D (radially symmetric)
Politecnico di Torino	POLITO	FRENETIC	POLITO	SFR	No system model
University of Fukui	U. Fukui	[-NETFLOW++ -RELAP5-3D] (ANSYS CFX for CFD)	-U.Fukui, -INL	-LWR, SFR -LWR	[1D volumes, axial flow] (CFD for 3D)
Korea Atomic Energy Research Institute	KAERI	MARS-LMR	KAERI	MARS developed for LWR	1D volumes, axial flow
Nuclear Research and Consultancy Group	NRG	SPECTRA (ANSYS CFX for CFD)	NRG	Developed for LWR, SFR, and HTR	1D volumes, axial flow (CFD for 3D)
Nuclear Safety Institute of the Russia Academy of Sciences	IBRAE	SOCRAT-BN	IBRAE	SFR	1D volumes, axial flow
Paul Scherrer Institute	PSI	TRACE (V5.0) (has 3D porous media capability)	NRC	LWR	1D volumes, axial flow
Argonne National Laboratory	ANL	SASSYS-1/SAS4A	ANL	SFR	1D volumes, axial flow
TerraPower	TP	SASSYS-1/SAS4A	ANL	SFR	1D volumes, axial flow

Other abbreviations: Nuclear Regulatory Commission (NRC), Idaho National Laboratory (INL), Commissariat à l'énergie atomique et aux énergies alternatives (CEA), Electricite de France (EDF), Japan Atomic Energy Agency (JAEA), High Temperature Reactor (HTR)

A couple participants (NCEPU, KIT) approximated the two separately coasting down pumps as a single pump coastdown. KIT, CIAE, and POLITO relied on the phase 2 results for pump #2 flow rates, and the latter two did not have system models for the primary sodium. The rest of the participants implemented the homologous pump model according to the benchmark specification equations, which allow for the pump speed to be converted into head. Most participants used simplified approaches to modeling the EM pump in the Z-pipe, and some did not model the voltage increase occurring at 10 minutes (e.g., TP).

The discretization of volumes in the primary system varied by almost 6 orders of magnitude; for example, some participants (TP, IRSN) used a single (fully mixed) volume to represent the cold pool. Others, notably U. Fukui and NRG implemented highly detailed CFD codes

(ANSYS CFX) containing up to 800,000 cells to model the details of the temperature distribution in the cold pool. NCEPU also implemented a large number of volumes (22 x 32 x 27) to represent the cold pool.

3.2. Core model and Reactivity Feedbacks

The EBR-II core contains 637 subassemblies (SA) arranged in 16 hexagonal rings. To simulate the subassemblies in the system code, they are typically first grouped and then approximated as channels that capture the average behavior of a large number of fuel pins. The 1D flow channels are axially and radially discretized to varying degrees, and greater details on this can be found in the summary document. With the exception of KIT (who used a radially symmetric discretization with cylinders and sequential annuli to represent portions of the core) all participants adopted the typical multiple, 1D channel approach to discretizing the inner core and outer core (blanket + reflectors).

To simplify computation, there are usually fewer channels than there are subassemblies. This approximation was made by all organizations except POLITO, who modelled every assembly individually [7]. Note that all properties (e.g., geometry, reactivity feedback, power, flow, temperature, etc.) of all subassemblies grouped into a single channel designation are inherently identical after being averaged, so the specific method of grouping assemblies could affect the net, modelled, behaviour. Since most of the reactivity feedbacks occur from the thermal expansions occurring in the fuelled, inner core, the level of detail in that region is more important. As part of the benchmark, ANL also provided

- Core expansion reactivity feedback coefficients for coolant, Doppler, radial, axial fuel and cladding, for axial slices of every assembly throughout the core. Radial expansion reactivity feedback coefficients were calculated by ANL assuming a uniform increase in the spacing of the fuel. Note that a separate benchmark exercise on feedback coefficients was carried out within the project [8].
- Control rod worth vs. insertion depth, which allows for the expansion of the control rod drivelines (CRDL) to be modelled.
- Delayed neutron data and reactor power history necessary for point kinetics and decay heat, respectively.

TABLE II summarizes the participant’s choices for channel grouping, reactivity feedback, and decay heat models.

TABLE II: CHANNEL GROUPING, REACTIVITY FEEDBACKS, AND DECAY HEAT.

Org.	**# of inner core ch.**	**# of outer core ch.**	**Justification for channel grouping**	**Reactivity feedbacks modeled**	**Reactivity feedbacks omitted**	**Decay heat model**
CIAE	8	5	P/F Ratio, SA Type	C,D,R,AF,AC	CRDL	✓
NCEPU	4	5	SA Type	R	C,D,AF,AC,CRDL	
XJTU	4	2	SA Type	C,D,R,AF,AC,CDRL		✓
IRSN	4	2	SA Type	D	C,R,AF,AC,CRDL	
KIT***	17 rings	17 rings	P/F Ratio, SA Type	C,D,R,AF,AC†	CRDL	✓
POLITO	127	510	1 to 1	C,D†	R,AF,AC, CRDL	
U. Fukui	8	2	SA Type	C,D,R,AF,AC,	CRDL	✓
KAERI	9	2	P/F Ratio, SA Type	C,D,R,AF,AC,CDRL		✓
NRG	10	2	SA Type	C,D,R,AF,AC,CDRL		✓
IBRAE	13	2	SA Type	C,D,R,AF	AC, CDRL	✓
PSI	4	3	P/F Ratio, SA Type	C,D,R,AF ,CDRL	AC	✓
ANL	13	9	P/F Ratio, SA Type	C,D,R,AF,AC,CDRL		✓
TP	10	3	P/F Ratio, SA Type, reactivity feedback	C,D,R,AF,AC,CDRL		✓

Abbreviations: P: Power, F: Flow, SA: Subassembly, C: Coolant (density), D: Doppler, R: Radial, AF: Axial fuel, AC: Axial cladding, CRDL: Control rod driveline †: Used spatial kinetics method

***Activity carried out together with Kyushu University (KU), Japan. KU developed specific SIMMER Equation of State for EBR-II metal fuel.

With regard to radial expansion, some core codes (e.g., SAS4A) use approximations to account for the conical, non-uniform radial displacements created by the EBR-II core restraint system. KIT's SIMMER and POLITO's FRENETIC did not require the pre-calculated reactivity feedback coefficients because they use unique internal neutron solver coupled with the thermal hydraulic results. With exception of KIT and POLITO who implemented spatial kinetics core models, all participants used the point kinetics approximation.

4. Results and Discussion

4.1. Core Power and Pump Flowrate

Although there were many quantities measured throughout the core and primary system, key measures of performance in the benchmark were the flow rate and fission power predictions (decay heat was not measured). The fission power prediction is the culmination of accurately predicting a large number of phenomena including flow, power, reactivity feedback, and temperatures. The peak temperatures and reactivity feedbacks occur at approximately 40-50 seconds into the transient. After 600 seconds, the core power is practically stabilized and determined by the decay heat production of the fuel. FIG. 1 compares the final predictions for fission power vs. time with the experimental measurements and FIG. 2 shows the predictions for pump #2 flow rate vs. time.

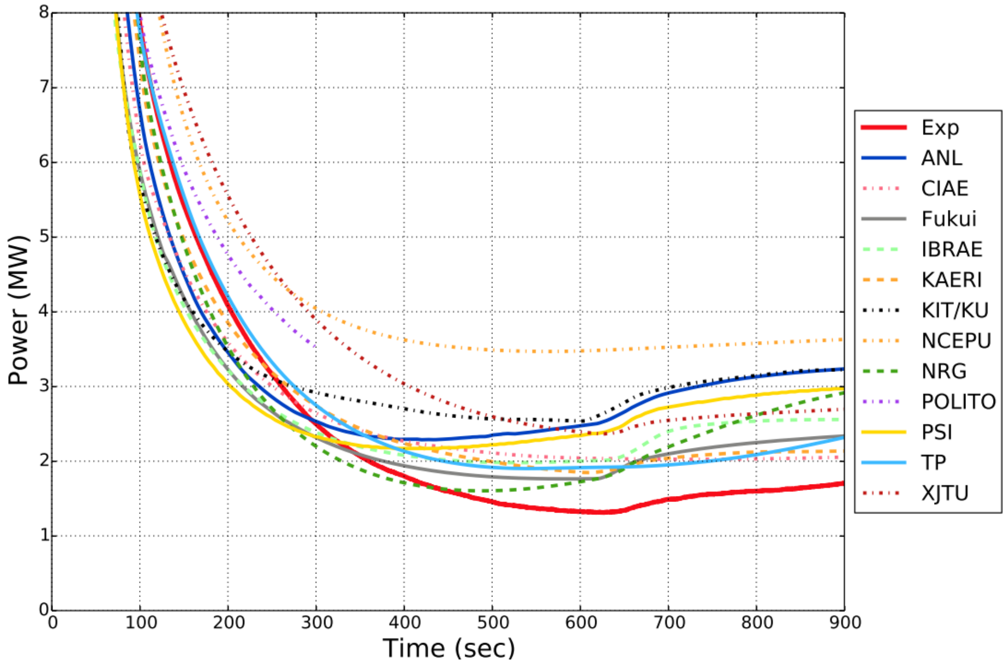

FIG. 1. SHRT-45R fission power (final phase predictions and experimental data in red) vs. time.

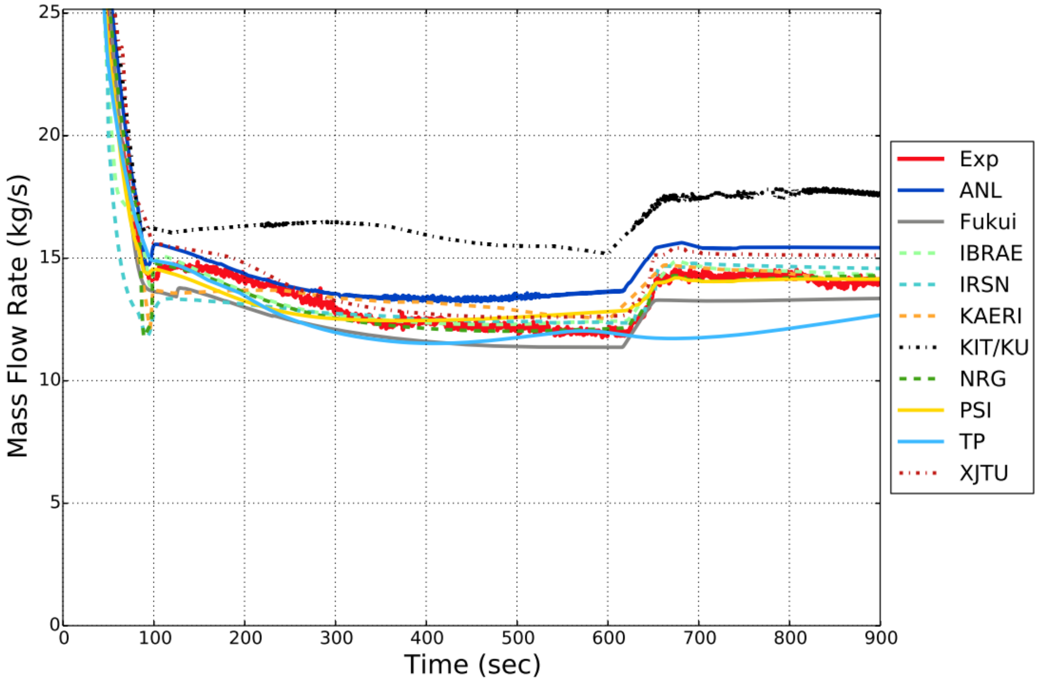

FIG. 2. SHRT-45R pump flow rate (final phase predictions and experimental data in red) vs. time.

4.2. Reactivity Feedbacks

It was generally agreed among participants that the coolant and radial expansions were the dominant feedbacks in passively limiting the power of the reactor. The time of the maximum reactivity feedback (and temperature) is predicted to occur at 60 ± 20 seconds and the peak reactivity feedback prediction varies from approximately -30¢ to -65¢. FIG. 3 shows a comparison of the predicted reactivity feedbacks vs. time. Note that the predictions for the components of the reactivity feedback showed much more variation than the total (especially considering that some participants did not model certain feedbacks).

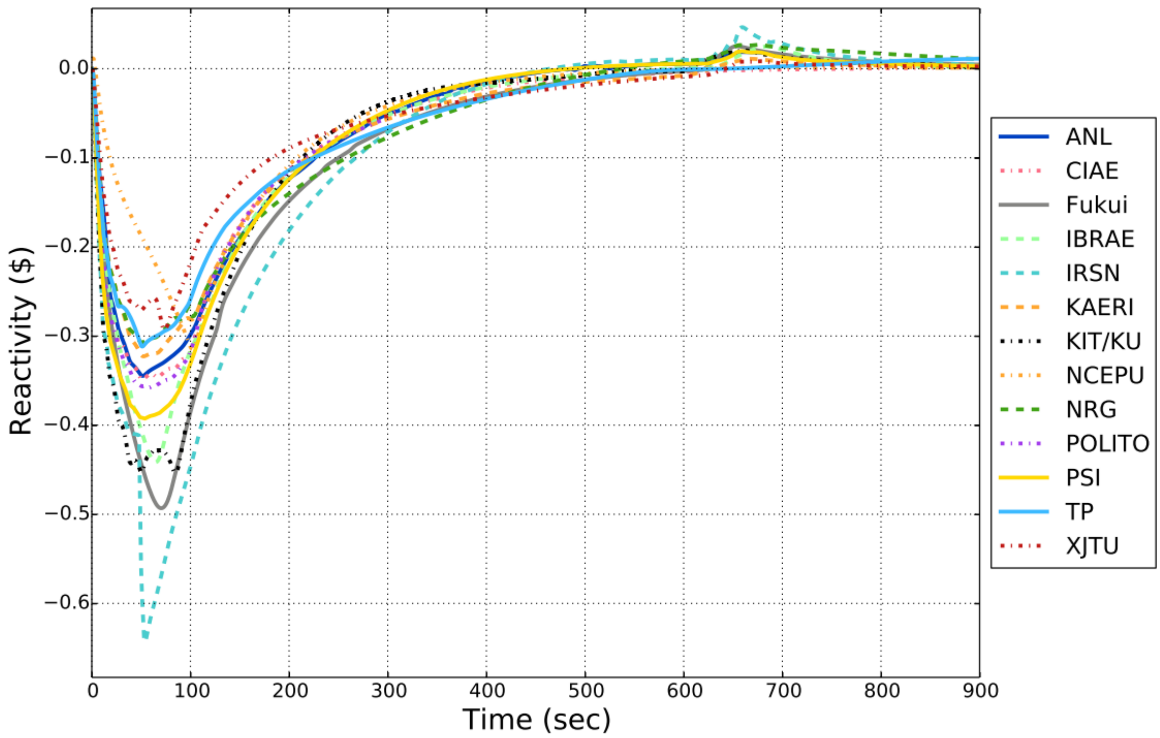

FIG. 3. SHRT-45R phase 2 total reactivity feedback (final phase predictions in red) vs. time.

Among the participants that modeled all the reactivity feedbacks and provided results for the components, there was not a consensus on the exact ranking of feedbacks at the time of peak temperatures. ANL and KAERI's results agreed and showed the feedbacks decreasing in importance from C>R>AF,AC>CRDL>D, whereas TP predicted radial dominating over coolant expansion. PSI's results showed the feedbacks decreasing in a slightly different order of C>AF>R>CRDL>D. All these participants used simple approximations for the radial and expansion of the core, which proved to be sufficient to predict the SHRT-45R fission power (some within few percent at the time of peak temperatures).

With regard to modeling choices, the 9-13 average channels used by KAERI, ANL, and TP appears to be sufficient to capture the heterogeneity of the 127 subassemblies of the inner core. Increasing to 1-1 (e.g., POLITO) did not significantly change reactivity feedback predictions (within the wide band of variations between predictions). The more complex CFD pool models used by U. Fukui and NRG don't appear to significantly improve the accuracy of flow rates, although the detailed models were better at predicting the reactor outlet and IHX temperatures.

4.3. Sensitivity Studies

In the final phases of the benchmark, some participants chose to conduct additional sensitivity studies to better understand important input parameters and modeling assumptions. TABLE III summarizes the sensitivity studies related to reactivity feedbacks and system models.

TABLE III. SYSTEM MODEL AND NEUTRONIC SENSITIVITY STUDIES CONDUCTED ON SHRT-45R.

Org.	**Summary of sensitivity studies**	**Findings**
KIT	Used alternative neutronics codes to calculate reactivity feedback coefficients.	Sodium density coefficient varied from -2.15 pcm/K to -1.98 pcm/K.
POLITO	Compared the predictor corrector quasi-static method to the point kinetics method with phase-space dependent operators.	Results in agreement within the imposed accuracy- point kinetics is acceptable.
U. Fukui	Analyzed uncertainties due to cross section variances [9].	Uncertainty in feedback coefficients due to cross sections is <6%.
	Pump friction torque varied [10].	Pump 1 has a larger friction torque compared to pump 2.
KAERI	Minor loss coefficient varied (Reynold's dependence) to improve pump prediction.	All factors made power and flow predictions closer to experimental data.
	Decay heat models varied.	
	Axial feedback reduced by 10%.	
NRG	Heat transfer correlations were varied between Mikityuk and Ushakov.	Results not sensitive to heat transfer coefficient.
	Fuel assumed to be solid (instead of 15% gas and 10% sodium porosity).	Noticeable effect on fuel temperatures.
	Doppler feedback modeled logarithmically vs. linearly	Little to no effect.
	Decay heat model varied to 11 groups	Noticeable effect on fission and decay power.
IBRAE	Parametric variations of coolant, radial, axial, Doppler, and decay heat ±10%.	Peak cladding temperature variation of ~20 °C, power variation <2%.
PSI	Parametric variations of coolant, radial, CRDL, axial, delayed neutron fraction.	All effects were less than 2 °C on the peak cladding temperature.
ANL	Specified core flow rate instead of using a primary model.	Reduced error in predicting power from 60% to 33%.
	Conducted parametric studies on reactivity feedbacks.	Varying coolant and radial expansion by large factors of 4-5 (respectively) is required to match power exactly. Axial and CRDL have minor effects.
	Upper plenum wall heat transfer coefficient varied from 50-1300 W/m^2-K and structure heat capacity was varied.	700 W/m^2-K produces the best agreement for Z-pipe inlet temperature vs. time. Variations in heat capacity insufficient to match results.
	Z-pipe wall heat transfer coefficient (heat loss to cold pool) varied from 904 to 5000 W/-m^2-K.	Still could not match IHX temperature profile. Concluded that IHX temperatures must be affected by specific location of thermocouple.
TP	Coolant reactivity feedback from reflector assemblies included. Assemblies grouped in channels by reactivity feedback.	Significant improvement in fission power predictions.
	Pressure drop correlations for subassemblies were varied.	Minor variations in core flow distribution.
	Heat transfer correlations for subassemblies were varied.	Results not sensitive to heat transfer coefficient.
	Axial expansion models were varied to range from fuel-driven to cladding-driven expansion.	Peak power not affected significantly. Error in predicting end of transient power affected by fuel expansion assumptions.

5. Conclusions

Overall, the benchmark study was a highly valuable experience and gives a good indication of the ability of a wide range of system codes to capture the highly complex phenomena during the SHRT-45R transient. The most reliable measurements available for comparison were the fission power and primary pump #2 flow rates. Pump behavior, coolant expansion, and radial expansion feedbacks were found to be the most important modeling assumptions. Regarding sensitivity studies, core heat transfer coefficients, assembly pressure drop correlations, and assumptions on fuel axial expansion and CRDL expansion, did not have large effects. Z-pipe temperatures could be matched, but this required sensitivity studies on the wall heat transfer coefficient of the upper plenum region. Other thermocouples (e.g., IHX) and flow meter measurements (XX09, pump #1) proved to be in locations difficult to model, or were from faulty instruments that were not replaceable. Investigating the IHX measurements led to the realization that the recorded data at this location did not represent an average temperature, and therefore could not be predicted accurately by a standard systems analysis approach. Thus, one of the key lessons learned from the benchmark was the importance of measurement device placement, uncertainty, and the ability to question the accuracy of experimental measurements.

REFERENCES

[1] L. Briggs, S. Monti, W. Hu, D. Sui, G. H. Su, L. Maas, B. Vezzoni, U. Partha Sarathy, A. Del Nevo, A. Petruzzi , R. Zanino, H. Ohira, H. Mochizuki and K. Morita, "EBR-II Passive Safety Demonstration Tests Benchmark Analyses- Phase 2," in *16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics*, Chicago, IL, Aug 30- Sep 4, 2015.

[2] L.Briggs, C. Choi, W. Hu, L. Maas, W. Maschek, B. Merk, K. Mikityuk, H. Mochizuki, S. Monti, K. Morita, A. D. Nevo, H. Ohira, A. Petruzzi, U. P. Sarathy, A. Shin, I. Shvetsov, M. Stempniewicz, D. Sui, B. Truong and G. H. Su, "Benchmark Analyses of the Shutdown Heat Removal Tests Performed in the EBR-II Reactor," in *International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13)*, Paris France, March 4-7, 2013.

[3] International Atomic Energy Agency, "Fourth RCM of the IAEA CRP on Benchmark Analyses of an EBR-II Shutdown Heat Removal Test (Ref. No: IAEA I31021-CR-4)," Vienna, Austria, April 26-29, 2016.

[4] H. P. Planchon, "Implications of the EBR-II inherent safety demonstration test," *Nuclear Engineering and Design,* vol. 101, pp. 75-90, 1987.

[5] T. Sofu, "A review of inherent safety characteristics of metal alloy sodium cooled fast reactor fuel against postulated accidents," *Nuclear Engineering and Technology,* 2015.

[6] P. Uppala, E. Bates, G. Su, A. D. Nevo, H. Ohira, H. Mochizuki, M. Stempniewicz, B. Truong, A. Moisseytsev and T. Sumner, "Thermal Hydraulic Investigation of EBR-II Instrumented Subassemblies during SHRT-17 and SHRT-45R Tests," in *International Conference on Fast Reactors and Related Fuel Cycles (FR17)*, Yekaterinburg, Russia, 2017.

[7] D. Caron, R. Bonifetto, S. Dulla, V. Mascolino, P. Ravetto, L. Savoldi, D. Valerio and R. Zanino, "Full-core coupled neutronic/thermal-hydraulic modeling of the EBR-II SHRT-

45R transient," *Int. J. Energy Res.,* 2016.

[8] B. Vezzoni, W. G. van Rooijen, T. Fei, Y. Zhang, M. Marchetti, P. Balestra and C. Parisi, "IAEA Neutronics Benchmark for EBR-II SHRT-45R," in *International Conference on Fast Reactors and Related Fuel Cycles (FR17)*, Yekaterinburg, Russia, 2017.

[9] W. G. van Rooijen and H. Mochizuki, "Analysis of the EBR-II SHRT-45R Unprotected Loss of Flow Experiment with ERANOS and RELAP," *Science and Technology of Nuclear Installations,* p. 14, 2015.

[10] H. Mochizuki, K. Muranaka, T. Asai and W. G. van Rooijen, "Benchmark Analyses for EBR-II Shutdown Heat Removal Tests SHRT-17 and SHRT-45R," *Nuclear Engineering and Design,* vol. 275, pp. 312-321, 2014.