Progress in Nuclear Energy 79 (2015) 32–39

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Preliminary thermal-hydraulic safety analysis of Tehran research reactor during fuel irradiation experiment

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ARTICLE INFO

Article history:
Received 9 August 2014
Received in revised form
29 September 2014
Accepted 20 October 2014
Available online

Keywords:
Thermal-hydraulic
Irradiation experiment
Safety
Tehran research reactor
Mixed-core

ABSTRACT

Following domestic fabrication of nuclear fuels in Iran, it is necessary to investigate fuel material behavior, fission gas release, fuel swelling, cladding material behavior and fuel integrity of domestic fuels at different burnup in a research reactor during irradiation. Currently, Tehran research reactor is the sole operating research reactor which can be used for fuel irradiation experiments in the country. In this regard, standard codes as well as developed complementary computer programs are applied to verify thermal-hydraulic performance of irradiating a domestic rod-type fuel assembly of natural UO_2 pellets in Tehran research reactor core, which itself contains 20% enriched plate-type U_3O_8–Al fuels. Maximum temperatures of fuel, clad and coolant, onset of nucleate boiling, onset of flow instability and departure from nucleate boiling during irradiation experiment are investigated by subchannel analysis as indicators to verify the reactor core safe operation during the experiment. The results give the confidence that during this irradiation experiment, thermal-hydraulic steady state safety criteria of the mixed-core are satisfied and the fuel irradiation experiment does not induce any significant operational change.

1. Introduction

In order to make sure of safe operation of newly fabricated nuclear fuels when exposed to neutron flux in a reactor core, two types of examination, i.e., during irradiation examinations and post irradiation examinations (PIEs) are indispensable. Research reactors are valuable means to irradiate fuels to be prepared for PIEs as well as to investigate main phenomena affecting the fuel performance during irradiation, namely fuel thermal behavior, clad corrosion, pellet–cladding interaction and fission gas release. This irradiation experiment must not induce any significant operational change and any reduction of the reactor safety and it must comply with safety criteria of the research reactor. ATR (Advanced Test Reactor) (Marshal, 2005), HALDEN boiling water reactor (HBWR) (IAEA, 2009), BR2 (Belgian Reactor 2) (IAEA, 2009), JMTR (Japan Materials Testing Reactor) (Inaba et al., 2011), JHR (Jules Horowitz Reactor) (Bignan, 2011), HANARO (IAEA, 2007) are a few examples of research reactors which have been designed or refurbished to conduct irradiation experiments. Tehran Research Reactor (TRR) is a 5 MW pool-type reactor with about 1.0×10^{14}n/cm^2 sec maximum local thermal neutron flux at full power in an irradiation box in the center of the core (Mirvakili et al., 2012). It can be currently used as an appropriate tool for nuclear fuels and materials irradiation studies in Iran in the absence of any other research reactor with higher neutron flux. However, in addition to a comprehensive safety analysis, new facilities for online monitoring of fuel, clad and coolant temperatures, online measurement of fuel dimension changes and clad integrity check to avoid leakage must be installed in the reactor to ensure safety of the irradiation program. The main objective of this study is thermal-hydraulic safety analysis of irradiating a newly fabricated fuel in the TRR core as one of the reactor fuel elements to study the behavior of this fuel during irradiation at nuclear reactor core.

TRR core lattice is a 9×6 array containing Standard Fuel Elements (SFEs), Control Fuel Elements (CFEs) which host safety and regulating absorber rods, irradiation boxes and graphite boxes as reflectors (AEOI, 2009). Main characteristics of TRR core and its fuel elements are presented in Table 1. Fuel elements of TRR are plate-type U_3O_8–Al with 20% enrichment while the test fuel is composed of natural UO_2 ceramic pellets which are stacked in zircaloy tubing with 1.363 cm diameter to an active length of 0.615 m. In this study, the test fuel is going to be treated as one of the TRR fuels. Thus, it will be cooled by the same cooling mechanism uses for heat removal from the TRR fuels. Pool water with total flow rate of 500 m^3/hr passes through fuel elements and transfers generated heat to the heat exchanger. The test fuel assembly contains 18 fuel rods and one central tube. Main characteristics of test fuel assembly are presented in Table 2.

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http://dx.doi.org/10.1016/j.pnucene.2014.10.009
0149-1970/

Table 1
Main characteristics of TRR core (AEOI, 2009).

Parameter	Value
Thermal power	5 MW
Fuel	20% enriched U_3O_8–Al fuel with Aluminum cladding
Number of plates per fuel element	19 for SFE 14 for CFE
Fuel elements dimensions	SFE: 8.01 × 7.71 × 89.7 cm CFE: 8.01 × 7.71 × 161.5 cm
Moderator and Coolant	Light Water
Primary Coolant Flow rate	500 m^3/h
Coolant inlet temperature in 5 MW	37.8 °C
Coolant outlet temperature in 5 MW	46 °C
Fuel Plate Thickness	0.15 cm
Fuel Meat Thickness	0.07 cm
Cladding Thickness	0.04 cm
Water Channel Thickness	0.27 cm
Fuel Meat Width	6 cm
Active height of the fuel plate	61.5 cm
Safety rods absorber	Ag: 80% In: 15% Cd: 5%
Regulating rod absorber	AISI-316L Stainless Steel

The difference between test fuel and TRR fuels in geometry and enrichment necessitates comprehensive neutronic, thermal hydraulic and transient investigations to ensure safe operation of this mixed-core during experiment. In this paper, steady state thermal-hydraulic aspect of this investigation is of concern. The aforementioned difference causes non-uniformity in the layout of the coolant channels throughout the core. Thus, the calculation of the new distribution of the coolant flow and then thermal analysis based on power distribution throughout the core must be conducted. Several phenomena must be considered when a thermal-hydraulic analysis of a core is conducted, namely, departure from nucleate boiling ratio (DNBR), onset of nucleate boiling (ONB), onset of flow instability (OFI) and fuel and clad maximum temperature. Among them, maximum permissible fuel temperature, DNBR and OFI are so critical that if they occur, partial destruction of the core will occur. However, ONB is undesirable from reactor hydrodynamics points of view. In the case of clad maximum temperature, it depends on whether the maximum temperature is defined to avoid corrosion or clad melt-down. The former is considered as undesirable while the latter is critical.

2. Methodology

In order to evaluate thermal-hydraulic characteristics of TRR fuels and the test fuel during irradiation experiment, a comprehensive thermal-hydraulic analysis is conducted to obtain maximum temperature of fuel, clad and coolant, ONB, OFI and DNBR during irradiation. The core configuration which is chosen for this irradiation experiment is shown in Fig. 1.

Table 2
Main characteristics of test fuel assembly.

Parameter	Value
Fuel:	
Material	UO_2 (natural)
Diameter	1.148×10^{-2}m
Height	0.615 m
Clad:	
Material	Zr+1% Nb
Outer Diameter	1.363×10^{-2}m
Inner Diameter	1.178×10^{-2}m
Height	0.713 m
Fuel rod pitch	0.016 m
Central tube:	
Material	Zr+2.5% Nb
Diameter	0.015 m

Thermal-hydraulic parameters of plate-type fuel elements of TRR, i.e., SFEs and CFEs are calculated applying CAUDVAP 3.60 (Abbate, 2003a; Doval, 1998) and TERMIC 4.1 (Abbate, 2003b; Doval, 1998) while, thermal-hydraulic subchannel analysis of the rod-type fuel assembly is conducted using COBRA-EN code (Basile et al., 1999).

CAUDVAP code can calculate the flow distribution through the core, the total pressure drop across the core, components of pressure drop in each channel and the coolant velocities along the different sections of each channel. Coolant velocity and flow distributions are essential to obtain the range of velocity required in the TERMIC code input. TERMIC is a steady-state thermal-hydraulic code intended to calculate maximum allowable powers and heat fluxes using selectable limiting criteria of ONB, critical heat flux (CHF) and flow instability as a function of the coolant velocity in plate type fuel assemblies.

In COBRA-EN, the thermal-hydraulic analysis is carried out in an array of parallel channels delimited by cylindrical fuel rods and open gaps. Both core analysis and subchannel analysis can be conducted using this code. In subchannel analysis, COBRA-EN concerns the detailed description of individual fuel rod bundles. To model the heat transfer between fuel rods and coolant, COBRA-EN considers a full boiling curve comprising five heat-transfer regimes, i.e., single-phase forced convection, subcooled nucleate boiling, saturated nucleate boiling, transition and film boiling (or post-CHF boiling) (Basile et al., 1999).

In order to compensate the COBRA-EN inability to calculate ONB and DNBR under the research reactor core condition, two complementary computer programs are written in MATLAB to make use of COBRA-EN thermal output data to calculate ONB and DNBR safety factors. In this study, thermal-hydraulic analysis is conducted under two conditions, i.e., irradiating rod-type fuel assembly in position A2 of the core, and irradiating it at the center of the core in position D6. Irradiating test fuel at the center of the core (i.e., position D6) is desired due to the fact that maximum thermal neutron flux occurs in D6 and therefore, higher burn-up within a shorter time can be achieved by irradiating fuel in that position. This fact is of utmost importance in evaluating the behavior of test fuel during irradiation at high burn-ups. However, in order to mitigate consequences of any probable failure in the test fuel, irradiating it in a position with lower thermal neutron flux (i.e., position A2) is also of concern.

2.1. Flow distribution calculation

As mentioned before, the presence of the test fuel in the TRR core leads to non-uniformity in the coolant flow channels throughout the core. In order to calculate the steady state coolant flow and velocity distribution throughout this mixed core, CAUDVAP code is used. Given the coolant flow rate and configuration of the reactor core, the flow distribution through all channels of the core and the coolant velocity along different axial parts of each core channel is calculated. This velocity will be used as input to the TERMIC code in order to calculate the safety margins to critical phenomena in SFEs and CFEs.

2.2. Thermal-hydraulic calculations

2.2.1. Thermal-hydraulic analysis of TRR fuels

Given a coolant velocity, TERMIC performs thermal-hydraulic analysis to calculate the safety margins to ONB, DNBR, OFI and maximum permissible temperature of clad for the SFE and CFE with maximum heat flux under forced coolant flow in TRR core. 105 °C is