Simulation of Loss-of-flow tests using MARS-LMR

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1. Introduction

Sodium fast reactor is being developed as one of generation IV reactor concepts to meet increasing energy demand in the world widely and, to reduce the volume of high level waste. To keep up with the world wide trend, Korea Atomic Energy Research Institute is developing core technologies for a future sodium fast reactor. The effort of core technologies in the area of safety analysis is focused on a system transient analysis code.

The MARS (Multi-dimensional Analysis for Reactor Safety) code is used to the analysis of transients for water cooled reactor systems [1]. To use the code to the analysis of transients for a liquid metal cooled reactor system, several models, such as, a liquid metal coolant property table, wall heat transfer coefficients related to liquid metal, and the friction factor correlations associated with wire spacers of fuel rod were implemented to the code then, the code was named as MARS-LMR.

For the verification of the feasibility of the MARS-LMR with new features, the SHRT-17, 39, and 45 tests conducted in the EBR-II reactor were analyzed by using the code. It was a pool-type plant with the reactor, primary pumps and the intermediate heat exchanger (IHX) submerged into the large volume of sodium pool.

2. Description of loss flow tests

Three of SHRT tests were selected for the verification of the MARS-LMR. The SHRT-17 is loss of flow test with reactor scram. Objectives of the test was to demonstrate and provide experimental data for the transition to natural circulation cooling following a sudden loss of forced circulation of primary sodium due to the loss of all AC power to the main primary pumps and auxiliary pumps. The test was initiated by tripping the main pumps and scramming the reactor from full power and flow[2].

The SHRT-39 and 45 are loss of flow tests by coolant pump trips without permitting a reactor scram. The SHRT-39 test was conducted from initial conditions of 100 % power 100 % flow, and a primary tank bulk temperature of 352 °C. Two primary pump coastdowns were actively controlled which the main power supply to the pump drive trains and the coastdown period was about 300 seconds. The SHRT-45 was carried out from initial power level of 100 %, primary flow of 100 % and a tank bulk temperature of 343 °C. The coastdowns of primary pumps were passively controlled wherein the main pump driving power was lost. These conditions were similar to those of a station blackout and the coastdown period was about 100 seconds. The reactor powers decreased due to the negative reactivity feedback associated with increased reactor temperatures. The resulting pump coastdown curves and reactor power for three tests are shown in Figs. 1 and 2.

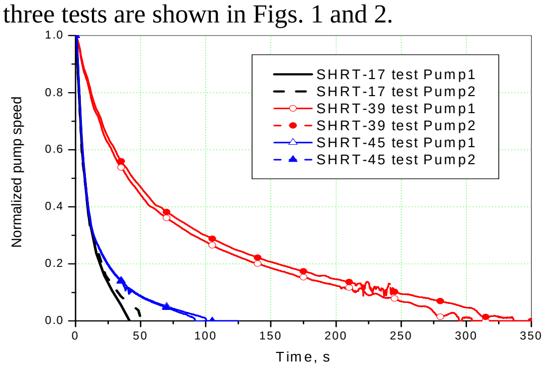


Fig. 1 Coastdown curves of main pumps

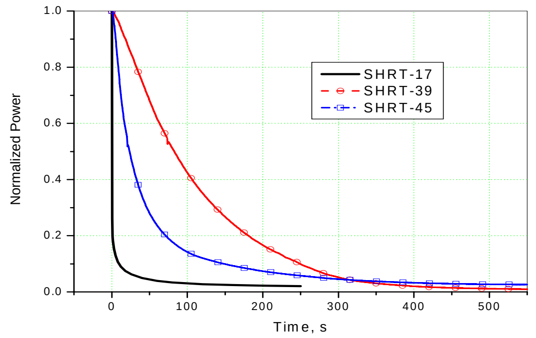


Fig. 2 Normalized Power histories

3. Simulation of the SHRT tests

In the primary system of EBR-II two main primary pumps take a suction of sodium from the pool and discharge it to inlet pipes. Then the primary flow is divided into two streams, one entering the high pressure chamber feeding fueled driver subassemblies and the other entering the low pressure chamber feeding blanket and reflector subassemblies. The sodium is heated through the core region and mixes in an outlet plenum of the reactor. Then the sodium goes through the outlet 'Z' type pipe and to the IHX in which it transfers its heat to the sodium of intermediate loop. The primary sodium leaving the IHX dumps directly back into the primary tank. The EBR-II plant was nodalized as shown in Fig. 3 which provides a schematic representation of MARS-LMR model.

The SHRT-17 transient temperature behavior near the top of the XX09 is shown in Fig. 4. After the reactor scram at time zero the power drops very rapidly; this leads to a rapid drop in coolant temperature. At the same time, pumps coast down and stop; this lead to higher coolant temperature until natural circulation heads build up and increase coolant flow rates. Again, the coolant temperature is decreased due to the increased flow rate.

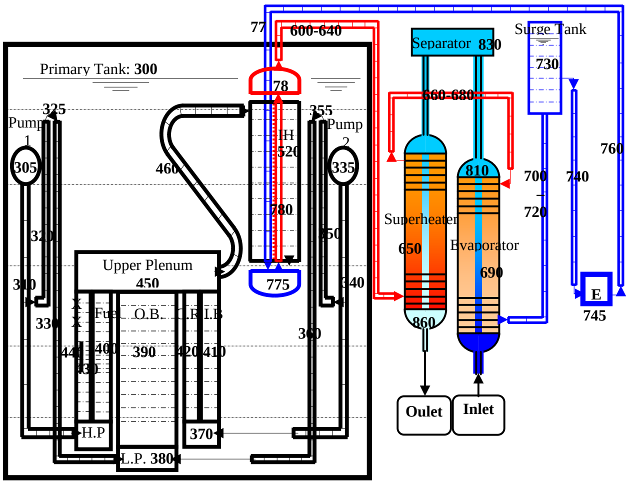


Fig. 3 MARS-LMR Nodalization for EBR-II plant

The trends in the measurements are consistent with the calculated data. The temperature behaviors after the coast down of the pump agree well but, the calculated temperature in the natural circulation regime is slightly high. This is caused by not to model the heat transfer through the duct wall with neighbor subassemblies. But the overall temperature behavior is very similar to the experimental data.

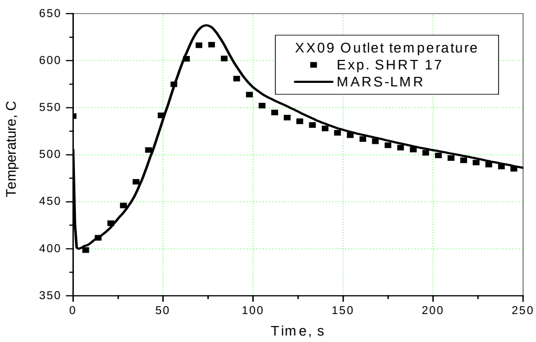
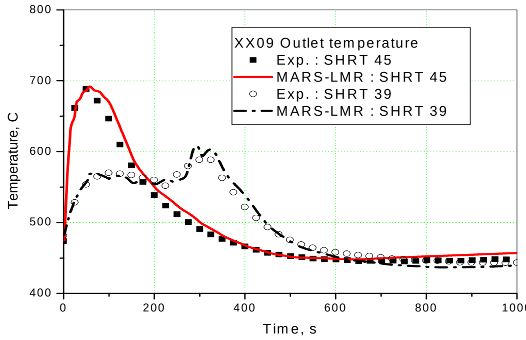


Fig. 4 MARS-LMR Nodalization for EBR-II plant

The peak temperatures near the top of XX09 subassembly are 592.4 °C in the SHRT-39 and 688.5 °C in the SHRT-45 while the MARS-LMR code predicts 608.5 °C and 691.8 °C respectively as can be seen in Fig. 6. As comparison of the SHRT-39 and 45, the transient peak temperatures are significantly reduced by longer pump coastdown times. There is good agreement between experimental data and calculation results. This suggests that the important thermal hydraulic phenomena during power reduction and coupled natural convective heat removal were modeled in the MARS-LMR code.



4. Conclusion

The MARS-LMR was verified through the analysis of three SHRT tests conducted in the EBR-II reactor. The SHRT-17 test involved simultaneous loss of electrical power to all pumps and a reactor scram from 100 % power and flow. The other two tests, SHRT-39 and 45 were the loss of flow tests by coolant pump trips without a reactor scram. It was shown that the MARS code predicted the experimental data for flow rate and temperature for an instrumented subassembly with good success and models of the code related to LMR were appropriate.

ACKNOWLEDGMENTS

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