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Safety requirement assessment of fuel sample test in Tehran research reactor

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ABSTRACT

In the present study, neutronic and thermo-hydraulic evaluation and the feasibility study of inserting a fissile material into the Tehran research reactor core are performed. The feasibility study is carried out by calculating safety related neutronic parameters of the core, i.e., excess reactivity, shutdown margin, reactivity worth of the test fuel, safety reactivity factor and power peaking factors, during irradiation experiment using MCNPX nuclear code. In addition, in order to estimate the temperatures of fuel, clad and coolant water, a detailed three dimensional heat transfer analysis is carried out using computational fluid dynamic (CFD). The simulation is performed with two different fuel enrichments: 1.2% and 1.5%. The results indicate that the maximum fuel enrichment percentage should not be more than 1.5% to keep the clad temperature below the nucleate boiling point. It is also found that insertion of this fuel rod sample in the reactor core has not substantial effect on the safety criteria of the other plate-type fuel assemblies.

1. Introduction

Research reactors are valuable means to conduct the fuel irradiation programs to be prepared for post irradiation examinations as well as for investigating main phenomena affecting the fuel performance during irradiation, namely: fuel thermal behavior, clad corrosion, pellet cladding interaction and fission gas release. Hence, as stated in the IAEA document, it should be shown that any experiment conducted in the reactor will not violate safety limits and safety requirement of regulatory body. The operation of experimental fuel must not induce any significant operational change and any reduction of the reactor safety. The irradiation experiment must comply with safety criteria of the research reactor, enabling the reactor to operate properly with its maximum availability (Safaei et al., 2015).

As reported in (IAEA, 2008), the safety limits are necessary to protect the integrity of the principal physical barrier that guards against uncontrolled radioactive releases in all operational states and design basis accidents. The safety limits should be established by means of a conservative approach that ensures that all the uncertainties of the safety analyses are taken into account. This implies that the exceeding of a single safety limit does not always lead to unacceptable consequences. Nevertheless, if any safety limit is exceeded, the reactor should be shut down and normal operation should be restored only after an appropriate evaluation has been performed and approval for restarting has been given in accordance with established procedures (IAEA, 2008).

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). The aforementioned irradiation box can be currently used as an appropriate tool for nuclear fuels and materials irradiation studies in Iran. According to potential capability of TRR for irradiation purpose, the main objective of the present study is to assess the safety requirement of fuel sample insertion test. So, a fuel sample with two different enrichments (1.2%, 1.5% enriched UO_2) are adopted to analyze the safety parameters of new fabricated fuels from the neutronic and thermal-hydraulic point of views. As can be seen in the next section, fuel elements of TRR are plate type U_3O_8–Al with 20% enrichment while the sample fuel rods are UO_2 ceramic pellets with 1.2% and 1.5% enrichments.

The difference between fuel rods and TRR fuels in geometry and also in enrichment necessitates detail neutronic and thermal hydraulic and transient investigations to ensure safe operation of this mixed-core during experiment. However, in the present investigation, the emphasis is on the steady sate behavior.

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2. Description of TRR core

TRR is a 5 MW pool-type reactor which can be used for research, training and gaining expertise in nuclear technology. The 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). Table 1 displays the main specifications of the TRR.

In TRR fork-type control rods, i.e. four shim safety rods and one regulating rod are responsible for reactivity safe control in the core. The cross-sectional view of LEU Standard Fuel Elements (SFE) and CFE are given in Fig. 1a and b (AEOI, 2009). Other details of LEU and HEU fuel assemblies and core parameters are given in TRR Safety Analysis Reports (SAR) (AEOI, 2009).

Fuel elements of TRR are plate-type U_3O_8–Al with 20% enrichment. 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 as the one uses for the TRR fuels in which, pool water with total flow rate of 500 m3/hr passes through fuel elements and transfers their heat to the heat exchanger. Due to the fact that TRR fuel elements and the test fuel differ in geometry and enrichment, a comprehensive investigation of both neutronic and thermal hydraulic aspects is indispensable to ensure safe operation of this mixed core.

3. Methodology

3.1. Validation of simulation methodology

In order to validate the simulation methodology, the first core of TRR is simulated. The MCNPX code (LANL, 2008) is also used to calculate the reactor neutronic parameters. This core contains 14 SFE, 5 CFE and water as reflector. The core configuration and specification are given in the reference documents (Zaker, 2003). The benchmark refers to the first configuration of TRR that allows reactor operation at maximum power level of 5 MW.

In order to have better comparison, three core states, i.e. Cold and Clean, Hot Zero Power (HZP) and Hot Full Power (HFP) are considered. In the cold and clean state, the core is at room temperature and there is no xenon concentration in the fuel. In HZP and HFP states, the core is at operating temperature. However, in the HFP state, the xenon concentration is in equilibrium at full power, i.e., 5 MW. For HZP and HFP, a MCNP library is produced for the operating temperature in TRR core using “makxsf” (Brown, 2006) auxiliary processing code. According to SAR, the average temperatures of the fuel and coolant at 5 MW power level for which new cross section library is constructed, are considered to be 65 °C and 43 °C, respectively.

Table 1
Main specifications of TRR.

Type	Open pool, water
Designer	AMF, USA
Power	5 MW
Original fuel	High enriched uranium, 93.5% (U–AL) alloy
Present fuel	Low enriched uranium, 20.0% (U_3O_8) alloy
Geometry of fuel	Plate type
Number of SFE	14
Number of CFE	5
Number of plates per SFE	19
Number of plates per CFE	14
Reflector	Water-graphite
Cooling system	Forced flow
Coolant, Moderator	Light Water
Primary flow	500 m^3/h
Secondary flow	512 m^3/h
Control rods Type	SS: Shim Safety Rod RR: Regulating Rod
Absorber material	Ag (80%)-In (15%)-Cd (5%)

The reactivity at different modes for the first operating core are calculated and compared with the value of SAR parameters in Table 2. The SAR parameters are calculated using MTR-PC package (INVAP, 2006). The MTR_PC package has been developed by INVAP S.E in order to perform neutronic, thermal hydraulic and shielding calculations of MTR-type reactor for personal computers (PC). Comparing the results presented in Table 2, shows that, the average percentage error with respect to SAR parameters is below 5% except for states that CFEs are inserted. In addition, Comparison between cold & clean state and HZP state shows that constructing a cross-section library for the operating temperature of the core will have an effect of about 189 pcm on the results.

As listed in Table 3, the other important neutronic parameters for the first operating core are calculated and compared with the values of SAR and parameters obtained by (Lashkari et al., 2012). The most important safety parameters in the plate-type reactors are the shutdown margin, the margin to onset of nucleate boiling (ONB), and the margin to onset of excursive flow instability.

It is worth noting that, the safety reactivity factor (SRF = control rod worth divided by core excess reactivity) for the TRR takes the value of 3.24 which is greater than 1.5 and satisfies the IAEA requirement. In this case, the radial power peaking factors of all assemblies are presented in Fig. 2. The maximum radial power peaking factor (1.63) occurs in the center of the reactor. Furthermore, fast and thermal flux distributions are shown in Figs. 3 and 4, respectively.

4. Safety limits

Safety limits are limits on process variables within which the operation of the research reactor facility has been shown to be safe (IAEA, 2008).

In order to analyze the neutronic behavior of the TRR core during fuel irradiation experiment, detailed calculations are conducted to obtain core power and flux distribution, cycle length and operation characteristics, core excess reactivity, shutdown margin and power peaking factors. Selection of the safety limits is of paramount importance and should be given careful consideration (IAEA, 2008).

The number of limiting conditions for safe operation may be large, even for a low power research reactor. One of the most important parts in limiting conditions is reactivity and reactivity control systems. For instance, maximum excess reactivity, minimum shutdown margin, reactivity worth of the reactivity control mechanism (e.g. regulating, shim, safety, pulse rods or blades), reactivity addition rates, total reactivity worth of all experiments, maximum reactivity worth of specific types of experiment should be in acceptable ranges.

The design of an experiment or modification should be such as to minimize additional demands on the reactor shutdown system (IAEA, 2012).

If the experimental device or modified system, or its failure, could lead to an increase in the reactivity of the reactor, the experiment or modification should be designed so as to limit the positive reactivity effects to those that can safely be accommodated by the reactor control and shutdown systems (IAEA, 2012). In our case, the following neutronic safety criteria, which must be met in each core configuration (AEOI, 2009), are investigated:

➢ For any core configuration, the shutdown margin in absolute value must be greater than 3000 pcm