ASSESSMENT OF SAFETY ASPECTS OF PERFORMING FUEL IRRADIATION EXPERIMENTS IN TRR CORE

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ABSTRACT

Fabrication of domestic fuels for operating and under construction nuclear reactors in Iran necessitates evaluating thermo-mechanical behavior of these fuels during irradiation in the reactor core and also after reaching desired burn-up level. Currently, Tehran Research Reactor (TRR) with appropriate neutron flux is the sole operating research reactor in the country which can be used for aforementioned evaluations on newly-fabricated fuels. In this study, an instrumented mini fuel assembly is designed and fabricated for possibility evaluation of irradiating rod-type fuels in TRR core from safety point of view and also to provide online data from fuel and coolant during reactor operation. Neutronic and thermal hydraulic analysis of the resultant mixed-core during fuel irradiation experiments reveals that considering appropriate core power and irradiation position, all safety parameters of the core remain within safe limits with appropriate margins. Thus, TRR can potentially be used for fuel irradiation experiments.

1. Introduction

Due to the necessity of evaluating thermo-mechanical behavior of domestic fuels during irradiation in a reactor core and also after reaching desired burn-up level, we tried to investigate the possibility of making use of Tehran Research Reactor (TRR) as a test reactor and also to provide requirements to collect online data from the fuel under test. Currently, TRR with average thermal neutron flux of about 3.1×10^{13} n/cm^2.sec [1] at 5 MW power is the sole operating research reactor in the country which can be used for fuel irradiation tests.

In this study, an instrumented rod-type mini fuel assembly containing 17 mini fuel rods and 2 guide channels was designed to be irradiated in TRR core as a test fuel assembly. Fig. 1 shows a schematic of this test fuel assembly which contains fuel rods with 3% enriched UO_2 pellets within Zr+1%Nb cladding of 9.1 mm outside diameter. This mini fuel assembly was instrumented with 10 type-K thermocouples; 5 thermocouples to measure axial distribution of the cladding temperature of the central fuel rod and 5 thermocouples to provide the profile of coolant temperature within the assembly.

TRR core is a lattice of 9×6 array containing Standard Fuel Elements (SFEs), Control Fuel Elements (CFEs), irradiation boxes and graphite boxes as reflectors. General information about TRR core and fuels are provided in Table 1. Fig. 2 shows the core configuration in which experiments of present study were performed. Differences between test fuel assembly and TRR fuels in geometry and enrichment result in a mixed-core, for which a comprehensive safety analysis considering both neutronic and thermal-hydraulic aspects is essential to ensure that safety parameters of TRR mixed-core will be within safe limits during fuel irradiation experiments.

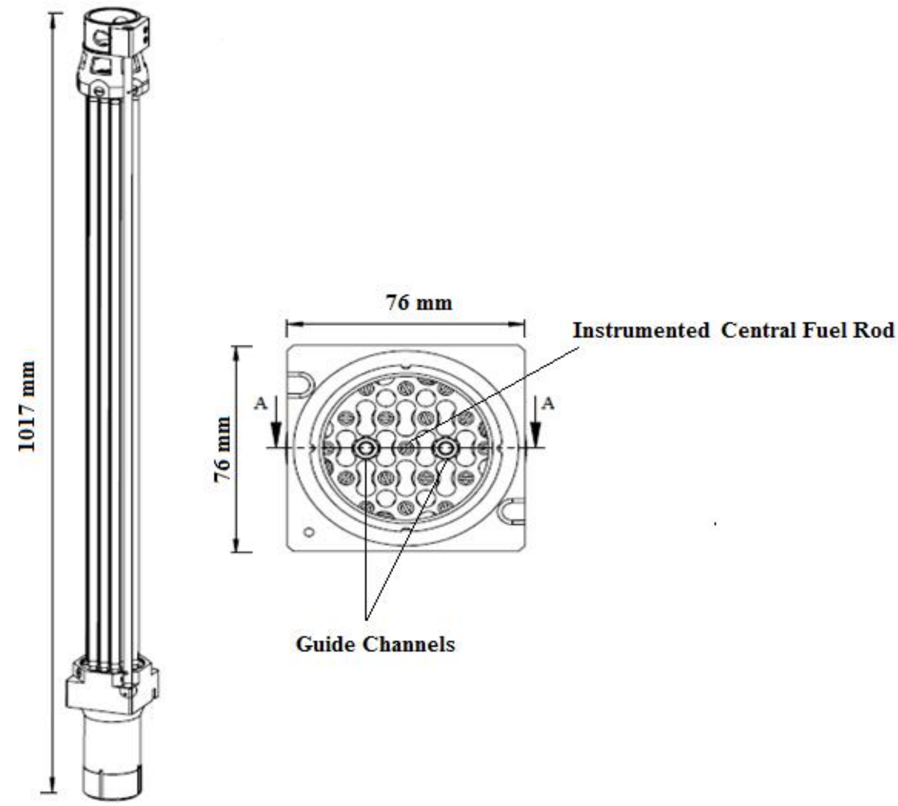

Fig 1. Instrumented rod-type mini fuel assembly

Tab 1: General information about TRR core and fuels [1]

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 ^oC
Coolant outlet temperature in 5 MW	46 ^oC
Fuel Plate Thickness	0.15 cm
Water Channel Thickness	0.27 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

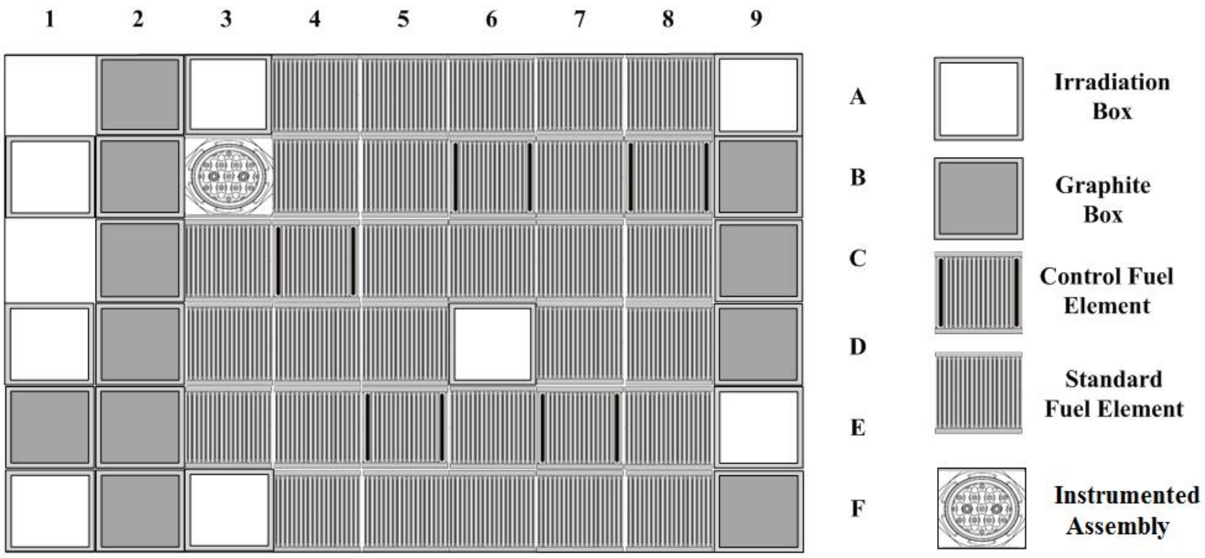

Fig 2. TRR mixed-core configuration in fuel irradiation experiment

2. Methodology

2.1 Neutronic Analysis

In the first step of a comprehensive safety analysis, neutronic analysis was conducted to ensure that excess reactivity, shutdown margin, safety reactivity factor and power peaking factors of the mixed-core are within the safe limits presented in FSAR (Final Safety Analysis Report) [1] and OLCs (Operational Limits and Conditions) [2] of the TRR. In order to perform analysis under real condition, burnup of all TRR fuel assemblies in the core were calculated using MCNPX [3] considering the period of time they have been irradiated in previous cores and the time period of reactor shutdowns between operating cycles. Also, average temperatures of fuel and coolant during operation were used to produce required cross-section libraries for the analysis.

In addition to safety margins, neutronic analysis can provide heat flux of each fuel element in the core which is a necessary parameter to calculate thermal-hydraulic safety margins to critical phenomena under operating condition.

Validity of the neutronic model was investigated by two methods. First, benchmarking the calculated data for the first core configuration of TRR against the data reported in the reactor FSAR for three core states, i.e. cold and clean, hot zero power and hot full power and second, benchmarking the calculated axial distribution of thermal neutron flux of an equilibrium core against experimental data [4].

2.2 Thermal-hydraulic Analysis

In order to ensure that there is enough cooling for TRR fuels and the rod-type fuel under test to prevent onset of nucleate boiling (ONB), onset of flow instability (OFI) and departure from nucleate boiling (DNB), the margins to these critical phenomena were calculated. As a prerequisite to calculate those margins, coolant flow distribution throughout the mixed-core was calculated. Presence of a rod-type fuel assembly in a core containing plate-type fuels causes non-uniformity in the coolant flow pattern throughout the core. Coolant flow rate and velocity through each channel of the mixed-core were calculated using CAUDVAP code [5]. Thereafter, due to different geometries of test fuel and the TRR fuels, various computational tools were applied to calculate thermal-hydraulic safety parameters.

2.2.1 Thermal-hydraulic analysis of TRR fuels

Calculated velocity in previous step was used as input to the TERMIC code [6] to calculate safety margins to OFI, ONB and DNB in plate-type SFEs and CFEs. Validity of the CAUDVAP and TERMIC models were investigated using data provided in FSAR for several core configurations [7].

2.2.2 Thermal-hydraulic analysis of rod-type mini fuel assembly

Thermal-hydraulic parameters of the rod-type test fuel assembly were calculated using COBRA-EN subchannel code [8]. In this regard, some computer programs were also written to calculate helium gap conductance, onset of nucleate boiling ratio (ONBR) and minimum departure from nucleate boiling ratio (MDNBR) for rod-type fuel based on applicable correlations under irradiation experiments condition. Fig 3 illustrates six groups of sub-channels in the test fuel assembly. Channels in each group have the same flow area, wetted and heated perimeters and hydraulic diameter as presented in Table 2.

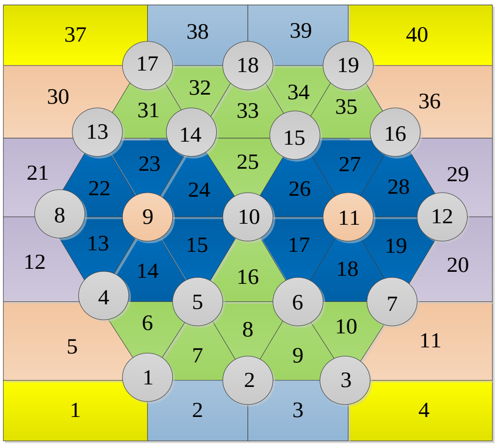

Fig 3. Numbering scheme of fuel rods and coolant sub-channels in the test fuel assembly

Tab 2: Parameters of sub-channels in test fuel assembly

Channel number	Flow area (m^2)	Wetted perimeter (m)	Heated perimeter (m)	Hydraulic Diameter (m)
1-4-37-40	4.40672E-04	2.36131E-02	7.14712E-03	7.46486E-02
5-11-30-36	2.38700E-04	2.53362E-02	1.42942E-02	3.76851E-02
12-20-21-29	1.68307E-04	2.53362E-02	1.42942E-02	2.65717E-02
2-3-38-39	1.77422E-04	1.42942E-02	1.42942E-02	4.96485E-02
6-7-8-9-10-16-25-31-32-33-34-35	3.78733E-05	1.42942E-02	1.42942E-02	1.05982E-02
13-14-15-17-18-19-22-23-24-26-27-28	3.78733E-05	1.42942E-02	9.52950E-03	1.05982E-02

2.3 Experiment

As the first step to provide requirements to collect online data from the instrumented test fuel assembly, an HMI screen was designed and implemented in data acquisition system of TRR to be able to monitor the measured parameters of fuel and coolant during experiments. The first experiment after providing the requirements to collect and record data under irradiation was evaluating the response of instrumented assembly to step change of reactor power. The experiment was performed considering the results of safety analysis to ensure safe operation of the mixed-core during experiment. Starting from 1MW power, data from instrumented assembly were recorded at each power level for about 10 minutes to reach steady state and then, power was increased to a higher level. Thereafter, the measured values were compared with the calculated values of clad and coolant temperatures under the same conditions.

3. Results and discussion

Neutronic safety parameters of the mixed-core while irradiating the rod-type fuel assembly at the core position B3 were calculated and the results were compared with acceptable limits in Table 3. As can be seen from these data, all neutronic safety parameters are within safe limits with significant margins.

This must be mentioned that, choosing an appropriate position to irradiate the instrumented rod-type fuel assembly in the core is dependent on two items; firstly, safety parameters of the mixed-core during irradiation experiment must remain within permissible limits and secondly, loading the instrumented assembly in the desired core position must be feasible for the TRR operators regarding all instruments and cables associated with the test fuel assembly.

Tab 3: Results of neutronic safety parameters of the mixed-core during irradiation of instrumented rod-type fuel assembly

Core Safety Parameter	***Value***	***Accepted limit***
Excess reactivity (pcm)	5786.59	-
Shutdown margin (pcm)	5405.18	> 3000
Shutdown margin in stuck rod condition (pcm)	1474.42	> 500
Integral Worth of SSRs (pcm)	11191.76	-
Safety Reactivity Factor	1.93	> 1.5
Reactivity worth of regulating rod (pcm)	398.44	< β_{eff}
Reactivity worth of test fuel assembly(pcm)	549.06	-
Maximum power peaking factor	1.95	< 3
Effective delayed neutron fraction (β_{eff})	794.22	-

As a prerequisite to calculate thermal hydraulic safety parameters of the rod-type test fuel assembly, hot SFE and hot CFE in the mixed-core, coolant flow distribution in each channel of the core was calculated and the results are presented in Table 4. As expected, the majority of the coolant flows through the SFEs because of the larger number of SFEs compared to other elements of the core.

Tab 4: Coolant flow rate and mass fraction through each channel of the mixed-core

Channel type	***Total flow mass fraction***	***Flow rate (m^3/hr)***
SFE	0.7886	394.3
CFE	0.1102	55.1
Rod-type fuel assembly	0.0166	8.3
Irradiation box	0.0023	1.1
Between assemblies	0.0823	41.0

Calculated coolant velocities in each element of the core were used as inputs to calculate thermal hydraulic safety parameters of the hot SFE and CFE using TERMIC code. In order to conduct a conservative thermal analysis, the total power peaking factor of the core was assumed to be 3. Thus, maximum heat flux of 33.87 W/cm^2 was considered in thermal analysis.

As demonstrated in Table 5, all thermal hydraulic safety parameters of hot SFE and CFE of the mixed-core during irradiation experiments are within the acceptable limits. In this Table, melting point of aluminum in U_3O_8-Al fuel plates and the temperature at which the corrosion of aluminum clad occurs are defined as the maximum permissible limits for the fuel and clad temperatures in SFEs and CFEs. The parameter “margin to Twall = 105 °C” is the ratio of heat flux when wall temperature reaches 105 °C to the maximum heat flux in the hot channel. Considering the parameters of coolant channel and flow and also the range of applicability of correlations, appropriate correlations to calculate margins to ONB, MDNB and OFI were used. Safety criteria for margins to aforementioned critical phenomena were defined based on reactor FSAR.

Tab 5: Thermal-hydraulic safety parameters of hot CFE and SFE in the mixed-core during irradiation of instrumented fuel assembly

Parameters	***Value***		***SAFETY CRITERIA***
	CFE	***SFE***	
Maximum fuel temperature (°C)	117.18	116.08	< 650
Maximum clad surface temperature (°C)	97.88	96.78	< 105
Maximum coolant temperature (°C)	63.54	62.94	< 116
Margin to ONB (Bergles-Rohsenow)	1.46	1.49	>1.3
Margin to MDNBR (Sudo-Mishima correlation)	12.29	12.31	>2
Margin to OFI (Whittle & Forgan correlation)	2.55	2.61	>2
Margin to Twall=105 °C	1.44	1.46	>1

Considering the calculated coolant flow rate and linear heat rate of each fuel rod in test assembly, thermal hydraulic safety parameters of the hot rod of rod-type test fuel assembly were calculated using sub-channel analysis. Gap conductance was calculated to be

$2806.1\,\frac{\mathrm{w}}{\mathrm{m^2.K}}$ and used as input in sub channel analysis. The results of sub-channel analysis are summarized in Table 6. In this Table, melting point of UO_2 and zirconium are defined as the maximum permissible limits for the fuel and clad temperatures in the rod-type fuel. In addition, saturation temperature of light water under 0.17 MPa pressure in the core is considered as the limit for coolant temperature.

Tab 6: Thermal-hydraulic safety parameters of the hot rod of instrumented assembly

Parameters	***Value***	***SAFETY CRITERIA***
Maximum fuel temperature (°C)	313.45	< 2850
Maximum clad surface temperature (°C)	101.21	< 1837
Maximum coolant temperature (°C)	47.1	< 116
Margin to ONB	1.34	> 1.3
Margin to MDNBR (Bernath correlation)	7.20	> 2

According to neutronic calculation, mini-rod #12 was the hot rod in test fuel assembly when this assembly was irradiated in core position B3. The results presented in Table 6 are thermal-hydraulic safety parameters of this mini rod under irradiation at 2.7MW core power. This power is the maximum power at which test fuel assembly can be irradiated in position B3 without violating any thermal-hydraulic safety criterion. Maximum coolant temperature reported in Table 6 is the temperature of hottest sub-channel around hot rod.

This must be mentioned that, even if the test fuel assembly is irradiated in core position B3 but at 5MW core power, all thermal-hydraulic safety parameters of SFEs and CFEs remain within safe limits. Under this condition, MDNBR as the most critical parameter of the test fuel assembly will be 3.7 that is within acceptable limit with significant margin but ONBR in the test fuel assembly violates safety criteria. However, ONB is undesirable from reactor hydrodynamics points of view and will not result in partial destruction of the core.

After ensuring the safe operation of the mixed-core, an experiment was conducted to obtain online data during fuel irradiation. Data obtained from the instrumented assembly in response to step change of reactor power are presented in Figs 4 and 5. Tables 7 and 8 demonstrate a detailed comparison between calculated and measured temperatures in the test assembly in response to step change of reactor power. In the case of clad temperature, the maximum average difference between calculated and measured values was 2.11% and in the case of coolant temperature, there was less than 0.7% difference between calculated and measured coolant temperatures.

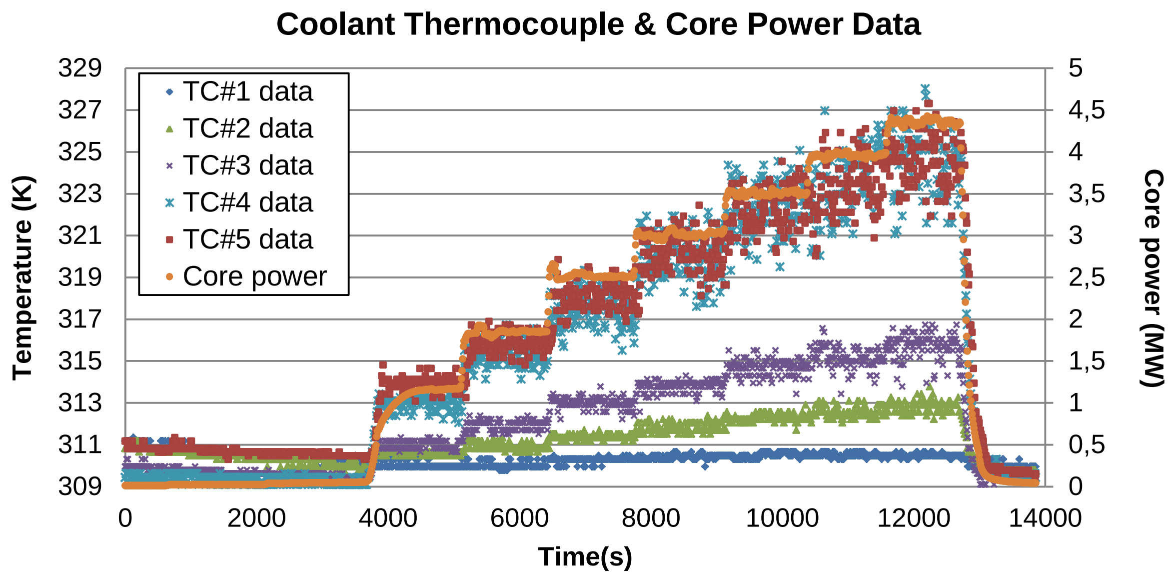

Fig 4. Coolant temperature measured by 5 thermocouples along the test assembly in response to step change of reactor power

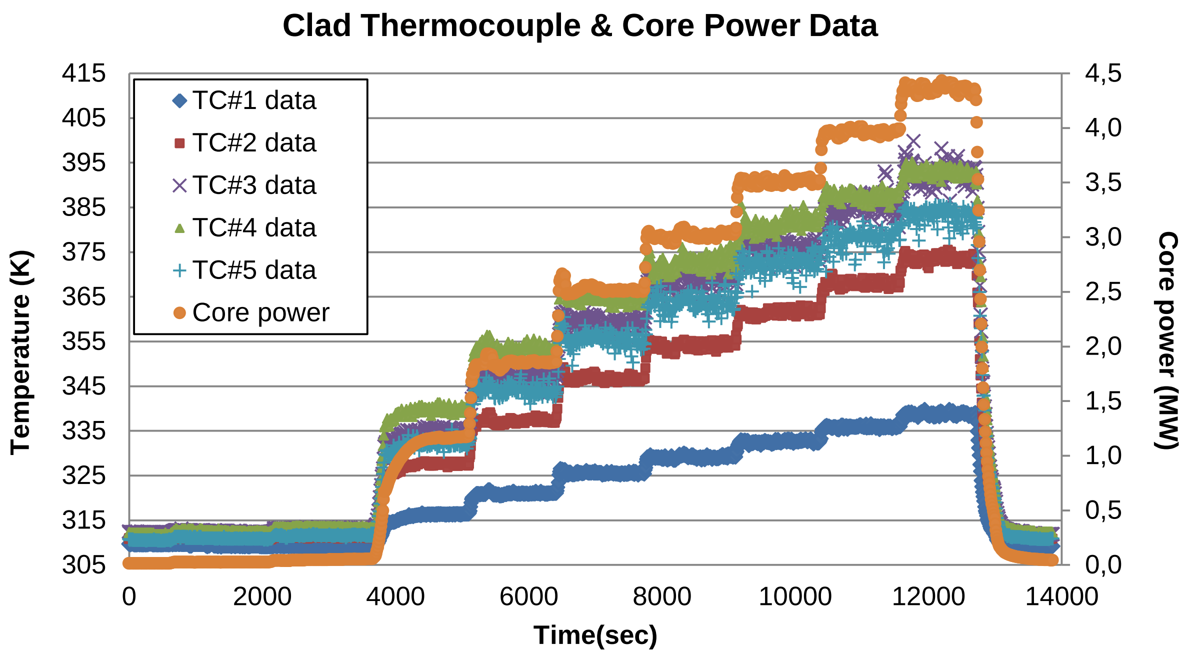

Fig 5. Clad temperature measured by 5 thermocouples along the instrumented rod of the test assembly in response to step change of reactor power

Table 7. Comparison between calculated and measured clad temperatures in 5 points along the instrumented rod of test assembly (where thermocouples are installed) in response to step change of reactor power

Power (MW)	TC#1		TC#2		TC#3		TC#4		TC#5	
	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.
1.0	315.7	316.3	324.0	327.6	329.8	334.9	332.7	339.8	328.6	332.3
2.0	319.9	320.9	332.6	337.2	341.3	346.8	345.4	353.4	339.3	344.1
2.5	324.3	325.5	341.2	346.6	352.6	359.2	357.9	364.6	349.9	355.3
3.0	327.3	329.0	347.2	353.9	360.4	368.6	366.5	372.4	357.2	363.6
3.5	330.4	332.5	353.1	361.4	368.1	375.5	374.9	381.2	364.3	372.1
4.0	333.4	335.8	358.9	367.9	375.5	385.3	383.0	387.5	371.2	378.0
4.3	335.5	338.8	363.0	373.5	380.8	392.7	388.7	393.0	376.0	382.9
Average Difference	0.53%		1.92%		2.11%		1.68%		1.64%	

Table 8. Comparison between calculated and measured coolant temperatures in 5 axial levels along the test assembly (where thermocouples are installed) in response to step change of reactor power

Power (MW)	TC#1		TC#2		TC#3		TC#4		TC#5	
	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.	Calc.	Exp.
1.0	310.02	310.01	310.95	310.56	312.11	311.02	313.55	313.92	314.68	313.05
2.0	310.03	310.03	311.50	310.91	313.31	311.90	315.57	315.82	317.33	315.27
2.5	310.04	310.25	312.07	311.36	314.55	313.02	317.66	318.00	320.09	317.61
3.0	310.04	310.40	312.48	311.81	315.46	313.78	319.18	320.16	322.09	320.00
3.5	310.05	310.48	312.89	312.28	316.36	314.69	320.69	322.20	324.09	322.24
4.0	310.06	310.50	313.31	312.54	317.26	315.20	322.20	323.32	326.10	323.55
4.3	310.06	310.48	313.60	312.80	317.91	315.73	323.30	324.60	327.55	325.07
Average Difference	0.09%		0.21%		0.53%		0.26%		0.68%	

4. Conclusion

In this study, a comprehensive neutronic and thermal hydraulic safety analysis was conducted on the feasibility of irradiating a rod-type instrumented fuel assembly in Tehran research reactor core which normally contains 20% enriched plate-type fuels. Comparing the results with permissible limits reported in reactor FSAR and OLCs shows that all neutronic safety criteria of the mixed-core, i.e., shutdown margin, core excess reactivity, safety reactivity factor and power peaking factor were satisfied. In addition, considering appropriate core power and irradiation position in the core, thermal-hydraulic parameters of both TRR plate-type fuels and the rod-type test assembly, i.e., maximum temperature of fuel, clad and coolant, ONBR, OFI and MDNBR were calculated to be within acceptable limits. The results prove that TRR can potentially be used for fuel irradiation experiments. In addition, preliminary experiment performed during irradiation of instrumented test fuel assembly shows the capability of TRR data acquisition system to collect and record online data from fuel and coolant during irradiation. However, the instrumentation needs to be improved in future studies to get more information from the fuel during irradiation in the core.

5. References

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