BENCHMARK ANALYSIS OF EBR-II PROTECTED LOSS-OF-FLOW TRANSIENT

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ABSTRACT

Coordinated Research Project (CRP) on EBR-II Shutdown Heat Removal Tests (SHRT) was established by International Atomic Energy Agency (IAEA). The objective of the project is to support and to improve validation of simulation tools and projects for Sodium-cooled Fast Reactors (SFR). The Experimental Breeder Reactor II (EBR-II) plant was a uranium metal-alloy-fuelled liquid-metal-cooled fast reactor designed and operated by Argonne National Laboratory (ANL) for the U.S. Department of Energy at the Argonne-West site.
In the frame of this project, benchmark analysis of one of the EBR-II shutdown heat removal tests, protected loss-of-flow transient (SHRT-17), has been performed at the Gruppo di Ricerca Nucleare San Piero a Grado (GRNSPG) in Pisa, Italy.
The aim of this paper is to present modeling of EBR-II reactor design using RELAP-3D, and to present results of the transient analysis of SHRT-17. Complete nodalization of the reactor was made from the beginning. Model is divided in primary side that contains core, pumps, reactor pool and, for this kind of reactor specific, Z pipe, and intermediate side that contains Intermediate Heat Exchanger (IHX). Core was modeled with 82 channels that represent all fuel assemblies, and 14 channels for reflector and blanket assemblies.
After achievement of acceptable steady-state results, transient analysis was performed. Starting from full power and flow, both the primary loop and intermediate loop coolant pumps were simultaneously tripped and the reactor was scrammed to simulate a protected loss-of-flow accident. In addition, the primary system auxiliary coolant pump, that normally had an emergency battery power supply, was turned off. Despite early rise of the temperature in the reactor, the natural circulation characteristics managed to keep it at acceptable levels and cooled the reactor down safely at decay heat power levels.
Thermal-hydraulics characteristics and plant behavior was focused on prediction of natural convection cooling by evaluating the reactor core flow and temperatures and their comparison with experimental data that were provided by ANL.

KEYWORDS
EBR-II, Sodium-cooled fast reactors, RELAP5-3D, protected loss-of-flow

1. INTRODUCTION

Experimental Breeder Reactor II (EBR-II) was a pool type Sodium-cooled Fast Reactor (SFR) located in Idaho, U.S.A., and it was designed and operated by Argonne National Laboratory. Operation began in 1964 and continued until 1994. During the last 15 years, EBR-II was used for experiments to demonstrate the importance of passive safety in liquid metal reactors (LMR).

In order to demonstrate the inherent safety of LMR type reactor, several loss of flow tests were conducted between 1984 and 1986, as a part of Shutdown Heat Removal Test (SHRT) series. SHRT-17, protected loss of flow transient, was one of the mentioned tests. At the beginning of the test the primary pumps were tripped and at the same time full control rod insertion was done. With this test it was successfully demonstrated the effectiveness of natural circulation cooling capability of the reactor, which makes them inherently safe under described accident conditions.

During normal operation the removal of heat is quite an efficient process, but during a transient such as loss of flow, decay heat becomes major concern for the safety of the reactor. The removal of decay heat is done by taking advantage of high thermal capacity of the heat structures and the primary fluid; and primary system, completely submerged into the pool of sodium, provides a passive heat removal system.

Experimental data, benchmark specifications and data requirements [1] were released as a part of an IAEA Coordinated Research Project (CRP) which aim was to improve the simulation capabilities in the fields of research and design of SFR through data and codes validation and qualification.

This paper will discuss description of the facility, SHRT-17 test and their RELAP5-3D thermal hydraulic model. This is followed with analysis of steady-state results and following protected loss of flow transient. The plan is to improve the model in future with replacing current models of core and pool with 3D models, and, eventually, coupling with neutronic codes for more accurate results.

2. DESCRIPITON OF THE EXPERIMENT

2.1. EBR-II Facility

The EBR-II plant was located in Idaho, and it was operated by ANL from the beginning of 1964 until 1994. EBR-II rated thermal power was 62.5MW, with electric output of 20 MW.

This reactor is a pool type reactor, with all major primary system components submerged in the sodium tank which contains approximately 340 m^3 of liquid sodium. Overview of the modeled system can be seen in Figure 1. Two primary pumps pump sodium from the pool, through high pressure and low pressure pipes to the high pressure inlet plenum and to the low pressure inlet plenum. The reactor vessel consists of 637 hexagonal subassemblies divided in three regions: central core, inner blanket and outer blanket. Central core consists of 61 subassemblies, precisely 2 safety rods, 8 control rods, and the rest was driver-fuel or experimental subassemblies of different types: Driver, Partial Driver, High flow Driver, Steel/Dummy, Instrumented. The inner region does not contain blanket subassemblies, but it was named as such because originally had blanket subassemblies. The outer blanket region contains the rest of the 510 subassemblies that were either blanket or reflector subassemblies.

Sodium from high pressure inlet plenum is distributed in the central core and inner blanket regions, and it was approximately 80-85% of total primary mass flow. The rest of the primary mass flow was distributed from low pressure inlet plenum to outer regions blanket and reflector subassemblies.

After passing through the reactor core, flow mixes at the upper plenum, and through reactor outlet pipe ("Z-pipe") enters Intermediate Heat Exchanger (IHX). Sodium then exits the IHX back into the primary sodium tank before entering sodium primary pumps.

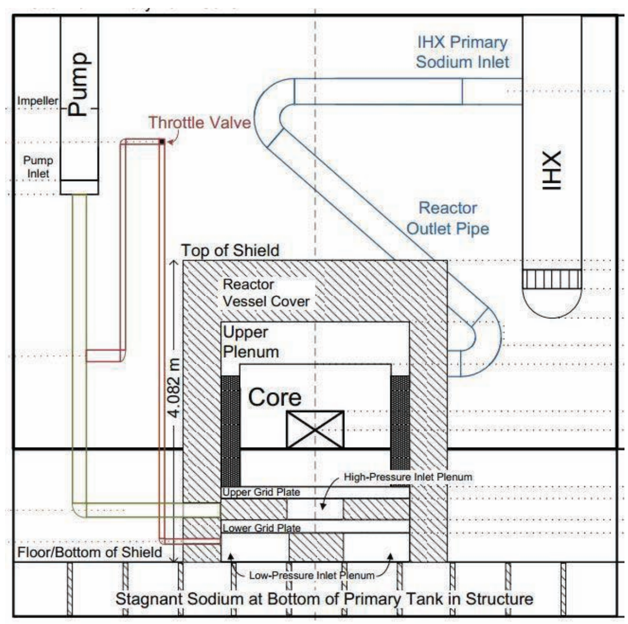

Figure 1. EBR-II Benchmark Model of Primary Vessel Components

2.2. SHRT-17 Test

Before the start of the transient, EBR-II was operated at full power and full flow for two hours in order to limit decay heat after the scram and that temperatures throughout the system had reached an equilibrium state. The control rods were active to allow control rod insertion at the start of the transient period.

The SHRT-17 transient was initiated by a trip of primary and intermediate pumps, which instantaneously scrammed the reactor. Primary pump flow coast down was governed by the kinetic energy stored in motor-generator sets. The auxiliary electromagnetic pump in the primary loop was irrelevant for this test because it was turned off and didn't receive power from battery backups.

Decreasing of the reactor decay power continued due to fission product decay. No automatic or operator action was taken after the start of the SHRT-17 test.

Table I. summarizes the initial conditions and transient initiators for the SHRT-17 test [1].

3. CODE AND NODALIZATION

The EBR-II benchmark specifications and designs [1] were used to develop the thermal hydraulic model of the reactor. The RELAP5-3D system thermal hydraulic code was used for preparing the nodalization [2]. The RELAP5-3D is an outgrowth of the one-dimensional RELAP5/MOD3 code developed at Idaho National Laboratory (INL) for the U.S. Nuclear Regulatory Commission (NRC). It is a transient, two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain non-condensable components in the vapor/gas phase and/or a soluble component in the liquid phase.

Table I. SHRT-17 transient description

Parameter	Condition
Initial Power	57.3 MW
Initial Primary Coolant Flow Through Core	456.7 kg/s
Initial Intermediate Coolant Flow	301 kg/s
Initial Core Inlet Temperature	624.7 K
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

Given the fact that the density of liquid sodium is much higher than water, a sliced approach was necessary for avoiding any kind of oscillations in the code calculation and unrealistic pressure differences among parallel flow paths. The sliced approach is a nodalization technique consisting in dividing the hardware in parallel slices in order to have the centers of each nodes in parallel pipes at the same elevation position. This is a good practice to better reproduce phenomena connected with natural circulation where small gravitational head difference play a relevant role.

A detailed nodalization, see Figure 2, reproducing each geometrical zone of the loop has been developed.

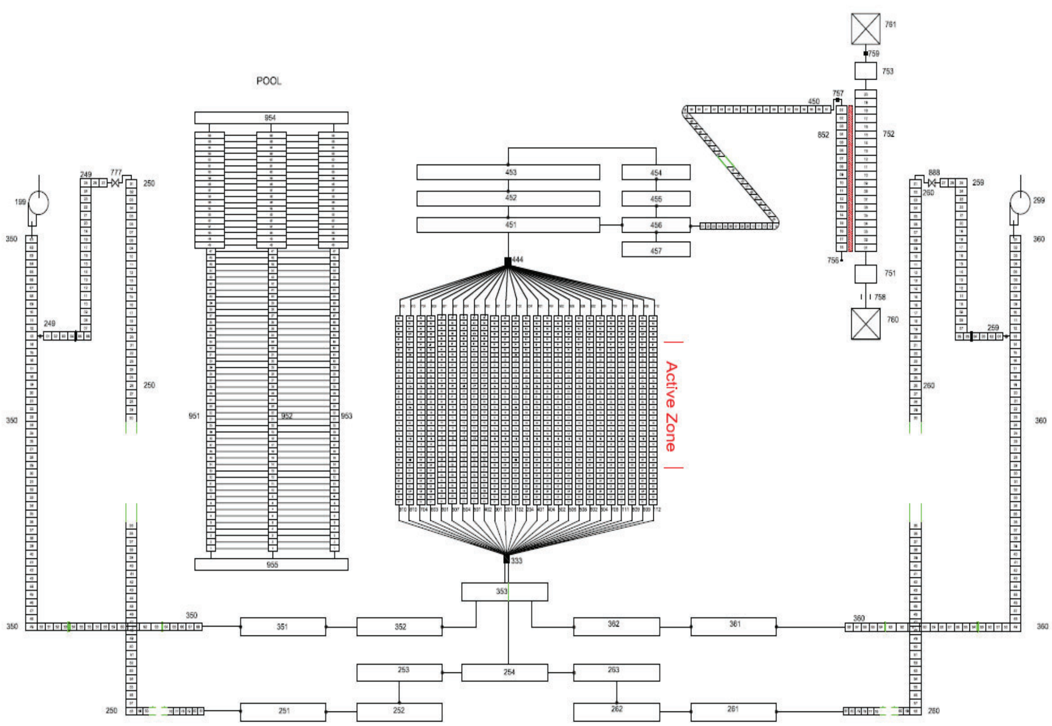

Figure 2. RELAP5-3D Nodalization of EBR-II Reactor

Hereafter some significant aspects of the nodalization development are summarized.

Core model
The core consists of 96 channels that are representing all 10 types of subassemblies used in the reactor, and two bypasses. As mentioned before reactor vessel consists of central core, inner blanket and outer blanket regions. These regions have first been divided into 16 rows. First 6 rows represent central core, and all channels have been modeled separately; except for safety/control rods that have been merged into one channel that represents high flow bypass. Rows 7 to 16 consist of reflector or blanket assemblies, and they have been modeled with one channel per type of assembly in each row. Each channel has been made up by 36 thermal hydraulics volumes, where active part of the reactor core has 24 volumes.

Pool model
Three parallel pipes with a different cross-section at the top and bottom, connected with branches at top and bottom are representing EBR-II reactor pool model. All the nodes between these pipes were connected to simulate mixing of sodium between them. The modeling of the pool as all other components followed the sliced approach and every pipe has been divided into 66 volumes.

Intermediate heat exchanger
Intermediate side of EBR-II reactor has been represented with the intermediate heat exchanger. The heat exchanger model is of counter-current flow type. The primary side of IHX has been modeled as a pipe, which takes inlet from the "Z-pipe" and the exit is connected to the pool. The intermediate side of IHX has also been modeled by a pipe equivalent to 3026 secondary tubes through which the intermediate sodium flows.

The boundary conditions for the intermediate side are imposed by the time dependant volume and time dependant junction.

Heat structures
An effective model of heat structures was developed to ensure the efficient heat transfer through the structures. As the whole high pressure and low pressure pipes are inside the pool, therefore cylindrical heat structures were constructed to take care of heat exchange between the pool and the piping. Following Z-pipe unusual design special heat structure that simulates stagnant sodium between inner and outer pipe has been developed. For the shield surrounding the core, three equivalent plates were constructed, each plate connected to one of the pipes of the pool.

For heat exchanger the cylindrical geometry was used. Secondary side is made up of around 3000 tubes so an equivalent length was used for the heat transfer.

The heat structure for each assembly was made by dividing the assembly in three parts: the active core, the gas plenum, and the hexagonal structure that connects each channel with the neighboring one. Active core heat structure was made taking into consideration the number of pins in the assembly and the number of assemblies. The gas plenum heat structure represents part of the assembly that is above active part and it filled with helium gas. The hexagonal structure was made for hexagonal tube that surrounds the whole assembly.

4. ANALYSIS OF CALCULATION RESULTS

Due to the proprietary reason the following figures do not include comparison against the recorded data.

4.1. Steady-State Calculations

A steady state calculation, by running the code with the “TRANSNT” (transient) option for 1400 s has been completed. Steady-state conditions were obtained after around 250 s, due to large passive heat structures and fine nodalization of assemblies.

As it can be seen at Figure 3, mass flows from low pressure plenum, high pressure plenum, outlet plenum, and one assembly (chosen as an example) are quickly stabilized. Problems were observed (Figure 4) with the power transferred from primary side of the reactor to intermediate side through intermediate heat exchanger heat structures. Although conditions have been stabilized after 250 s, discrepancies between rated and transferred power exists. Problems with modeling the pool, and recirculation in it, were identified as one of possible reasons for that.

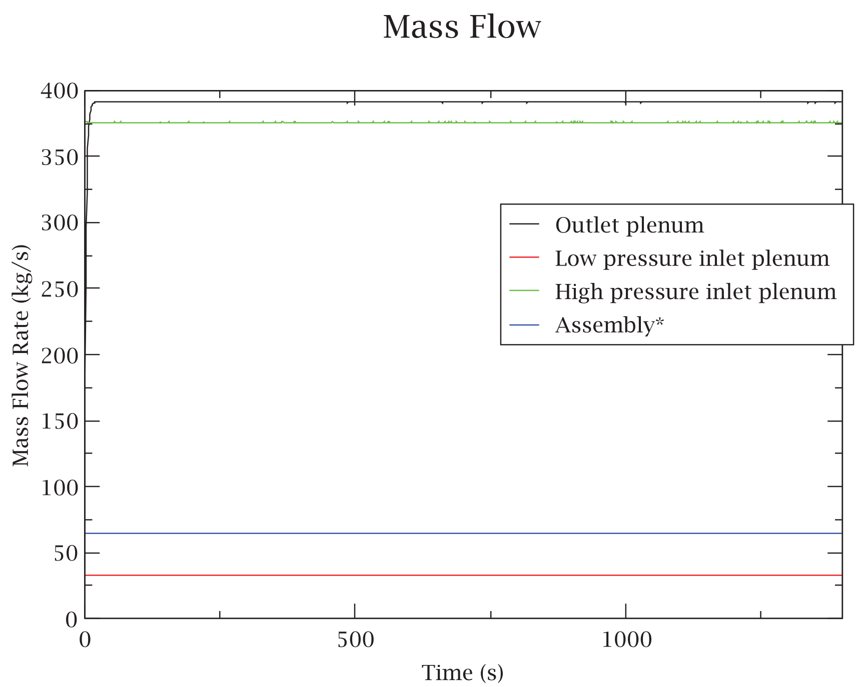

Figure 3. Mass Flow at Inlet, Outlet and One Assembly of the Core

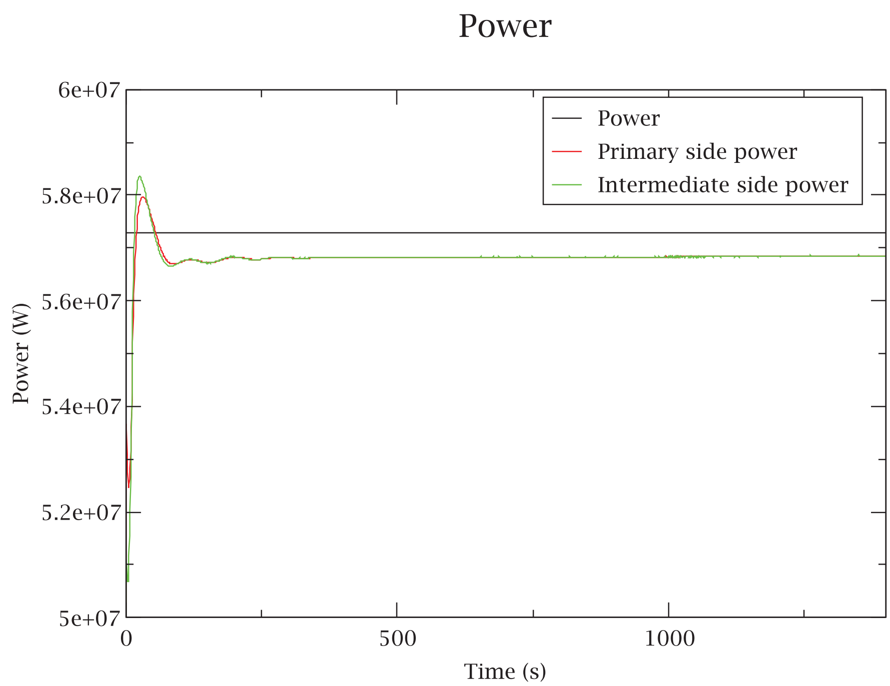

Figure 4. Power vs. Time

4.2. Transient Analysis

After obtaining steady-state conditions transient calculation was performed. The loss of flow transient is initiated by the primary pump trip, and both pumps were stopped simultaneously. The control rods are active and the reactor is scrammed instantaneously. Table II gives list of events [1]. The intermediate side boundary conditions are imposed by gradually reducing the mass flow by 95% (Figure 5) and inlet temperature by 3% during the transient. Power was imposed by provided table [1], unlike by the control variables for steady-state calculation.

Table II. SHRT-17 transient description, imposed and resulting events

Event	Value
Pump coastdown initiation	0 s
Scram of the reactor	0 s
Pump coastdown end	90 s
Primary power <10% nominal power	9 s
Minimum mass flow through pump (value)	3.5 kg/s
Minimum mass flow through pump (time)	96 s
Minimum mass flow through assemblies	1.5-2 %
Secondary side mass flow trip	0 s
Secondary side temperature trip	0 s
End of Transient	900 s

The EBR-II primary pumps should be “identical”, but pump #2 has higher frictional torque losses and stops sooner during the coastdown. This has been shown on Figure 6. The natural circulation through the loop has been demonstrated with mass flow through primary pumps after they were stopped.

Figure 7 shows axial distribution of hottest channel’s cladding temperatures, were it can be observed a fall in temperature as the scram occurs, but then the temperature starts to increase because the mass flow decreases considerably due to stopping of the pump, and later again we have slow decreasing. This confirms that natural circulation can effectively cool down the reactor.

The inlet and outlet temperatures of IHX primary and intermediate side are shown in Figure 8 and Figure 9. The temperature peaks can be observed here as well, in correspondence with the core temperature peaks.

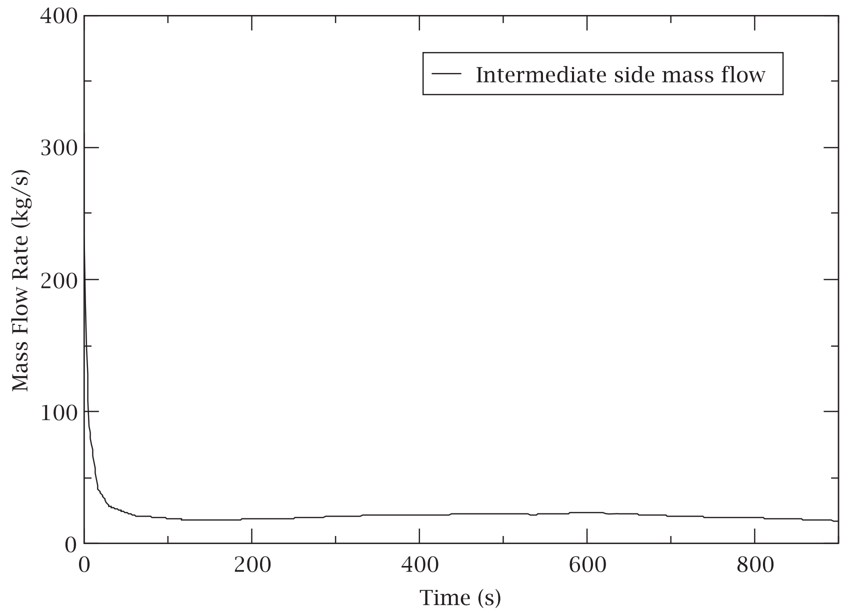

Figure 5. Intermediate Side Mass Flow

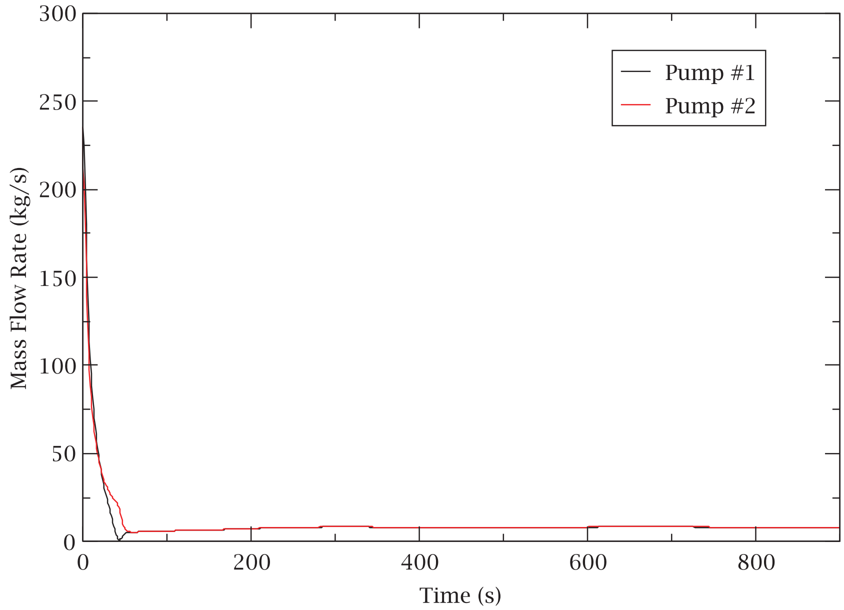

Figure 6. Primary Pump Mass Flow

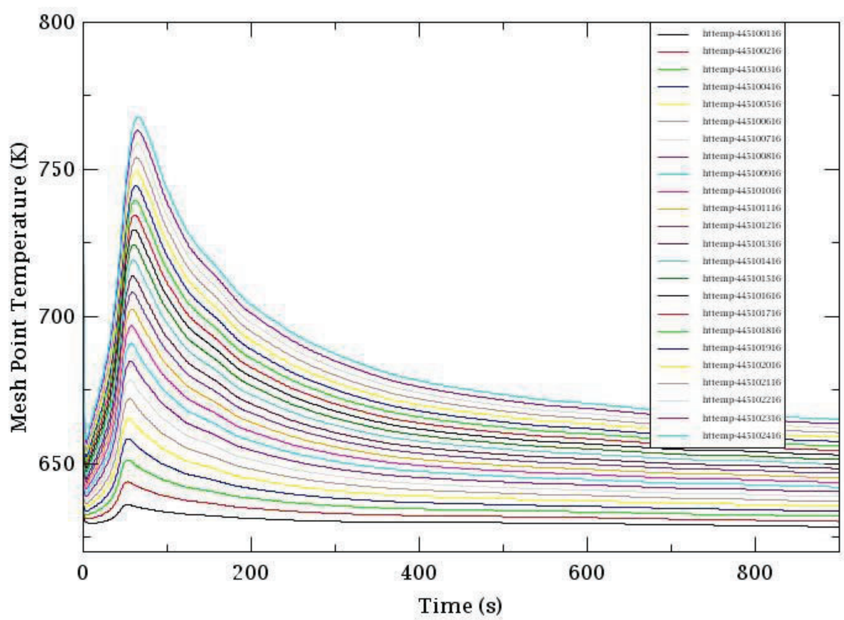

Figure 7. Axial Distribution of Cladding Temperatures in the Hottest Channel

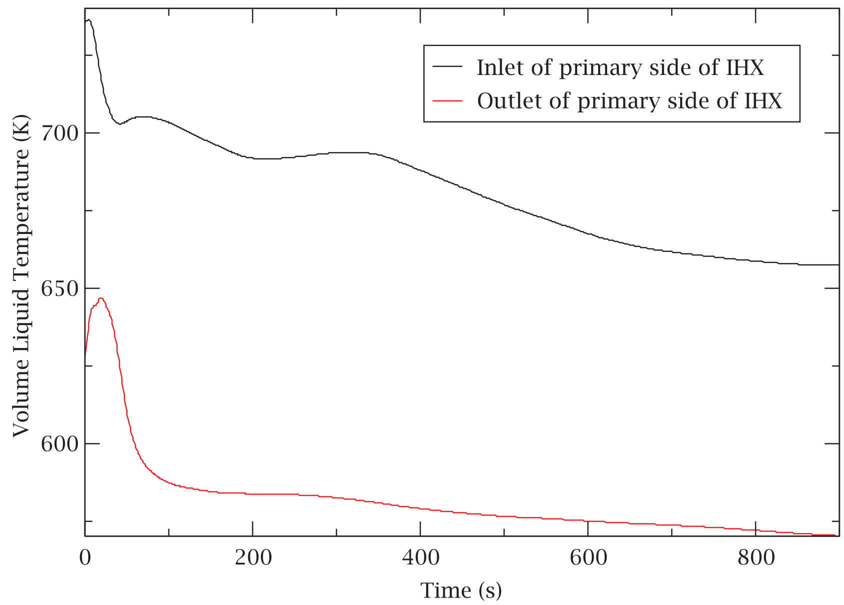

Figure 8. Primary Side of IHX Inlet and Outlet Temperatures

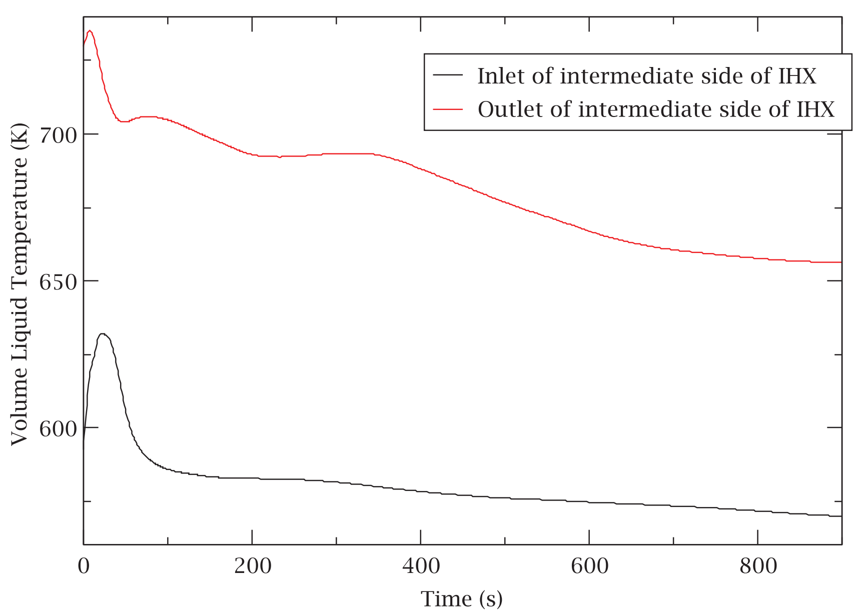

Figure 9. Intermediate Side of IHX Inlet and Outlet Temperatures

5. CONCLUSIONS

This paper briefly presented description of EBR-II facility, and SHRT-17 transient. One of the objectives of the SHRT-17 transient was to demonstrate passive reactor shutdown and decay heat removal in response to protected loss of flow transient. The obtained results showed that the natural phenomena, such as expansion of the sodium coolant and thermal inertia of the primary sodium pool, can be effective in successful cooling of the EBR-II type reactor.

The results showed before refer to the phase 1 of the benchmark, during which the blind simulation was performed. These results, compared with the experimental data, show that:

- some parameters are in good agreement with the experimental data such as the primary pumps mass flow rate and the coolant temperature in the inlet plena;
- others parameters have some margins for improvement, as the Z-pipe inlet temperature and the IHX primary side inlet temperature.

These discrepancy are mainly due to a non-adequate modeling of the upper plenum and to the heat losses to the cold pool or possible thermal stratification in the Z-pipe.

During the phase 2 of the project, which is developing during the writing of this paper, a series of modeling refinements should be implemented in order to improve the accuracy of the results. These refinements include replacing the core with 3d model were all assemblies will be represented 1:1, replacing power imposed by table with preparing 0-D neutron kinetics model, making completely new 3D sodium pool model and improve the IHX model.

In scope of IAEA's coordinated research project on EBR-II shutdown heat removal tests, SHRT-45R transient [3] will also be analyzed in future activities.

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