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## Reduction of radiation exposure probability at Tehran research reactor equipped with a second shutdown system

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### HIGHLIGHTS

- One independent and diverse shutdown system is designed for the Tehran Research Reactor.
- Design of Second Shutdown System (SSS) is based on the injection of a liquid neutron absorber.
- Design calculations are performed using the MCNPX code, PARET code and the Pipe Flow Expert software.
- ORIGEN2 and HotSpot codes are also used for simulation of environmental pollution release.
- After equipping the TRR with the SSS, it is forbidden from significant release around the reactor.

### ABSTRACT

A second shutdown system (SSS) is designed for the Tehran Research Reactor (TRR) completely independent and diverse from the existing First Shutdown System (FSS). Given limitations, specifications, and requirements of the reactor, the design of SSS is based on the injection of liquid neutron absorber. The plan has the ability to satisfy the major criterion of required negative reactivity worth, to transfer the reactor to subcritical state in needed time, with necessary shutdown margin and for the required duration. Design calculations are performed using the stochastic code MCNPX2.6.0, deterministic code PARET and Pipe Flow Expert software. The ORIGEN2 code and HotSpot health physics code are also used for simulation of environmental pollution release. The SSS chambers cause a decrease of about 5% and 15% in total and thermal neutron flux, respectively. To demonstrate the SSS role in enhancing reactor safety, the probable accident of core meltdown is investigated. As a consequence of this accident, the radioactive pollution in and out of reactor containment is released. Without existing the SSS and in case of failure of FSS, the residents within 58000 m<sup>2</sup> of the reactor perimeter would receive about 1 mSv which is more than the annual limit of absorbed dose for the community.

### KEYWORDS

Tehran research reactor  
Safety  
Second shutdown system  
MCNPX code  
ORIGEN2 code  
PARET code  
HotSpot code

### HISTORY

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## 1 Introduction

The nuclear accidents such as Chernobyl and Fukushima have global effects. Challenges of considering the safety issues have given rise to mass pressure on national and international responsible organizations to regarding the highest level of nuclear safety. All attempts in research reactors must be done for improving safety in accordance with the latest safety standards. Continuous upgrade, modification and providing Engineered Safety Features (ESFs) such as Second Shutdown System (SSS) must be considered for improving nuclear safety. SSS is used in different types of reactors such as light and heavy water power reactors, fast breeder reactor, pool type reactors, low power reactors

and finally open pool research reactors (Boffie et al., 2012; Bond et al., 2005; Böning and Blombach, 1995; Boustani et al., 2016; Kim, 2006; Vanmaercke et al., 2012).

The safety issue is of prime importance for Tehran Research Reactor (TRR) regarding that this reactor is situated in a residential region, having a continuous operation, and is an aged and the only usable reactor for producing some radioisotopes in the country.

In this paper two probable scenarios in which the core meltdown is possible are studied to illustrate the importance of SSS in enhancing the reactor safety. These scenarios are reactivity insertion and safety flapper opening when the First Shutdown System (FSS) is not actuated properly for shutting down the reactor. It is assumed that

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the emergency ventilation system isn't available for having the worst case scenario in pollution calculations.

A brief review of TRR is given in section 2. In section 3 first a concise description of MCNPX code, PARET code, Pipe flow expert software, ORIGEN2 code, and HotSpot health physics code is presented. Then SSS characteristics and performance are given in section 3.1., and accidents sequences and related issues in pollution release are studied in section 3.2. Section 4 is devoted to relevant calculations for SSS design, characteristics and pollution release, and finally the conclusions are brought in section 5.

## 2 Description of TRR

TRR is situated in a residential region of Tehran as shown in Fig. 1. It is a medium power research reactor with a nominal power of 5 MW used for research, training, irradiation activities and etc. The FSS of this reactor is its only shutdown system with the ability for transferring the reactor to a subcritical condition in less than 1 s. The FSS has 4 Shim Safety Sods (SSRs) for bringing the reactor to the critical state, increasing and decreasing power and one Regulating Rod (RR) for fine flux and power regulations. The proposed SSS is based on the liquid neutron absorber injection for transferring the reactor the subcritical condition. The reactor has one main ventilation system equipped with High-Efficiency Air Particulate (HEPA) filters and one emergency ventilation system equipped with charcoal filters (AEOI, 2009b).

As could be seen in Fig. 1, this reactor which was out of urban in establishment time in 1946, nowadays is in the residential region of Tehran. Upon the documents, the probability of severe core damage from any accident must be less than  $10^{-4}$  per year and fail of shutdown system must be less than  $10^{-2}$  per demand (ARPANSA, 2001; IAEA, 1999). This probability decreases noticeably after equipping the reactor with this feature, indicating the safety enhancement due to the SSS (Boustani and Khakshournia, 2017).



Figure 1: Situation of TRR in Tehran city.

## 3 Calculations

Neutronic calculations are done using the MCNPX2.6.0 code. Having a continuous library of cross sections for numerous isotopes, the ability to accurately simulate problem geometry, and following a large number of particles are some of its characteristics (Waters et al., 2002). The PARET code is a part of MTR\_PC package which is used for investigation of reactivity insertion following FSS fail (INVAP, 2006). The hydraulic calculations are done using Pipe Flow Expert software belonged to Daxesoft Ltd. This software is for problems analysis and solution in which the pressure drop in pipe network should be determined. This software has the capability of drawing the complicated piping and analysis of related processes (Daxesoft-Ltd, 2010).

### 3.1 Second Shutdown System

The available plans for SSS could be divided into two categories based on used absorber material:

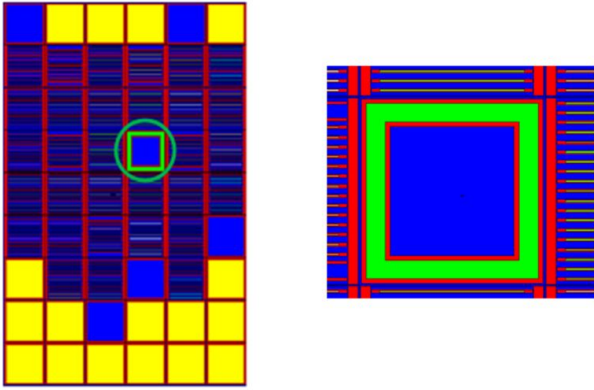
1. Plans don't include liquid neutron absorber: The study on a neutron absorber rod as a control rod in one Miniature Neutron Source Reactor (MNSR) or an absorber string for one low power research reactor in the Imperial college are samples in this category (Boffie et al., 2012; Bond et al., 2005). Shortage of some basic design principles such as diversity is the main drawback to this design.
2. Plans based on liquid neutron absorber: the IAEA has encouraged responsible organizations to move towards the use of these type of plans in their research reactors (IAEA, 2009).

There are some limitations for using liquid neutron absorber in TRR; first, a considerable part of outer space of the TRR core has been occupied with beam tubes, rabbits, and thermal column, and so it is not feasible to use this space for SSS. Second, regarding the small volume of the core, there is a little free space into the core for this purpose. As a result, any plan for designing the SSS into the core needs a minimum amount of space, besides satisfying the design requirements.

Considering these limitations, the usage of an empty chamber in the most important irradiation position of the core, namely D6 as shown in Fig. 2, is proposed as the place of SSS chamber into the core. This chamber is empty in normal operation and filled with liquid neutron absorber in needed occurrences.

Boric acid and Gadolinium nitrate solutions are used as neutron absorber solutions. These two absorbers could be provided with 100% of B-10 and Gd-157, respectively (Dalle et al., 2013; Lintereur et al., 2012).

As could be seen from Fig. 2, the proposed design for SSS chamber is such that it does not cause a major limitation on the normal usage of the irradiation position. The final and main characteristic parameters of proposed SSS chamber are given in Table 1.



**Figure 2:** General view (left) and specific view of SSS chamber in TRR core.

**Table 1:** Main characteristic parameters of SSS chamber.

Quantity	Amount (%)
External dimensions of SSS chamber (cm)	$7.35 \times 6.95 \times 73.05$
Internal dimensions of SSS chamber (cm)	$5.55 \times 5.15 \times 72.65$
Wall thickness (mm)	2
SSS chamber volume (cm <sup>3</sup> )	137
Evacuation time (s)	1
Evacuation pressure (bar)	10
Normal pressure (kPa)	20
Valve diameter (in)	0.75

In the TRR, two conceivable accidents leading to core damage are falling of Control Fuel Elements (CFEs) and spurious opening of flapper valve following a failure in FSS. Having equipped the reactor with SSS, in the event of falling one CFE into the core with 1.5 \$ worth in 0.7 s, the performance of SSS causes to prevent from core meltdown as could be seen from Fig. 3.

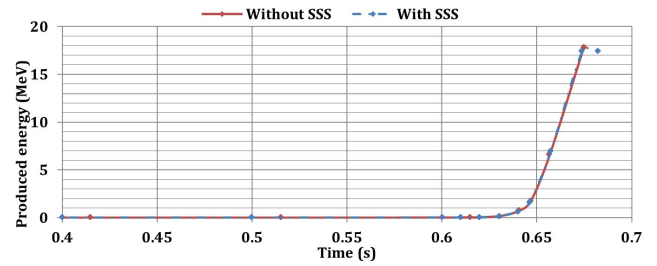
The produced energy in the hottest fuel element in TRR depending on the existence of SSS is shown in Fig. 3. According to this figure, the difference in produced energy depending on the actuation of SSS is 0.5 MJ which caused an increase in the fuel temperature beyond 650 °C and finally occurrence of the core meltdown.

$$\Delta E = (17.9 - 17.4) \text{ MJ} = 0.5 \text{ MJ} = mL_f \quad (1)$$

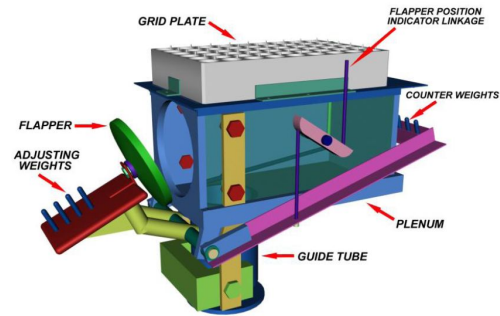
$$L_f = 395.4 \text{ J} \rightarrow m = 1.26 \text{ kg} \quad (2)$$

where  $L_f$  is the latent heat and  $m$  is the melted mass. Considering the 73.8 kg as the total weight of core, the percent of core meltdown due to this accident is about 1.27%.

The grid plate, plenum and safety flapper of TRR are shown in Fig. 4. As it is seen, the flapper valve is connected to one side of plenum and when it is regulated using the weights, it will open as soon as the flow is reduced from a predetermined value. This leads to opening one direction for coolant beside the core. As the flow through the core decreased to 20% of nominal value following spurious opening of flapper valve, failure of FSS could pessimistically cause 10% core meltdown.



**Figure 3:** Effect of the SSS performance on produced energy.



**Figure 4:** Grid plate, plenum and safety flapper in TRR.

### 3.2 Radioactive release

ORIGEN2 code is an upgrade version of ORIGEN code. This code is used for production and depletion calculations of isotopes. This code solves the growth and decay equations of numerous isotopes for each arbitrary coupling (Ludwig and Croff, 2002). The health physics HotSpot code performs a first-order approximation of radiation effects of radioactive release to the atmosphere. This program is designed for short-term releases, short region release in open regions, and simple meteorological condition (Homann, 2009).

For uranium fuel of 20% enrichment with U-235, each 1% of fuel is equivalent to 1617 MWd per ton uranium. For the studied TRR equilibrium core with 45.7 kg uranium and total mass of 73.8 kg, 1% burnup is equivalent to 74 MWd. This core has 28.26% burnup which is equivalent to 2091 MWd energy.

The radioactive material release from core to coolant and from the coolant to the containment is different for various radionuclides. Also, the amount of release from containment to environment highly depends on number, type and efficiency of filters used in the ventilation system. This study is conducted for unavailability of the emergency ventilation system considering the pessimistic condition.

The special attention is needed for radionuclides selection effective to environment pollution. For research reactors, the activation products and transuranium isotopes are often less than fission products. Ultimately, the radioactive release major hazard of core meltdown in research reactors is considered to be from fission and activation products (IAEA, 2008). To conduct the pollution survey in environment, the needed coefficients for radionuclides released from core to the coolant, then from the coolant to the containment, and finally from containment

**Table 2:** Radionuclides release fraction for TRR.

Isotope	Release fraction from		
	Core to coolant	Coolant to containment	Cont. to environment
Inert gases	1	1	1
Iodine	0.25	0.00109	1
Fission products	0.1	0.0001	0.015

to the environment are given in Table 2 (AEOI, 2009b; IAEA, 1991).

The equilibrium core inventory is computed by ORIGEN2 code includes hundreds of radionuclide. In this study, besides radionuclides determined specifically in documents, the radionuclides with a concentration higher than  $10^6$  Ci are considered. In this section, only 20 radionuclides with their release fractions are given in Table 3 as a sample.

The critical group for TRR could be a broad spectrum of people which are inhabitant in the outer region of reactor site, ranging from hundreds of meters to some kilometers. To fulfill the pessimistic condition, it is assumed that the accident occurs under the worst meteorological condition. Some of the main needed parameters for this case study are given in Table 4.

**Table 3:** Equilibrium core inventory.

No.	Isotope	Radioactivity (Ci)		
		Core	Coolant	Containment
1	I-131	1.46E+05	1.59E+00	1.59E+00
2	I-133	2.75E+05	3.00E+00	3.00E+00
3	I-135	2.53E+05	2.75E+00	2.75E+00
4	Xe-133	2.77E+05	2.77E+04	2.77E+04
5	Xe-135	2.42E+05	2.42E+04	2.42E+04
6	Kr-87	4.43E+04	4.43E+03	4.43E+03
7	Kr-88	6.26E+04	6.26E+03	6.26E+03
8	Kr-89	7.03E+04	7.03E+03	7.03E+03
9	La-140	2.27E+05	5.68E-01	8.53E-03
10	Ce-141	2.19E+05	5.47E-01	8.22E-03
11	Ce-143	1.87E+05	4.69E-01	7.04E-03
12	Ce-144	1.85E+05	4.63E-01	6.95E-03
13	Pr-144	1.85E+05	4.63E-01	6.96E-03
14	Sr-90	6.15E+03	1.54E-02	2.31E-04
15	Te-127	1.89E+04	4.72E-02	7.09E-04
16	Te-129	5.33E+04	1.33E-01	2.00E-03
17	Cs-134	1.54E+03	3.85E-03	5.79E-05
18	Cs-137	1.57E+04	3.91E-02	5.88E-04
19	Nd-147	8.39E+04	2.10E-01	3.15E-03
20	Pm-149	5.08E+04	1.27E-01	1.91E-03

**Table 4:** TRR main characteristics.

Quantity	Amount
Breathing rate ( $\text{m}^3/\text{s}$ )	3.33E-04
Receptor height (m)	1.5
Ambient wind velocity (m/s)	2
Stack height (m)	57
Exhaust gas velocity (m/s)	1.95
Exhaust gas temperature ( $^{\circ}\text{C}$ )	17.4
Peripheral temperature ( $^{\circ}\text{C}$ )	17.4
Stack diameter (m)	2.5

## 4 Results

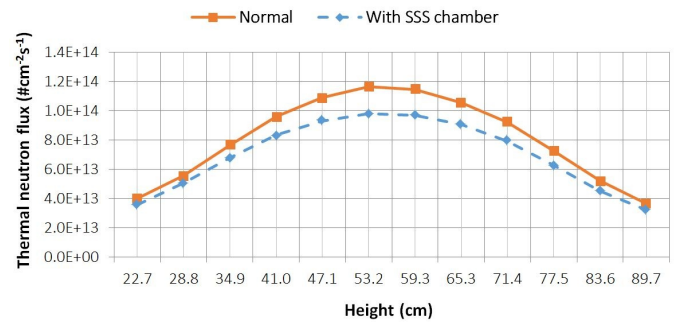
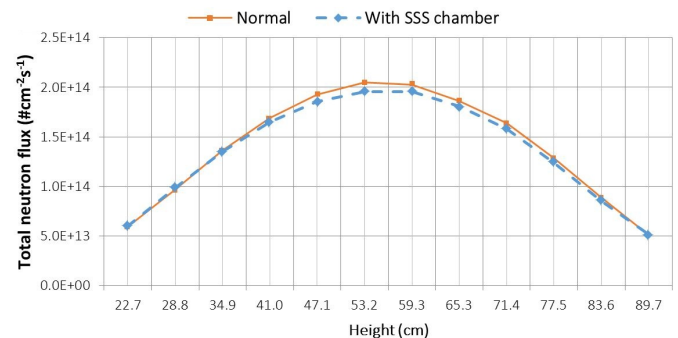
### 4.1 Neutronic calculations

The obtained results for proposed SSS considering equilibrium core No. 42 as a reference core are given in Table 5 (AEOI, 2009a).

These calculations are performed for the equilibrium core in which SSS chamber is located into the irradiation position D6 of the core.

It is found that the effect of SSS insertion into the reactor core is local, so that its influence on distant locations is negligible. The axial distributions of thermal neutron flux and total neutron flux for the most important irradiation position D6 in which this proposed SSS is located are calculated and shown in Figs. 5 and 6, respectively. The relative error for obtained results is less than 1% that couldn't be seen in the shown figures.

As could be seen from Figs. 5 and 6 for thermal and total neutron flux respectively, the insertion of SSS chamber into the reactor core leads to a decrease in thermal neutron flux up to 15%, although it does not have any remarkable effect on total neutron flux.

**Figure 5:** Thermal neutron flux in D6 position before and after insertion of SSS chamber.**Figure 6:** Total neutron flux in D6 position before and after insertion of SSS chamber.

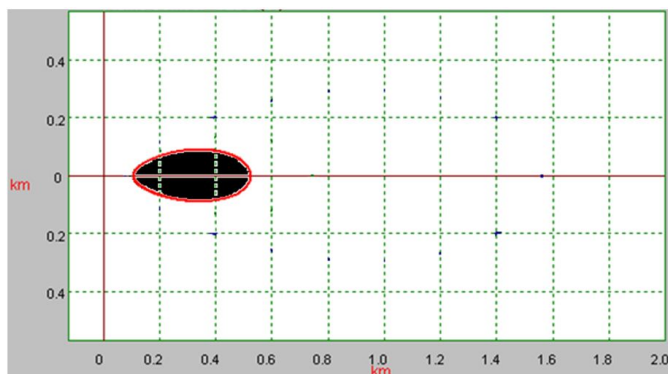
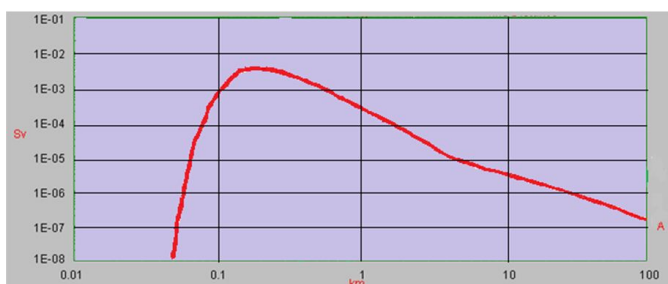
**Table 5:** Equilibrium core No. 42 reactivity.

Condition	Reactivity (pcm)
Equilibrium core No. 42	$1970 \pm 66$
SSS chamber is empty	$2558 \pm 66$
SSS chamber is filled with water	$2058 \pm 56$
SSS chamber is filled with gadolinium nitrate solution	$-2809 \pm 60$
SSS chamber is filled with boric acid solution	$-2526 \pm 56$

#### 4.2 Radiological calculations

Numerous parameters such as meteorological conditions, ground anomalies, and nature of radioactive sources have substantial effects on the individual absorbed dose. Total Effective Dose Equivalent (TEDE) is the sum of absorbed dose in the human body which stems from the internal and external factors. The TEDE for class A leading to the maximum of absorbed dose for people around the reactor is shown in Fig. 7.

The abscissa and ordinate are distances from the reactor core situated in the origin in km. Region with TEDE bigger than 1 mSv is shown with red ellipsoid filled with black colour. This accident causes the radioactive release yielding an absorbed dose over the permissible annual limit 1 mSv in people living in the area of about 58000 m<sup>2</sup> around the TRR. TEDE for atmospheric stability class A for the reactor core is given in Fig. 8.

**Figure 7:** TEDE for class A.**Figure 8:** TEDE for all atmospheric stability classes.

## 5 Conclusion

As a result of this study, the proposed SSS is capable of shutting down the reactor with enough shutdown margin and in the acceptable time. The insertion of this system into the reactor core caused a little negative effect of about 5% on the total neutron flux and a significant one of about 15% on the thermal neutron flux. This large negative effect of the SSS chamber on thermal neutron flux is due to the chamber width which could be decreased further because of considerable shutdown margin. The insertion of SSS chamber caused an increase in the reactivity in normal operation which is one of the major advantages of this plan.

The TEDE of TRR radioactive release in the accident condition resulting from partial core meltdown and unavailability of emergency ventilation system was calculated. This accident occurred following the existing FSS failure, and without an SSS for the reactor and gave rise to absorbed dose in people over 1 mSv in 58000 m<sup>2</sup> around the TRR region. The probability of radioactive release in and out of reactor containment could be decreased remarkably after equipping this reactor with SSS as a safety upgrading project.

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