# A PRAGMATIC APPROACH TOWARDS DESIGNING A SECOND SHUTDOWN SYSTEM FOR TEHRAN RESEARCH REACTOR

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One second shutdown system is proposed for the Tehran Research Reactor to achieve the goal of higher safety in compliance with current operational requirements and regulations and improve the overall reliability of the reactor shutdown system. The proposed second shutdown system is a diverse, independent shutdown system compared to the existing rod based one that intends to achieve and maintain sub-criticality condition with an enough shutdown margin in many of abnormal situations. It is designed as much as practical based on neutron absorber solution injection into the existing core while the changes and interferences with the existing core structure are kept to a minimum. Core neutronic calculations were performed using MCNPX 2.6.0 and MTR\_PC package for the current operational core equipped with the second shutdown system, and one experiment was conducted in the Tehran Research Reactor to test the neutronic calculations. A good agreement was seen between theoretical results and experimental ones. In addition, capability of the second shutdown system in the case of occurrence of design basis accident in the Tehran Research Reactor is demonstrated using PARET program.

Key words: Tehran research reactor, second shutdown system, safety, validating experiment, MCNPX, MTR\_PC

# INTRODUCTION

The shutdown system is a key element for a nuclear reactor, which directly affects its safe operation. The aim of the shutdown system is to quickly shut down the reactor by rapid insertion of sufficient negative reactivity into the core to bring the reactor subcritical and maintain it subcritical for all core configurations under all operational states and design basis accident conditions.

According to the IAEA criteria, the obligation exists for power reactors to have one additional shutdown system [1], and for research reactors that the probability of happening an accident is conceivable, IAEA recommends the use of two shutdown systems fully independent from each other to guarantee that in the event of failure in one of them the second will proceed to shut down the reactor [2, 3]. In particular, in light of the lessons learned from the Fukushima-Daiichi accident, IAEA currently recommends enhancing preventive and mitigation measures for research reactors through installation of new earthquake resistant equipment, including installation of a diverse shutdown system such as one based on the injection of neutron absorber solution [4].

While insertion of the control rods through gravity is used for any type of nuclear reactor as first shutdown system, depending on design specifications of research or power reactors, different types of the second shutdown system (SSS) are designed and used. For example, neutron absorbing boric acid injection into primary water in PWR is used as an SSS [5]. In heavy water reactors in which heavy water is used as a coolant and moderator, injection of gadolinium nitrate solution into heavy water is used for SSS [6]. In fast reactors using liquid metal as a coolant, there is no liquid absorber that can be dissolved in sufficient quantity in the liquid metal, and even if such an absorber would exist, purification of coolant would be very expensive. Hence for inserting negative reactivity into the core a number of approaches are suggested such as using spheres made of a neutron absorbing material, tantalum, that are hydraulically suspended by the upward flow of the sodium coolant, liquid lithium neutron poison in tubes, etc. [7].

In the case of research reactors, if the reactor reflector is heavy water containing within a reflector vessel, the SSS consists in partially emptying the reflector vessel by dumping the reflector heavy water into a storage tank beneath the core. In the absence of reflector material the neutron losses are high enough to

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drive the core to a sub-critical condition and producing the required shutdown [6]. Among the open pool research reactors using light water for cooling and moderating and graphite or beryllium as a reflector, 22 MW Egyptian Second Testing Research Reactor (ETRR-2) has an SSS consisting in the release of a gadolinium nitrate solution from storage tanks into chambers located within the core chimney and the strong neutron absorption of the solution injected guarantees immediate reactor shutdown [8].

Tehran Research Reactor (TRR) is a 5 MW research reactor that for the past four decades has served as one of major research and isotope production facilities in Iran. Following domestic fabrication of nuclear fuels in the country in recent years, the TRR has been also used for fuel irradiation experiments. Hence according to the IAEA categorization upon the hazard risk, this reactor is in category 1 of 4 existing categories [9].

In this respect, design of a diverse and independent SSS for the TRR is motivated, aiming at improving the reliability of the overall shutdown function. But due to the limited remaining operating life envisaged for TRR as an old reactor, designing an SSS with minimum interferences and changes to the existing core structure is preferred. Very recently, a study was undertaken to compare possible methods for SSS which could be used in TRR [10]. Among them, use of heavy water as reflector tank around the reactor core was selected as the optimum system for TRR, but the study was limited to the first operating core of TRR and the selected method requires major changes to the existing core structure and grid plate that in the present work as a requirement we avoid it as much as possible.

In this paper, at first the TRR and its existing First Shutdown System are introduced. We next give the requirements of an SSS for the TRR, followed by a short overview of the possible plans to this problem. The paper finally presents an appropriate design of SSS that can be implemented in the TRR. Accident analysis is done and the performance of proposed SSS is studied using PARET program. The design is validated from the neutronic aspects by both simulation and experiment.

# DESCRIPTION OF TRR AND ITS EXISTING SHUTDOWN SYSTEM

The TRR is a 5 MW pool type research reactor with MTR type fuel cooled and moderated by light water, and is reflected by graphite. This reactor has one  $9 \times 6$  grid plate for insertion of core elements such as SFE, IR-BOX or GR-BOX into it. Its first shutdown system consists of 4 Shim Safety Rods together with 1 Regulating Rod. The absorber rods are by weight composed of 80 % silver, 15 % indium, and 5 % cadmium and regulating rod material is made of Stainless Steel. The FSS is activated manually or automatically and shuts the reactor down when a specified threshold value is exceeded. The reactor protection system can receive the trip signal of different parameters such as neutron flux, core temperature, failed fuel, core flow, area radiation monitoring, *etc.* [11].

# SECOND SHUTDOWN SYSTEM SELECTION

# **Design requirements**

We consider the following requirements for an SSS in the TRR:

- it must have the capability to make reactor subcritical in acceptable time, with enough shutdown margin and keep the reactor in this state for required time; minimum shutdown margin for the SSS must be 1000 pcm<sup>1</sup> [6, 12, 13],
- it must use of diverse hardware, including signal sensors and associated electronic, and fundamentally different modes of operation to reduce the potential for common mode failures,
- it must not have a significant negative effect on overall neutronic characteristics of core such as local neutron flux, neutron flux distribution, excess reactivity and corresponding operating fuel cycle length,
- it must not have a significant negative effect on reactor utilization, specifically accessible places for irradiating samples, and
- it must not have major changes to the core structure, its cooling system and grid plate, and must be easily serviced and maintained.

# Conceivable designs and final proposed SSS

Given that the surrounding space of TRR core has been filled with beam tubes, thermal column, and rabbit system and in compliance with the requirement that not to make major changes to the core structure in this way, we must narrow conceivable options for an SSS in the TRR down to those which are based on injection of neutron absorber solution into the core. In the following we weigh up a number of these designs.

- Locating one or two vertical, horizontal or cross empty thin bar in the middle of core that is filled with moderator in normal operation and will be filled with neutron absorber solution in emergency condition to make the reactor subcritical. The need to alter the core grid plate is the most important disadvantage of this design.
- Designing and locating one empty container in the position D6 which will be filled with one neutron

 $<sup>11 \</sup>text{ pcm} = 10^{-5}$ 

absorber solution in emergency condition to make reactor subcritical. Depending on current usage of D6 position, this container could have different geometries.

Allocating the most effective space in the core for irradiation to the SSS, and interfering needed facilities for injecting absorber with existing structure located above the core center are of the main drawbacks to this design.

 Designing some empty spaces in two selected rows of the core, row 2 and row 9, leading to smaller width for GR- boxes placed in these rows. These empty spaces will be filled with neutron absorber in case of an accident to make the reactor subcritical.

In this work the number 3 of proposed designs is selected as the SSS for TRR and shown to meet the requirements.

This design as shown in fig. 1 is composed of 8 empty right rectangular boxes so that their main characteristics can be seen in tab. 1.

As can be seen in fig. 1, the only variation in reactor core being equipped with the selected SSS is altering 8 boxes of rows 2 and 9 and locating the SSS boxes in these two rows.



Table 1.	Main	characteristics	of SSS	hoves
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SSS components	Characteristics	
Material of walls	Al-6061	
Wall thickness [mm]	2	
Height [mm]	703	
Length [mm]	320	
Width [mm]	30	

This system will be filled with nitrogen in normal operation at approximately 10 kPa pressure. The liquid absorber which will be used is aqueous enriched boric acid with 99 % <sup>10</sup>B. Boric acid is a white crystal with density of 1.435 g/cm<sup>3</sup> which is soluble in water with maximum 5 % weight fraction at 25 °C. The density of the resulting liquid is 1.03 g/cm<sup>3</sup> [14].

# Hydraulic design of neutron absorber injection circuit

This system is composed of absorber storage, absorber injection circuits, pressure capsule, injection boxes into the core, valves, pressure gauges and pump. Storage tank has a capacity of approximately 20 L that will be located at a height of about 10 m above the core and can sustain up to 10 bar<sup>2</sup> pressure. If a specified threshold value is exceeded, the electrical panel will be opened to inject the neutron absorber solution under pressure into the SSS boxes and fill them in less than a few seconds. This is noticeable to mention that with actuation of SSS, this signal would be sent to the FSS which provides diversity for FSS trip signal. For monitoring system and fast detection of any leakage, the SSS has certain pressure in normal condition. For filling SSS boxes within 3 s, pipes with 1.91 cm (0.75 inches) in diameter and pressure of 4 bar in circuit are required. The schematic plan for proposed SSS is shown in fig. 2.



Figure 2. Schematic plan for proposed SSS

 $\frac{1}{2}$  1 bar = 10<sup>5</sup> Pa

#### CALCULATIONS AND RESULTS

Neutronic calculations were performed using MCNPX 2.6.0 code with continuous energy and ENDF/B-VII library [15] and MTR PC package [16]. The MTR PC is an integrated system composed of several of the codes used for research reactor design calculations. For the cell calculation, WIMSD-4 code [17] was used, which has its own nuclear data library with updates from ENDF/B-IV of Ag, In, Cd, and Gd, and with the collision probability option in one dimensional geometry (slab) and 69 neutron energy groups. POS-WIMS code was used between cell calculation and core calculation to make second homogenization and get library cross sections in core codes format. CITVAP code which is an improved version of the well known CITATION-II [18] code was used to perform core calculations in the x-y-z geometry and to follow up the loading of fuel in all loading cycles.

The accident analysis is carried out utilizing the PARET code. PARET is a digital computer program designed for use in predicting the course and consequences of non-destructive reactivity accidents in small reactor cores. It is basically a coupled neutronics-thermohydraulics-heat transfer code employing point kinetics, one dimensional hydrodynamics and one dimensional heat transfer. The PARET model consists of a water-cooled core represented by a maximum of four fuel elements and associated coolant channels [19].

#### **Reactivity of proposed SSS**

Effect of injection of neutron absorber solution on the reactivity of TRR core being equipped with the SSS boxes was examined using MCNPX 2.6.0 code to determine if this proposed system could meet the safety requirements. The decrease of core reactivity with increasing the injection of enriched boric acid solution into the SSS boxes is depicted in fig. 3. We see that the maximum negative reactivity of the SSS is nearly 2800 pcm that is enough to shutdown the operating reactor safely with required shutdown margin of 1000 pcm.



Figure 3. Core reactivity variation with amount of absorber in SSS boxes

It is remarkable that the maximum decrease in reactivity occurs after injecting about 6 L solution into the SSS boxes when the level of injected solution reaches the middle of the box where the neutron flux is peaking, as would be expected and can be seen from the SSS reactivity vs. injected solution in fig. 3.

# Impact of proposed SSS on core neutronic performance

Given that the elapsed time for complete injection of neutron absorber into SSS boxes is 3 s, we study the impact of SSS on the core neutronic performance in the following categories.

- The neutron flux distribution within the irradiation boxes D6, A9, and F3 with and without the SSS boxes in the equilibrium core no. 70 was calculated using MCNPX 2.6.0. D6 which is the most important irradiation box with respect to the neutron flux and A9 and F3 boxes are the nearest irradiation places to the SSS boxes.
- The obtained results for spatial (vertical) distributions of neutron flux density in three groups of thermal (0.0-0.625 eV), epithermal (0.626 eV to 0.821 MeV), and fast (0.822-10 MeV) are shown in fig. 4 to fig. 6, respectively, with less than 2 % uncertainty.

We see that the insertion of the SSS boxes into the core does not have any negative effect on neutron flux density distribution within the irradiation boxes as required.



Figure 4. Thermal neutron flux density distribution in irradiation positions D6, A9, and F3 with and without SSS boxes in the core



Figure 5. Epithermal neutron flux density distribution in irradiation positions D6, A3, and F3 with and without SSS boxes in the core



Figure 6. Fast neutron flux density distribution in irradiation positions D6, A3, and F3 with and without SSS boxes in the core

 Table 2. Core calculated reactivity before and after locating SSS

State	Situations		Reactivity [pcm]	
1	Equilibrium core without SSS boxes	2227	53	
2	Equilibrium core after inserting SSS boxes	1548	69	

Locating the SSS boxes in the core leads to a decrease of core excess reactivity estimated to be almost 700 pcm as calculated results are given in tab. 2. In order to reduce this effect one may add an SFE instead of GR-Box or IR-Box in the core.

Fuel and moderator temperature reactivity coefficients for the burned up core without and with the SSS boxes calculated by MTR-PC code, given as the slope of reactivity *vs*. fuel and moderator temperature, are shown in figs. 7 and 8, respectively. We see that the fuel temperature reactivity coefficient remains nearly constant but the moderator temperature reactivity coefficient increases for the core with SSS boxes.

The core kinetics parameters for the core without and with SSS boxes calculated by MTR-PC system are given in tab. 3. It is noticeable that both the effective delayed neutron fraction and prompt neutron gen-



Figure 7. Fuel temperature reactivity coefficients without and with SSS boxes



Figure 8. Moderator temperature reactivity coefficients without and with SSS boxes

Table 3. Effect of SSS	boxes on core	kinetic parameters
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State	Λ (μs)	$\beta_{\rm eff}$ (pcm)
With SSS	41.12	754
Without SSS	39.66	748

eration time increase slightly for the core with SSS boxes.

# Capability of SSS in reactivity insertion accident

In this section we analyse a credible accident caused by uncontrolled reactivity insertion into the core being equipped with the SSS. In this respect, we postulate a positive reactivity of  $1.5 \,$ <sup>3</sup> inserted within 0.7 s into the core while the reactor is initially critical at startup state with power level of 1 mW.

This can be associated with accidental drop of a lateral CFE on the core when control rods are withdrawn which inserts a reactivity of about 1.5 \$ into the core, and from the data on fuel element drop test, time required by the fuel element to reach the grid plate has been calculated to be 0.7 s.

The resulting transients of the reactor power for the case without scram but with and without SSS boxes in the core are shown in fig. 9. It is seen that the inherent safety feature of core remains nearly the same for the core with and without SSS boxes.

Time variation of reactivity after reactivity insertion of 1.50 \$ in 0.7 s into the core without scram and with the SSS scram is shown in fig. 10, demonstrating the function of SSS scram in case of reactivity



Figure 9. Evolution of reactor power after reactivity insertion of 1.50 \$ in 0.7 s with and without SSS boxes



Figure 10. Evolution of reactivity without scram and with SSS scram after reactivity insertion of  $1.50 \pm in 0.7 s$ 

<sup>&</sup>lt;sup>3</sup> 1\$ is equal to  $(\Delta k/k)\beta_{eff}$ 

insertion accident. It is worth mentioning since in PARET code trips are triggered by reactor overpower from determined set-point, in case of FSS failure the SSS is actuated when reactor power exceeds 110 % of set point (more than 5.5 MW).

The time evolutions of peak clad temperature for the cases without scram and scram with the SSS are compared in fig. 11, demonstrating the capability of SSS for controlling clad temperature in case of reactivity insertion accident while the FSS fails.

Assuming the failure of FSS, the capability of SSS to shut down the reactor for different reactivity insertions is shown in fig. 12.

The sharp decrease in reactivity which is seen in fig. 12 is due to high negative reactivity feedback of fuel temperature in the reactor.

As can be seen form fig. 12, for reactivity insertions in the core up to 2.0 \$ in 0.7 s, the state of reactor changes to subcritical state with almost 1000 pcm shutdown margin.

Most of credible reactivity insertion accidents that may occur in the TRR, including inadvertent removal of irradiation sample, beam tube flooding, and fall of lateral control fuel element or graphite on the core have a reactivity insertion less than 2.0 \$ in 0.7 s [11] and thus the SSS has capability to shut down the reactor for these abnormal conditions if the FSS fails.



Figure 11. Evolution of peak clad temperature after reactivity insertion of 1.50 \$ in 0.7 s with no scram and scram by SSS



Figure 12. Evolution of reactivity for different insertion accidents with scram by SSS while FSS fails

#### **BENCHMARK EXPERIMENT IN TRR**

An experiment was conducted in the TRR to validate the SSS reactivity calculated using neutronic codes with measurements. It is worth mentioning that this experiment was performed while reactor was in the very low power of 100 W with negligible coolant motion accordingly. Prior to the experiment the following cases should have been considered.

### Xenon effect

The reactor was operated at a thermal power of 4 MW for about 144 h before the shutdown. The experiment was started around 34 h after the reactor shutdown and lasted for approximately 4 h. Because of xenon build-up in the core, we need to take into account the reactivity of xenon accumulated in the core at the start of the experiment.

Letting *I* and *X* denote the atomic densities of iodine and xenon, respectively, we consider the following coupled rate equations describing xenon concentration transient [20]:

$$\frac{\mathrm{d}I}{\mathrm{d}t} \quad \gamma_{\mathrm{I}} \quad {}_{\mathrm{f}}\phi_{\mathrm{T}} \quad \lambda_{\mathrm{I}}I \tag{1}$$

$$\frac{\mathrm{d}X}{\mathrm{d}t} \quad \lambda_{\mathrm{I}}I \quad \gamma_{\mathrm{X}} \quad {}_{\mathrm{f}}\phi_{\mathrm{T}} \quad \lambda_{\mathrm{X}}X \quad \sigma_{\mathrm{aX}}\phi_{\mathrm{T}}X \qquad (2)$$

where  $\lambda_1$  and  $\lambda_X$  are the decay constants for iodine and xenon, respectively,  $\gamma_1$  and  $\gamma_X$  are the effective yields of these isotopes, while  $_f$  and  $\phi_T$  are the thermal fission cross-section and average thermal neutron flux density, respectively.

After simultaneously solving the above equations, Xe concentration is obtained following the reactor start-up at t = 0 h and then shutdown at t = 144 h as shown in fig. 13.

Considering the core xenon concentrations at the shutdown time and experiment time (the continuous and dotted black lines, respectively in fig. 13) and taking into account that the xenon reactivity for the shutdown time corresponding to the saturation concentration of xenon has been calculated to be nearly 2800 pcm [11], we obtain the reactivity of xenon at the experiment time to be approximately 1600 pcm.



Figure 13. Evolution of xenon concentration in core

# Leakage test for SSS boxes

To make sure that the SSS boxes were manufactured as non-leaking, the leak test of boxes was done using the air gas gauge pressure at 1 bar, together with soap solution for more assurance.

#### **Boric acid production**

Boric acid is a white crystalline powder and its solubility in water depends on water temperature. For producing natural boric acid solution, we dissolved 350 g of boric acid powder in 7 L of distilled water.

### Calibration of the control rod SR2

The procedure was followed to measure the total reactivity of control rod SR2 needed for the experiment. First, to diminish the rods shadowing effect, the control rod SR4 located at the farthest distance from the measured rod SR2 was considered as compensating rod. Then the rod SR2 was fully inserted and the rod SR4 was fully withdrawn, while the remaining two safety rods (SR1 and SR3) together with the regulating rod (RR) were kept properly inserted to make the reactor critical and maintain the power at a certain level. The insertion position of these three control rods was kept constant during the whole experimental procedure. The reactivity of rod SR2 was measured by performing successive withdrawals from 100 % insertion up to 0 % and recording the doubling time of reactor power at each step. After each withdrawal of the control rod SR2, the rod SR4 was properly inserted to balance the reactivity. These steps were repeated until the rod SR2 was fully withdrawn. At each step, the reactivity change was determined using the inhour equation. The total rod reactivity was finally obtained by summing the reactivity changes for all successive steps. The reactivity of control rod SR2 using this method was found to be approximately 1800 pcm that with taking into account 400 pcm contribution each safety rod from the sum of 1600 pcm for the reactivity of xenon poison we ended up with the 2200 pcm for the total reactivity of rod SR2.

In tab. 4, the calculated results for the reactivity of SR2 are compared with the experimental ones.

Table 4. Comparison of calculated values for reactivity of control rod SR2 with experiment

	Description	Calculation method	Reactivi	Relative	
State			Calculation	Experiment	error (to experiment)
1	SR2 reactivity	MTR_PC	1635	1780	-0.08
2	SR2 reactivity	MCNPX 2.6.0	1693 50	1780	-0.05

### **Experiment description and results**

An absorber storage of SSS, which is seen in fig. 15 was made and put in the anticipated location within the core configuration arranged for this experiment as shown in fig. 14.

	Thermal column					
a		Water	Water	Water	Water	
3	GR-BOX		Second shu	ıtdown box		GR-DOX
8	SFE	RR	SFE	SFE	SFE	SFE
7	SFE	SFE	SFE	SFE	SR2	SFE
6	SFE	SR1	SFE	IR-BOX	SFE	SFE
5	SFE	SFE	SFE	SFE	SR3	SFE
4	SFE	SFE	SR4	SFE	SFE	SFE
3	IR-BOX	SFE	SFE	SFE	SFE	IR-BOX
2	GR-BOX	Water	SFE	SFE	GR-BOX	GR-BOX
1	GR-BOX	GR-BOX	GR-BOX	GR-BOX	GR-BOX	GR-BOX
	A	В	С	D	Е	F

Figure 14. TRR core configuration arranged for conducted experiment



Figure 15. Experimental sample of absorber storage

The exact characteristics of absorber storage used in the experiment are brought in tab. 5.

The experiment consists of first getting one criticality point before injecting neutron absorber solution

Parameter	Value
Wall material	Al-6061
Wall thickness [mm]	2
Wall height [mm]	730
Wall length [mm]	29.5
Storage width [mm]	30
Total volume of box [L]	6

 Table 5. Characteristics of SSS box used in experiment

into SSS box, then determining four criticality points after successively pouring the neutron absorber solution into the SSS box, and finally getting one critical point after finishing the injection experiment and taking out the SSS from the core. The results of this experiment are compared with the calculated values as shown in tab. 6. The written numbers for SR and RR are the extraction percent from the core and the uncertainty of results calculated with MCNPX 2.6.0 is less than 50 pcm.

The calculated values in tab. 6 obtained using the MCNP and MTR-PC codes are in fact the reactivity of equivalent to deviation from the critical point reached in the experiment in each case, showing a good agreement between experiment and calculation.

### CONCLUSIONS

In this paper the motivation for considering a diverse SSS in the TRR was explained. Given the limited remaining operating life envisaged for TRR, among the possible approaches for implementation of SSS a simple but effective approach to designing an SSS based on injection of neutron absorber into the existing core was taken. Neutronic calculations were made using Monte Carlo code MCNPX 2.6.0 and deterministic MTR\_PC package for the selected SSS in core of TRR, and an experiment was conducted in the TRR to test the theoretical results. A good agreement was seen between calculated results and experimental ones.

To see the capability of the SSS during the design basis accidents a credible scenario in the TRR, based

on positive reactivity insertion was analyzed using the PARET code.

The obtained results show that the proposed SSS designed with minimum changes to the existing core structure has enough shutdown capability with minimal impact on the core neutronic performance. Furthermore, the SSS has the ability to shut down the reactor with only one out of two boxes filled with neutron absorber that is of importance from the safety point of view. It turns out that the selected SSS is a pragmatic design compatible with an aged reactor like TRR to ensure the reactor shutdown in case that the FSS fails.

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#### **AUTHORS' CONTRIBUTIONS**

All calculations using MCNPX, MTR\_PC, and also experiment were performed by E. Boustani. Overall guidance about the work and also check on the theoretical and experimental procedures were done by S. Khakshournia and H. Khalafi.

### ACRONYMS

FSS	<ul> <li>first shutdown system</li> </ul>
GR-BOX	<ul> <li>graphite box</li> </ul>
IAEA	<ul> <li>International Atomic Energy Agency</li> </ul>
IR-BOX	<ul> <li>irradiation box</li> </ul>
MTR	<ul> <li>material testing reactor</li> </ul>
OPAL	- Open Pool Australian Light Water Reactor
RR	<ul> <li>regulating rod</li> </ul>
PWR	<ul> <li>pressurized water reactor</li> </ul>
SR	<ul> <li>shim safety rod</li> </ul>
SFE	<ul> <li>standard fuel element</li> </ul>

#### Table 6. Comparison between calculated results for differential reactivity of SSS with experiment

		Results			
	State	Critical point (extraction percent)	Reactivity difference [pcm]		
		Experiment	MTR_PC	MCNPX 2.6.0	
1	First criticality point	SR1, 3 = 80, SR4 = 100, RR = 55.7 SR2 = 0, inserted reactivity = 0	321	299	
2	Criticality point, after pouring 2 L of boric acid	SR1, 3 = 80, SR4 = 100, RR = 55.7, SR2 = 22.5, inserted reactivity = 306 pcm	-66	-43	
3	Criticality point, after pouring 3 L of boric acid	SR1, 3 = 80, SR4 = 100, RR = 55.7, SR2 = 32.3, inserted reactivity = 548 pcm	-45	-61	
4	Criticality point, after pouring 4 L of boric acid	SR1, 3 = 80, SR4 = 100, RR = 55.7, SR2 = 35.2, inserted reactivity = 633 pcm	-155	-53	
5	Criticality point, after pouring 6 L of boric acid	SR1, 3 = 80, SR4 = 100, RR = 55.7, SR2 = 36.2, inserted reactivity = 665 pcm	-231	-209	
6	End point, finishing experiment, core without SSS box	SR1,3 = 80, SR4 = 100, RR = 55.7, SR2 = 8.5, inserted reactivity = 0	120	186	

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# Ехсан БУСТАНИ, Самад КАКШУРНИЈА, Хосеин КАЛАФИ

# ПРАКТИЧАН ПРИСТУП ПРОЈЕКТОВАЊУ ДРУГОГ СИСТЕМА ЗА ЗАУСТАВЉАЊЕ ТЕХЕРАНСКОГ ИСТРАЖИВАЧКОГ РЕАКТОРА

Да би се постигла већа сигурност у складу са актуелним оперативним захтевима и прописима и побољшала укупна поузданост реакторског система за заустављање, предложен је и други систем за заустављање Техеранског истраживачког реактора. Предложен је различит и независан систем за заустављање у односу на постојећи заснован на сигурносној шипци, чија је намера да постигне и одржи поткритично стање са довољном маргином за заустављање у многим ненормалним околностима. Пројектован је на принципу убризгавања раствора неутронског апсорбера у постојеће језгро, при чему су промене и мењање постојеће структуре језгра сведене на минимум. Неутронски прорачуни језгра обављени су коришћењем MCNPX 2.6.0 и MTR\_PC пакета програма за текућа оперативна језгра са системом за заустављање, а спроведен је и један експеримент на реактору ради тестирања неутронског прорачуна. Уочено је добро слагање теоријских и експерименталних резултата. Поред тога, способност другог система за заустављање у случају појаве пројектованих акцидената на Техеранском истраживачком реактору проверена је коришћењем PARET програма.

Кључне речи: Техерански исшраживаки реакшор, други сисшем за заусшављање, сигурносш, йошврђивање експерименша, MCNPX, MTR\_PC