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Validation of the finned sodium–air heat exchanger model in MARS-LMR

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ABSTRACT

A Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR), for which the Korea Atomic Energy Research Institute (KAERI) has designed a pool type sodium-cooled fast reactor, has a decay heat removal system (DHRS). The DHRS consists of sodium-to-sodium and sodium-to-air heat exchangers and their connecting pipes. There are four loops, which have two active and two passive type sodium–air heat exchangers, that are called a finned-tube sodium-to-air heat exchanger (FHX) and a helical-tube sodium-to-air heat exchanger (AHX), respectively. Recently, Zukauskaus's air–sodium heat transfer models have been added in MARS-LMR, which is a safety analysis code for the PGSFR. In this study, to validate the newly added heat transfer models for the FHX, two experiments are selected: one is a performance test for a sodium to air heat exchanger (AHX) in the steam generator test facility (SGTF) for the prototype fast breeder reactor (PFBR), and the other is a start-up test with a dump heat exchanger (DHX) in JOYO. All validation results indicate that Zukauskaus's correlation slightly over-estimates the heat transfer. One possible reasons is a smaller number of rows in the test bundle, which was already mentioned by Zukauskaus and Karni. The RMS values for the prediction of sodium temperature for the PFBR and the JOYO are 16.25% and 27.5%, respectively. When a correction factor is applied, their RMS values improve to 3.19% and 20.71%, respectively. In addition, the MARS-LMR's prediction for the JOYO shows a much better accuracy with RMS of 7.45% when corrected sodium flow rates are applied.

1. Introduction

The Korea Atomic Energy Research Institute (KAERI) has designed a pool type sodium-cooled fast reactor, which is called the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) (Kim et al., 2013). Fig. 1 shows a schematic of the PGSFR. The PGSFR has a decay heat removal system (DHRS) has independently passive and active loops to satisfying a system diversity. The decay heat removal system (DHRS) is a designated safety grade component providing a sufficient decay heat removal capability during abnormal conditions, such as a loss of heat sink (LOHS) accident. The passive DHRS (PDHRS) relies exclusively on a natural convection heat transfer, i.e., natural circulation on the sodium side and natural draft on the air side. And its loop is equipped with one sodium-to-sodium decay heat exchanger (DHX), natural-draft sodium-to-air heat exchanger (AHX), and connecting pipes. The active DHRS (ADHRS) is operated by a EM-pump in a loop-side and a blower in the air-side. And its loop is integrated with same components in the PDHRS, except a forced-draft sodium-to-air heat exchanger (FHX) for an air-side heat exchanger. The AHX is a helical type air-sodium heat exchanger in the PDHRS, and the FHX is a finned serpentine-type air-sodium heat exchanger in the ADHRS. The DHX located in a cold pool, removes heat from a primary side and transfers heat to the loop side in the DHRS. And heat in the loop is transferred to ultimate heat sink of ambient air by capability of the AHX or FHX, which depends on the opening area of the damper at the inlet region. Under normal operation, a small amount of heat is removed by the AHX and FHX through a small open area of the damper. It was designed to prevent the solidification of sodium, and to ensure operability when it fully operates during accident conditions. The current heat removal capacities are about 0.3 MW and 2.5 MW for normal and accident conditions, respectively.

The MARS-LMR code has been used for a safety analysis of the PGSFR. The code is based on the MARS which was developed for the transient analysis of a light water reactor. The MARS code basically employs the three-dimensional transient two-fluid model for the two-phase flow system (Jeong et al., 1999). Also the point kinetics equation and the heat conduction equation are modeled to calculate the neutron behavior in the reactor core and the heat transfer from the heat structure to fluid. This code was modified to simulate the sodium thermal–hydraulics and neutronic behaviors in a transient condition for a liquid metal cooled fast reactor (Ha and Jeong, 2010). Recently, the heat transfer models for the

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Nomenclature

Nu	Nusselt number
Pr	fluid Prandtl number
Pr_w	wall Prandtl number
$Re_{D,max}$	bundle Reynolds number
Re	Reynolds number
C_1	empirical coefficient in Eq. (1)
C_z	correction factor in Fig. 6
m	empirical exponent in Eq. (1)
V	fluid velocity (m/s)
V_{max}	maximum fluid velocity in a bundle (m/s)
D	tube outer diameter (m)
t	tube thickness (m)
ρ	fluid density (m^3/kg)
μ	fluid viscosity (Pa·s)
S_T	transversal pitch (m)
S_L	longitudinal pitch (m)
S_D	diagonal pitch (m)
N	number of tubes
C_2	empirical coefficient in Eq. (7)
a	empirical exponent in Eq. (7)
b	empirical exponent in Eq. (7)
h	fin height (m)
s	fin pitch (m)
$A_{fin/fluid}$	area ratio of fin and fluid
δ	fin thickness (m)
ΔT	temperature difference of inlet and outlet (°C)

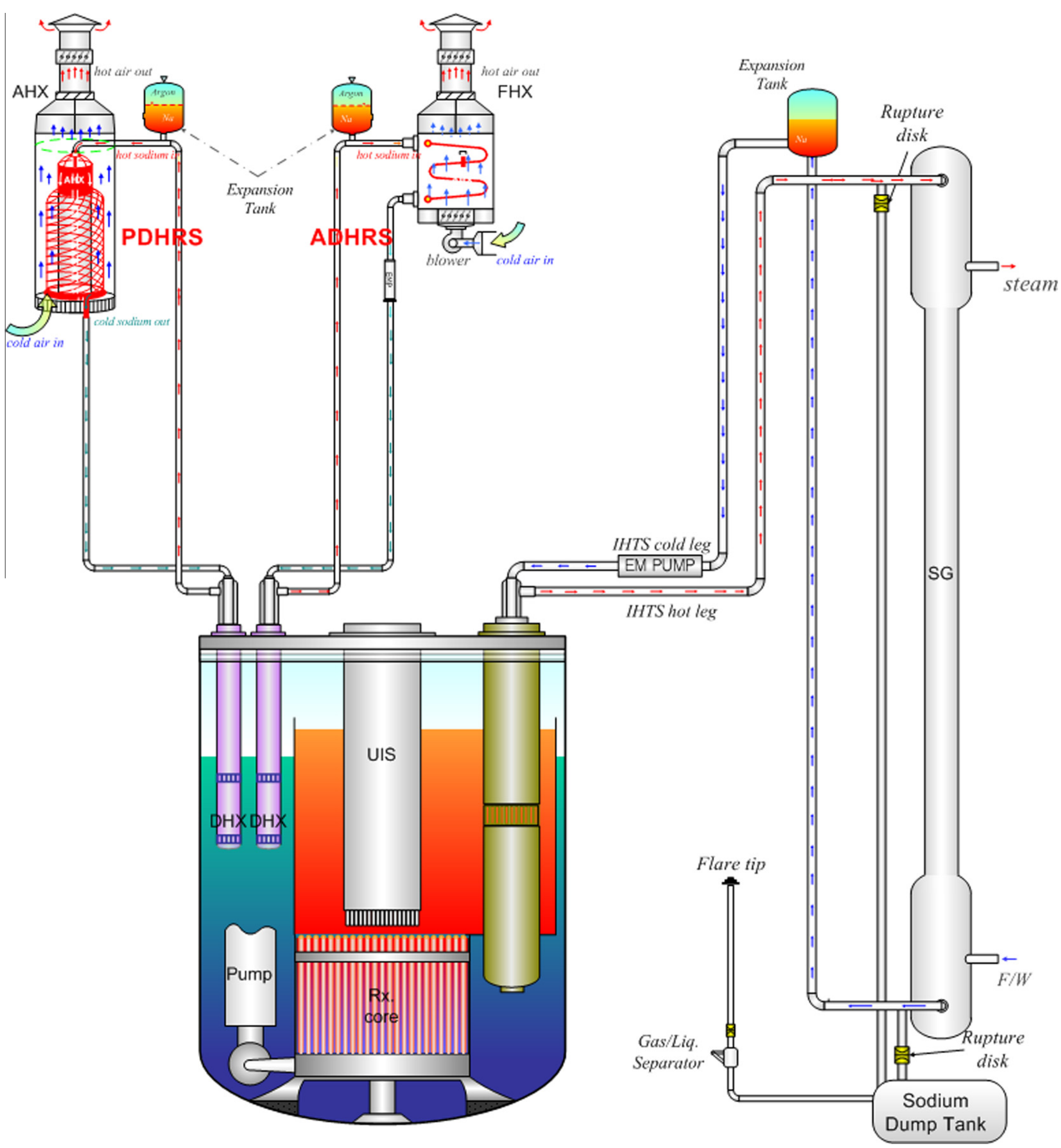

Fig. 1. Schematics of the PGSFR.