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Assessment calculation of MARS-LMR using EBR-II SHRT-45R

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HIGHLIGHTS

- Neutronic and thermal-hydraulic behavior predicted by MARS-LMR is validated with EBR-II SHRT-45R test data.
- Decay heat model of ANS-94 give better prediction of the fission power.
- The core power is well predicted by reactivity feedback during initial transient, however, the predicted power after approximately 200 s is over-estimated. The study of the reactivity feedback model of the EBR-II is necessary for the better calculation of the power.
- Heat transfer between inter-subassemblies is the most important parameter, especially, a low flow and power subassembly, like non-fueled subassembly.

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ABSTRACT

KAERI has designed a prototype Gen-IV SFR (PGSFR) with metallic fuel. And the safety analysis code for the PGSFR, MARS-LMR, is based on the MARS code, and supplemented with various liquid metal related features including sodium properties, heat transfer, pressure drop, and reactivity feedback models. In order to validate the newly developed MARS-LMR, KAERI has joined the International Atomic Energy Agency (IAEA) coordinated research project (CRP) on "Benchmark Analysis of an EBR-II Shutdown Heat Removal Test (SHRT)". Argonne National Laboratory (ANL) has technically supported and participated in this program. One of benchmark analysis tests is SHRT-45R, which is an unprotected loss of flow test in an EBR-II. So, sodium natural circulation and reactivity feedbacks are major phenomena of interest. A benchmark analysis was conducted using MARS-LMR with original input data provided by ANL. MARS-LMR well predicts the core flow and power change by reactivity feedbacks in the core. Except the results of the XX10, the temperature and flow in the XX09 agreed well with the experiments. Moreover, sensitivity tests were carried out for a decay heat model, reactivity feedback model, inter-subassembly heat transfer, internal heat structures and so on, to evaluate their sensitivity and get a better prediction. The decay heat model of ANS-94 shows better results of fission power, however, the fission power is still over-estimated in the long-term transient region by the reactivity feedbacks. The inter-subassembly heat transfer is the most influential parameter, especially for the non-fueled XX10, which has a low flow and power subassembly. In addition, the appropriate internal heat structure model can be an influential parameter. Finally, the corrected results are proposed with reasonably conjectured parameters. This study can give the validation data for the MARS-LMR and better understanding of the EBR-II SHRT-45R.

1. Introduction

An Experimental Breeder Reactor II (EBR-II), which is located at Idaho National Laboratory (INL), was operated from 1964 to 1994 by Argonne National Laboratory (ANL) for the U.S. Department of Energy (DOE). The EBR-II was rated for a thermal power of 62.5 MW with an electric output of 20 MW. In 1974, a thermal hydraulic testing program at the EBR-II conducted to support the continued safe and reliable operation of the EBR-II was primarily directed toward understanding the detailed response of the EBR-II to a wide variety of accident conditions and utilizing this knowledge to validate general-purpose thermal-hydraulic-neutronic system analysis codes for application to new plant designs. They used in-core subassemblies XX07, XX08, XX09, and XX10 (Poloncsik et al., 1982; Singer et al., 1977; Gillette et al.,

Abbreviations: SFR, sodium-cooled fast reactor; EBR-II, Experimental Breeder Reactor-II; SHRT, Shut-down Heat Removal Test; ANL, Argonne National Laboratory; IHX, intermediate heat exchanger; GP, grid plate; ACLP, above core load pad; CRDL, control rod drive-line; RV, reactor vessel.

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Nomenclature

r_0	initial reactivity at time zero [$]
r_B	bias reactivity [$]
r_{si}	user input reactivity at i-th node [$]
V_{ci}	control variable for user-defined for reactivity at i-th node
$W_{\rho i}$	weighting factor for density reactivity at i-th node
R_ρ	density reactivity [$]
a_{Wi}	temperature coefficient for density reactivity at i-th node
T_{Wi}	sodium mean temperature at i-th node
W_{Fi}	weighting factor for Doppler reactivity at i-th node
R_F	Doppler reactivity
T_{Fi}	fuel mean temperature at i-th node
a_{Fi}	temperature coefficient for Doppler reactivity at i-th node
C_b	boron concentration
B_W	differential boron worth
r^A	fuel rod axial expansion reactivity
r^R	core radial expansion reactivity
$r^{CRDL/RV}$	control rod driveline and reactor vessel expansion reactivity
C_f^A	fuel expansion reactivity coefficient
C_c^A	clad expansion reactivity coefficient
$\varepsilon_{f,i}$	fuel strain at i-th node
$\varepsilon_{c,i}$	clad strain at i-th node
C^R	core radial expansion reactivity coefficient
W_{GP}	weighting factor for grid plate expansion
W_{ACLP}	weighting factor for above core load pad expansion
ε_{GP}	grid plate strain
ε_{ACLP}	above core load pad strain
W	control rod worth, which is function of insertion length
Z_0	initial control rod location
ΔZ_{CRDL}	control rod driveline displacement
ΔZ_{RV}	reactor vessel length displacement

1980) to get a real-time detection of the in-core sodium temperature and flow rate. XX07 and XX08 data during various transients were used to validate the whole core/subassembly code COBRA-WC (Khan et al., 1981), SSC (Madni, 1984), MINET (Van Tuyle, 1983), etc. Based on early experimental works, the Shutdown Heat Removal Test (SHRT) program was developed by the Department of Energy (DOE). Major goals for the SHRT program are demonstrations of passive removal of decay heat by natural circulation of primary sodium coolant, passive reactor shutdown following a loss of forced circulation, passive reactor shutdown following a loss of heat sink, and the generation of test data for validating computer codes used in the design, licensing, and operation of LMRs. The total number of tests was 58 with five types: (A) loss of flow (LOF)/scram to natural circulation, (B) scram with delayed LOF to natural circulation, (C) reactivity feedback characterization, (D) LOF without scram, and (E) loss of heat sink (LOHS) without scram, are selected.

The International Atomic Energy Agency (IAEA) launched a program, the "Benchmark Analysis of an EBR-II Shutdown Heat Removal Tests," as a part of an IAEA coordinated research project (CRP) in 2012, which is technically supported by ANL (Briggs et al., 2013, 2015). Various institutes in various countries joined this program to compare a benchmark analysis with their own codes. The program has major three tasks: a system analysis of the SHRT-17 test and SHRT-45R, and neutronic analysis of the SHRT45R (Sumner and Wei, 2012). The SHRT-17 and SHRT45R are a loss of flow test with scram and without scram, respectively. KAERI has currently designed a prototype Gen-IV sodium-cooled fast reactor (PGSFR), whose safety analysis code is the MARS-LMR. To validate the MARS-LMR code, Korea Atomic Energy Research Institute (KAERI) has participated in this IAEA-CRP. The reactivity feedback and thermal hydraulic behaviors in the MARS-LMR were validated with EBR-II SHRT-45R test data. Moreover, sensitivity tests for certain parameters were conducted to get a better prediction and understanding of physical phenomena during the EBR-II SHRT-45 test. For example, the effects of decay heat models, reactivity feedbacks, heat transfer between subassemblies, and internal heat structure are studied to evaluate their sensitivity.

2. Overview of EBR-II reactor plant

The EBR-II plant is illustrated in Fig. 1. All major system components are submerged in the primary pool, which contains about 340 m^3 of liquid sodium at 370 °C. Two primary pumps draw a sodium flow from the cold pool to their outlet pipe, which is bifurcated into two inlet plenums, high-pressure and low-pressure plenums, which are controlled by a throttle valve on the top of the pipe connected to the low-pressure inlet plenum. Subassemblies in the inner core (fuel driver region) and extended core regions receiving sodium from the high-pressure inlet plenum, accounting for approximately 85% of the total core flow. Fig. 2 shows a high- and low-pressure inlet plenum. The blanket and reflector subassemblies in the outer core (blanket region) receive sodium from the lower-pressure inlet plenum. Hot sodium from a core outlet flows into an upper plenum and mixes before going through the Z-shaped pipe, referred to as a Z-pipe, and into the intermediate heat exchanger (IHX). Then, the cooled sodium through the IHX flows back into the primary pool before entering the primary sodium pumps again. Therefore, EBR-II has only a single primary pool and the upper plenum is connected to the Z-pipe. The sodium in the intermediate loop traveled from the IHX to the steam generator, where its heat was transferred to the balance-of-plant (BOP).

The core consists of 637 hexagonal subassemblies. The subassemblies can be divided into three regions: core, inner blanket (IB), and outer blanket (OB). Since the EBR-II is an experimental reactor, the core configuration was changed to an appropriate purpose with experimental subassemblies. The central core comprised the 61 subassemblies in the first five rows. Two of these positions contained safety rod subassemblies, and eight positions contained control rod subassemblies. The remaining central core subassemblies were either diver-fuel or experimental irradiation subassemblies of various types. Rows 6 and 7 formed the inner blanket region. The subassemblies in rows 8–16 formed the outer blanket region. Fig. 3 shows the subassembly arrangement of the reactor and their types when the SHRT-45R test was conducted. In Fig. 3, subassemblies labeled with K were steel subassemblies, and subassemblies whose label begins with the letter X were experimental subassemblies. The labels D, HFD, and P identify a fuel, high-flow driver, and partial drivers, respectively. The partial driver is the driver fuel subassembly where approximately half of the fuel elements are replaced by steel elements. The label S and C identify a safety-rod and control-rod subassembly, respectively. The dark blue colored subassemblies in Fig. 3 indicate the reflector type subassemblies. The subassemblies in rows 11–16 are all blanket type subassemblies except reflector subassemblies. Instrumented subassemblies in the SHRT-45R test are XX09 and XX10 subassemblies. The XX09 and XX10 are a fuel driver and non-fueled steel subassemblies, respectively.