1st International & 28th National Conference on Nuclear Science & Technology 2022 (ICNST22)
Nuclear Science & Technology Research Institute

Thermal-hydraulic Analysis of irradiation capsule of TRR fuel test loop

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Abstract

Due to the importance of performing irradiation tests on domestic fuels, in this study the thermal-hydraulic analysis of an irradiation capsule, which has been designed to perform irradiation tests on three sample fuel rods in Tehran reactor test loop under thermal-hydraulic conditions that fuels will face in main reactor core, is accomplished. In spite of common thermal-hydraulic codes in nuclear industry such as COBRA-EN, Fluent software has the capability to simulate the cylindrical geometry of sample fuel rods within irradiation capsule and the rectangular geometry of fuel plates adjacent to the capsule in Tehran research reactor core, simultaneously. In present study we applied this capability to investigate thermal-hydraulic parameters of the fuel rods under test condition in irradiation capsule and the effects of loading the capsule in Tehran research reactor core on the two plate-type fuel assemblies of TRR core which are placed adjacent to capsule.

Keywords: irradiation test, irradiation capsule, thermal-hydraulic analysis, Tehran research reactor

Introduction

Newly fabricated fuels must be tested under desired neutronic and thermal-hydraulic operating condition prior to utilization in the reactor core[1]. Currently, Tehran Research Reactor (TRR), with desired neutron flux and also being equipped with a fuel test loop, is the only available national facility for performing irradiation tests and also irradiating the fuels and preparing them for post-irradiation tests in different amounts of fuel burnup[2]. TRR fuel test loop is designed and built to enhance the applications of Tehran Research Reactor in the field of fuel and nuclear materials testing and thus, to perform irradiation tests on samples of domestic fuels. This test loop with nominal pressure of 10 bar and nominal volumetric flow rate of 20 m^3/h is designed to simulate the thermal-hydraulic conditions of the reactor core in which the fuel will be used[2]. In this study, thermal-hydraulic parameters of the irradiation capsule as the in-pile section (IPS) of the loop and the effects of loading it in Tehran research reactor core on the cooling of adjacent fuel plates were investigated.

Methodology

The geometery considered in this simultion consists of a cylindical irradiation capsule with 3 sample fuel rods inside it which will be loaded in one of the irradiation position of TRR core and two fuel plates of the two TRR fuel assemblies which will be loaded adjacent to the capsule in the core, as shown in Figure 1. Power distributions in 3 sample fuel rods and two fuel plates of this geometry were obtained by MCNPX code in 10 axial levels. In order to use these power distributions in Fluent software, a 4th degree polynomial has been fitted to each power distribution and was used as user defined function (UDF). In addition, for properties which depend on temperature such as thermal conductivity, specific heat capacity and density, UDFs were written. In present work, Ansys Fluent 19.2 and k-ε realizeable model with enhanced wall treatment function were applied to obtain thermal-hydraulic parameters of concern. In order to calculate minimum departure from nucleate boiling ratio (MDNBR) within the capsule as an important safety parameter, bernath correlation[3] was applied considering coolant velocity, pressure and bulk temperature obtained by Fluent. In this study coolant pressure and mass flow rate within capsule were considered 4.4 bar and 3.9 kg/s, respectively.

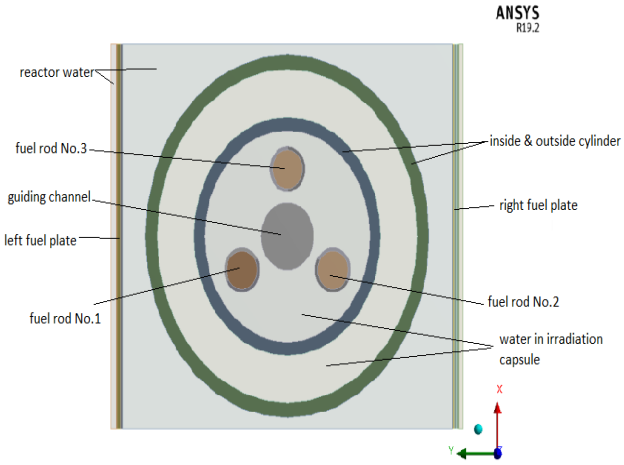


Figure 1. A schematic view of fuel rods and fuel plates

Results and discussion

Thermal-hydraulic parameters of test fuel rods and the fuel plates adjacent to the capsule, namely maximum temperature of fuel, clad and coolant (in irradiation capsule and also between the fuel plates of adjacent fuel

assemblies) were obtained and compared with safety criteria in table 1. In order to evaluate the Fluent simulation, Fluent results were compared with COBRA-EN simulation results. Average temperature of clad surface in 10 axial levels of each test fuel sample were obtained by Fluent and compared with COBRA-EN results in figure 2. Figures a, b and c depict data obtained by Fluent and COBRA-EN for 3 test fuel rods.

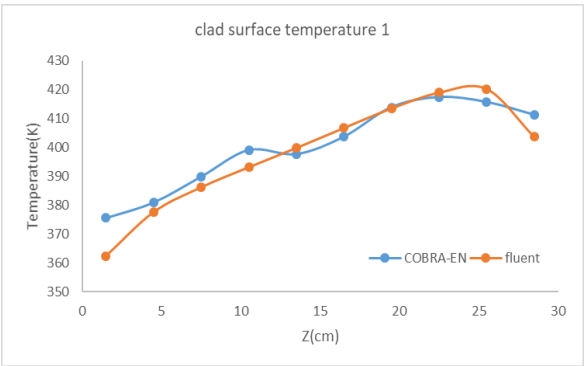
As can be seen in figure 2, the axial temperature distribution obtained by fluent is in good accordance with COBRA-EN results.

The result of MDNBR calculation based on bernath correlation shows that MDNBR is "4.91" which is much more than the safe criteria "2" determined by fuel designer.

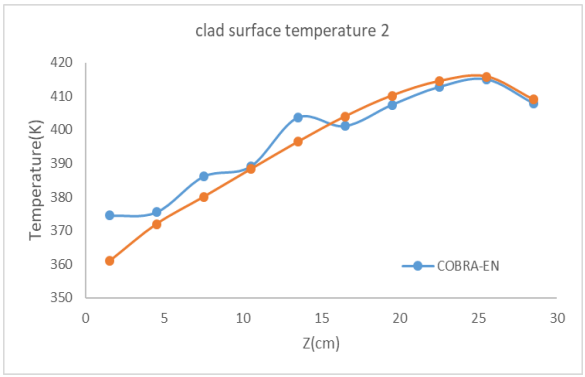
Table 1. thermal-hydraulic safety parameters of fuel rods and fuel plates

Parameter	Fuel rods		Fuel plates	
	Result of simulation	Safety criteria	Result of simulation	[4] Safety criteria
Maximum temperature of fuel(℃)	1519.16	< 2240	102.73	<400
Maximum temperature of clad(℃)	237.67	< 625	96.77	<105
Maximum temperature of coolant(℃)	93.57	<147	76.41	<116

Also outlet temperature of water IPS is 315.15 K and wall tmperature of inside and outside cylinder of IPS are 313.54 K and 312. 23 K, respectively.



(a)



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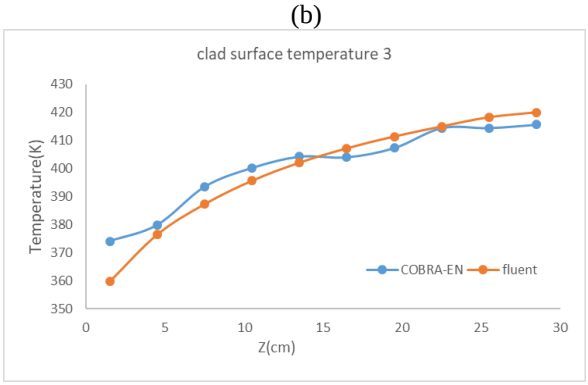

(c)

Figure 2. Compering clad surface temperature obtained by Fluent and COBRA-EN for 3 test fuel samples within capsule

Conclusions

This steady-state thermal-hydraulic calculation shows that by considering the aforemened coolant pressure and mass flow rate within the irradiation capsule, all safety criteria of test fuels as well as the fuel plates of TRR will be met during the irradiation test and therefore, safe irradiation and test of those sample fuels rods in TRR test loop is gauranteed.

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