Loss of Coolant Flow Accident Analysis for the Fluoride Salt Cooled High Temperature Reactor

Yao FU, Yang YANG, Yang ZOU, Qiang SUN and Jie ZHANG
Key Laboratory of Nuclear Radiation and Nuclear Energy Technology, Shanghai Institute of Applied Physics, Chinese Academy of Sciences
2019 Jia Luo Road, Jiading District, Shanghai 201800, P. R. China
fuyao@sinap.ac.cn

ABSTRACT

The Fluoride salt cooled High temperature Reactor (FHR) is an innovative reactor design that uses conventional TRISO high temperature fuel with a low-pressure liquid salt coolant. Design of this reactor is currently in progress in many countries especially in China and the United States. An FHR based on ordered pebble bed core is suggested by Shanghai Institute of Applied Physics (SINAP). Reactor coolant system (RCS) and reactor coolant pump (RCP) design is one of the most important safety problems. Reactor coolant pump coasting down is chosen as the main parameters to analyze the transient behavior of FHR during a loss of flow accident. Based on a modified version of RELAP5/MOD4.0, a model of this ordered pebble bed FHR is established. This paper presents the simulation results for complete loss of forced reactor coolant flow accident with different RCP power-off transient rotation characteristics. It is suggested by the results that main pump without moment of inertia is a feasible method and current safety system design is able to provide effective cooling to the reactor in the loss of coolant accident.

KEYWORDS
FHR, RELAP5/MOD4.0, Reactor Coolant Pump, Loss of coolant flow

1. INTRODUCTION

Fluoride salt cooled High temperature Reactors(FHR) are an emerging reactor class that combines attractive attributes from previously developed reactor classes. FHR concepts feature low-pressure liquid fluoride salt cooling, coated particle fuel, and a high-temperature power cycle. Several studies of fluoride-salt cooled, high temperature reactors delineate the potential for attractive economic performance while meeting high standards for reactor safety and security by Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL) and the University of California at Berkeley (UCB) [1,2] . In China, the Chinese Academy of Sciences (CAS) has initiated a large FHR development program to develop and refine future nuclear energy concepts that have the potential to provide significant safety and economic improvements over existing reactor concepts. Shanghai Institute of Applied Physics (SINAP) is leading the CAS FHR program. Within the scope of this project, SINAP will develop Thorium-based Molten Salt Reactor nuclear energy system (TMSR), and plans to construct a test reactor. A TMSR reactor with a fluoride cooled ordered pebble bed design has been suggested by SINAP [3,4], and the design is currently in progress.

The nature and operation of Reactor Coolant System (RCS) is one of the most important considerations in a nuclear reactor and Reactor Coolant Pump (RCP) is “the heart” of this system. Its security rank is the highest level. Hence, a long-term secure and reliable operation is required. If an accident take place, the RCS of TMSR is supposed to prevent the worst consequences. The design and manufacture of the RCS has a key function for TMSR safety. The prediction of flow and temperature as function of time, during a loss of flow accident, is a very important factor in RCS and RCP design.

The transient analysis is still preliminary, but such a preliminary study can give some information about necessary and indispensable data for detailed system design which will effect RCS design in the future.

2. DESIGN OF TMSR

The ordered pebble bed design suggested by SINAP is one possible FHR design [3]. Current design includes reactor power, fuel element, coolant, core, reactivity control, reactor proper, loop system, residual heat removal (RHR), pebble loading and discharge, materials and safety facilities, considering existing technology, large safety margin, uncertainty and simplify of design. Table 1 shows the major parameters of TMSR.

Table 1. Main design parameter of TMSR

Parameter	Characteristic
Fuel	sphere of diameter: 6.0cm
Moderator	Graphite
Coolant(primary, secondary, third)	FLIBE/FLINAK/Air
Power	10.0MW
Inlet temperature	873.0K
Mass flow of coolant(primary, secondary, third)	150.0, 211.0, 46.1kg/s
The number of fuel spheres	8317.0
Pressure of core cover gas	0.3Mpa
Fuel temperature coefficient of reactivity	
Active Region of fuel sphere	-0.00365$/K
Pebble shell of fuel sphere	-0.00505$/K
Reflector	0.002183$/K
Coolant	-0.00197$/K

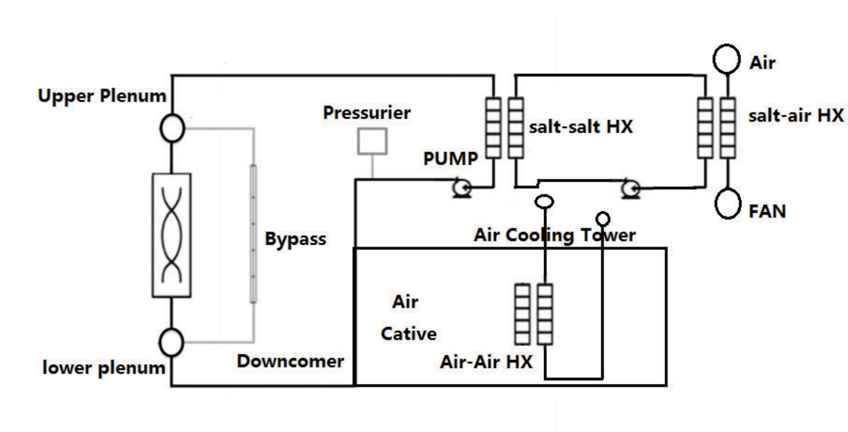

Figure 1. Schematic diagram of TMSR coolant system

Fig. 1 is schematic diagram of TMSR coolant system. It has two molten salt loops including reactor vessel, pump, intermediate molten salt heat exchanger, and molten salt-air heat exchanger in the secondary loop. The coolant in the primary loop is a binary molten salt system of the 66.67%LiF-33.33%BeF_2 (mol) lithium-beryllium-fluoride (FLiBe) originally used in the molten salt reactor [5,6], the coolant in secondary loop is lithium-soduim-potassium-fluoride (FLiNaK) and the coolant in the third loop is air.

This reactor is a graphite-moderated thermal reactor. The core of TMSR consists of active area and graphite reflector. The active region of TMSR is comprised of an ordered pebble bed area and coolant area. Fuel spheres in the pebble zone are static and in an ordered arrangement, and the packing fraction is 68.06%. The gaps between pebbles form a flow channel to remove heat from bottom to up by reactor coolant. A cross sectional view of reactor core is presented in Fig. 2. The active region is 180.0cm high, and the opposite side distances are 139.0cm and 134.6cm. The diameter of experiment and control channels is 13.0cm.

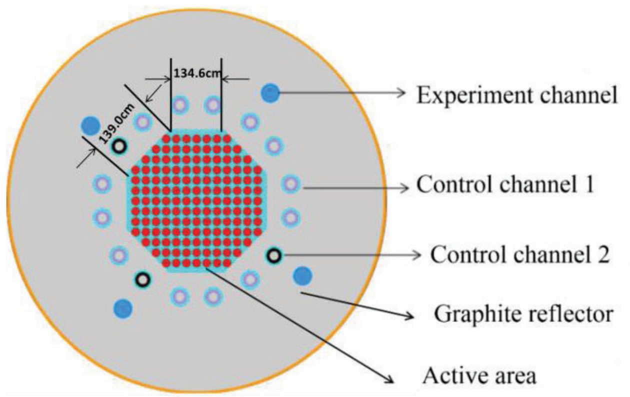

Figure 2. TMSR core cross sectional view

TMSR uses spherical fuel elements with TRISO particles containing UO_2. The pebble diameter is 6.0cm and fuel zone diameter is 5.0cm with 0.5cm thickness graphite shell. The TRISO fuel particle consists of a UO_2 fuel kernel surrounded by a porous buffer layer, and successive isotropic layers of dense inner pyro carbon (IPyC), chemically vapor deposited silicon carbide (SiC), and dense outer pyro carbon (OPyC). Fig. 3 presents the structure of the fuel pebble, and the geometric and material definitions are listed in Table 2. The same fuel element is used in HTR-10 [7].

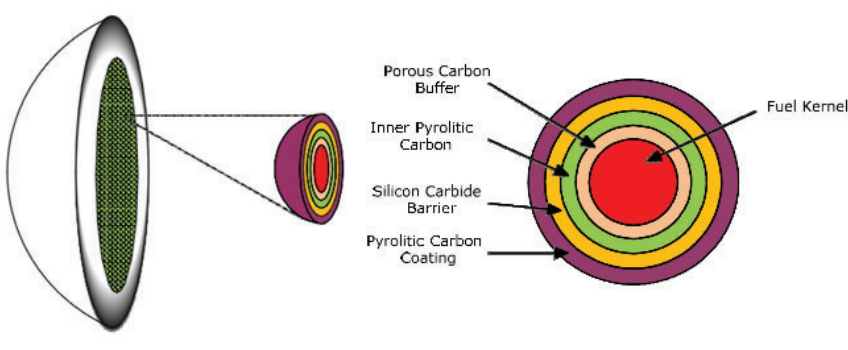

Figure 3. TMSR fuel sphere

Table 2. Geometry and composition of TMSR fuel sphere [7, 8]

Region	Parameters	Component Dimension	Value (mm)	Material	Density (g/cm3)
Pebble Fuel	Active Region	Outer Radius	25	TRISO/ Matrix	1.75
	Pebble shell	Thickness	5	Graphite	1.75
	Pebble fuel	Outside Radius	30		
Triso Particle	Fuel Kernel	Outer Radius	250	UO_2	10.40
	Porous Pyrolytic Carbon	Thickness	95	Porous PyC	1.05
	Dense Inner Pyro carbon	Thickness	40	PyC	1.90
	Silicon Carbide	Thickness	35	SiC	3.18
	Dense Outer Pyro carbon	Thickness	40	PyC	1.90
	Triso Particle	Outside Particle	460		

3. SIMULATION OF TMSR BY RELAP5/MOD4.0

RELAP5 is a generic transient analysis code for thermal-hydraulic systems using a fluid that may be a mixture of steam, water, noncondensables, and a nonvolatile solute [9, 10, 11]. Due to its successful performance in analysis of light water reactor accidents and its availability, a modified version of RELAP5 is expected to be the basis transient analysis code for TMSR . RELAP5/MOD4.0 is the latest version among RELAP5 family, developed by Innovative system software [12].

To model TMSR reactor, thermodynamic properties of FLiBe, FLiNaK and air, and a simple heat transfer correlation for forced convection through pebble beds are implemented in RELAP5/MOD4.0. The thermal hydraulics correlations for an ordered pebble bed are obtained by considering the bed as a porous media with a given porosity [13]. The transient analysis of the TMSR reactor is performed by the RELAP5/MOD4.0 code for hot full power reactor operating conditions at the beginning of life. The components of a nuclear reactor are represented with a user-defined nodalization that contains hydrodynamic control volumes and junctions that represent flow paths between control volumes and heat structures. Nodlization of TMSR system is shown in Fig. 4.

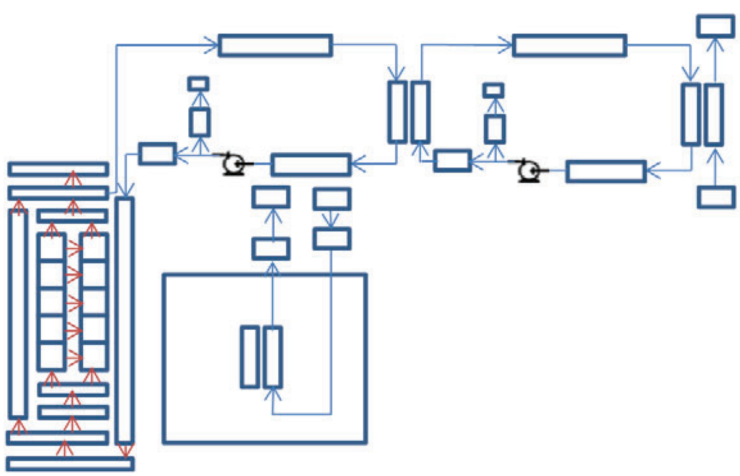

Figure 4. Nodalization of TMSR System

TMSR system and components, particularly the reactor core, differs from light water reactor systems. While Relap5/MOD4.0 code is used to model the reactor, the precise description of each of the components of TMSR is defined as follows.

3.1. Reactor core

Fuel sphere elements of TMSR-SF1 are in random arrangement in the core active area. The hydraulics control volume of activity area is defined as the coolant of core activity area, and the equivalent model is presented as follows: 1) the same coolant volume, 2) the same height of activity area, 3) the same pressure drop.

Heat structure modeling of core activity area: area is defined as the fuel sphere elements. The equivalent medel is presented as follows: 1) sphere fuel element, 2) homogeneous fuel area 3) the same porosity 4) modified wakao formula [14, 15] is used to calculate heat transfer for coolant through pebble bed and the formula is as follows:

$$\mathrm{Nu} = 2 + 1.1 \times (Re_d \times 0.32)^{0.6} \times \mathrm{Pr}^{1/3} \qquad (1)$$

Where Re_d is Reynolds number based on superficial velocity, and 0.32 is factor by porosity of porous medium packing fraction .

RELAP5 does not allow an outer surface to have mixed boundary conditions [8]. This brings a significant inadequacy when modelling the reflector region with the coolant channels. In order to tackle the reflector with the coolant channels, two heat structures have been defined, one on each side of the coolant channels (molten salt channel shown in Fig. 5). Heat can then be transferred from the hotter to the colder heat structure, but solely by convection of the molten salt.

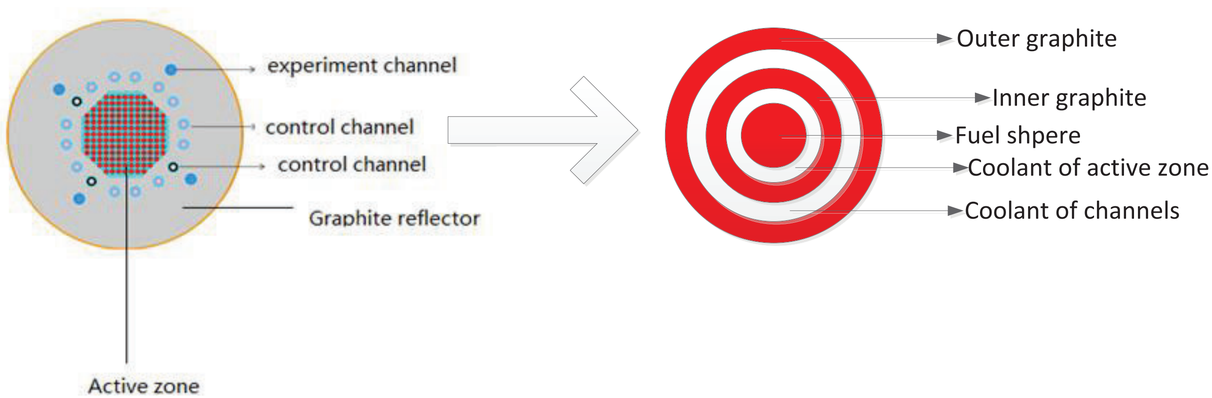

Figure 5. Heat structure and molten channel of reflector

TMSR core is divided into several channels to simulate core activity hot spot temperature. Assuming the peak temperature of the fuel pebble in the central channel is hot spot temperature, power and flow is the same as the design value in modeling. In present study, the core is divided in two parallel channels (cooling channel and hot channel) and thirteen axial layers. A portion of the power is allocated to each layer during the steady state conditions. During transients, the power causes a temperature variation in the layers which introduces a reactivity disturbance. The net reactivity is then fed back into the defined point reactor kinetics model. It is important to notice that the axial power profile has been chosen to be fixed and has been determined once from the reference core. The radial and axial power peaking factors are respectively 1.21 and 1.27.

3.2 Heat exchanger

The primary heat exchanger is a tube-and–shell design. The secondary loop uses molten salt-air heat exchanger to transfer heat to the final heat sink. To model coolant flows along the tube side modeling , the equivalence model is presented as follows: 1) flow area is the area sum of all heat exchange tubes; 2) same hydraulics diameter; 3) same flow length; 4) same pressure drop.

Coolant channels are complicated on shell side and the equivalence model is presented as follows: 1) same coolant volume; 2)same channel length; 3) hydraulics diameter by default; 4) same pressure drop.

The equivalence method for heat structure of heat exchanger is presented as follows: 1) cylindrical heat structure; 2) same heat exchange area; 3) same heat transfer hydraulic diameter; 3) tube and shell side adopts the generalized DB heat transfer formula, and keeping heat transfer capacity the same as design value by adjusting FOULING factor.

3.3 Pipe and molten salt pump

The overall layout of pipe keeps the same as total length, elevation and pressure drop, by using general hydraulics control volume and heat structure.

The molten salt pump is a vertical cantilever fluid centrifugal pump. There is blanket gas and an overflow port in the pump bowl. The overflow port is used to control molten salt level. When molten salt liquid level exceeds the overflow port position, molten salt of the pump bowl flows into the overflow tank through overflow port. The default pump model of code can't simulate blanket gas and overflow port effect.

Molten salt pump function is divided into two parts: pressure regulating function gained through blanket gas and pressure driven by pump. To simulate the pressure regulating function gained through cover gas of pump bowl, a pressurizer is added in the pump inlet. To simulate the pressure driven, a pump component is chosen and the equivalence is presented as follows: 1) the same volume 2) the same volume flow 3) the same elevation and pressure head 4) built-in Westinghouse pump 5) adjusting the ration of initial pump velocity to rated pump velocity according to the design value 6) minimum moment of inertia simulates pump without moment of inertia.

4. TRANSENT ANALYSIS FOR LOSS OF COOLANT FLOW ACCIDENT

A number of faults could result in a mass flow decrease in the reactor coolant system. Only complete loss of forced reactor coolant flow is discussed in this paper, considering TMSR system design. A complete loss of flow accident may result from a mechanical or electrical failure in RCPs or from a fault in power supply to the pump. This accident is classified as condition Ⅱ accident of TMSR. The protection for loss of coolant flow accident is provided by under voltage or under frequency signal or low coolant loop flow. According to the conceptual design report of SINAP [4], TMSR reactor is operating at 10.0MW with only one main pump without moment of inertia in the coolant circuit. The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions. When electric power is lost, the reactor control system is activated with assuming max delay equals to 2.0 s and the control rods drop to the core with 6.0 s, and the passive residual heat removal system with 120.0KW.

There is hardly boiling phenomenon in the TMSR, since FLiBe and FLiNaK boiling temperature are over 1400K [6]. The maximum fuel temperature allowed by TMSR is 1600K [12]. The most important limits for TMSR will ultimately be established primarily by structural limits, considering TMSR vessel structure material is Alloy N and it's working temperature is under 1100K [13]. So transient analysis for loss of coolant flow is focus on fuel temperature and coolant temperature of outlet.

To model the loss of coolant flow transient of TMSR, variable moment of inertia for the main pump was used to calculate the flow changes of the reactor during the transient. The transients include: 1) moment of inertia of main pump equals to 20.0, 15.0, 10.0, 5.0,1.0,0.01 (kg·m^2) without startup passive residual heat removal system in short time; 2) moment of inertia of main pump equals to 0.01(kg·m^2) with startup passive residual heat removal system in long time. The results of these simulations are presented in the following section.

4.1. Loss of coolant flow in short time

In the analysis, mass flow, nuclear power, fuel peak temperature, fuel peak temperature and core outlet coolant temperature are calculated.

In the loss of coolant flow accident, the loop flow coasts down by flywheel inertia. Fig. 5 presents the mass flow transient time dependence with different moment of inertia for main pump. When the moment of inertia is smaller, the coast down flow is decreased more quickly. When the moment of inertia is 0.01 or 20.0(kg·m^2), the time of half mass flow is 0.61 or 26.15s. So the coast down flow is sensible to flywheel inertia.

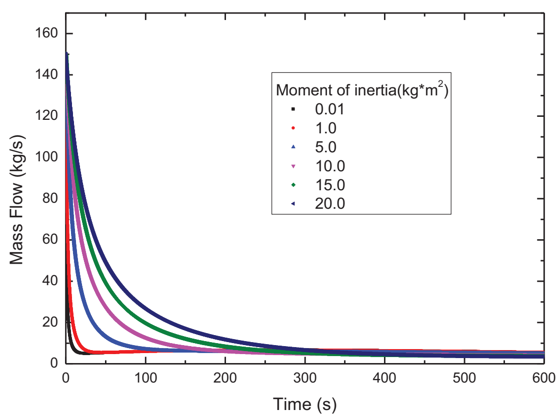

Figure 5. Mass flow

Following the reactor trip on low primary coolant loop flow, the reactor scrams and shuts down. Natural circulation is established in the primary loop and decay heat removal is handled through the resident heat remove system to the atmosphere. When the reactor shuts down, the nuclear power is decreased. The calculated reactor power, peak fuel temperature, peak temperature of fuel surface and coolant temperature of core outlet are shown in Fig. 6-9. The peak fuel temperature is decreased, because of nuclear power is rapidly decreased. The peak temperature is reached soon (3.82-7.59 sec) after transient initiation because the reactor's control system activation delay of 2.0 sec and the trip time of coolant downward flow slowing down at different rate. Meantime, there is small difference of these peak temperatures (1000.2-1001.3 K) that is because the thermal inertia of fuel sphere elements is very large. Increase of core outlet coolant temperatures is less than 45 K and a large increase in the coolant temperature when the moment of inertia is 0.01 or 1.0(kg • m^2), but the peak temperature is less than limited temperature. The temperature rise is ceased by the decrease of power and heat flux after reactor trip.

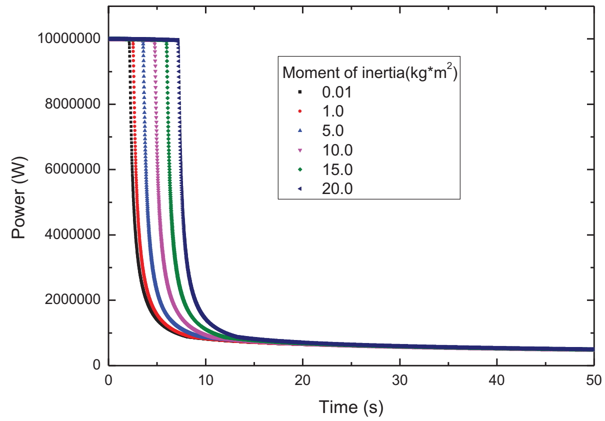

Figure 6. Reactor nuclear power

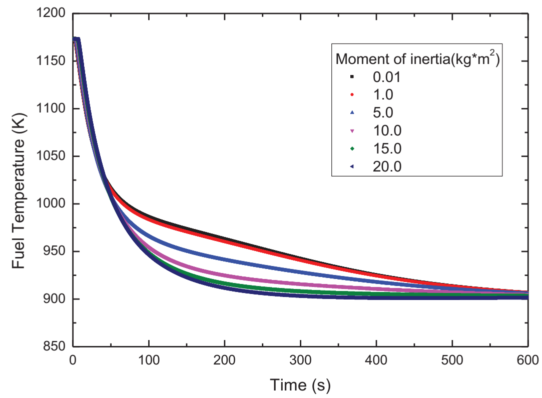

Figure 7. Peak fuel temperature

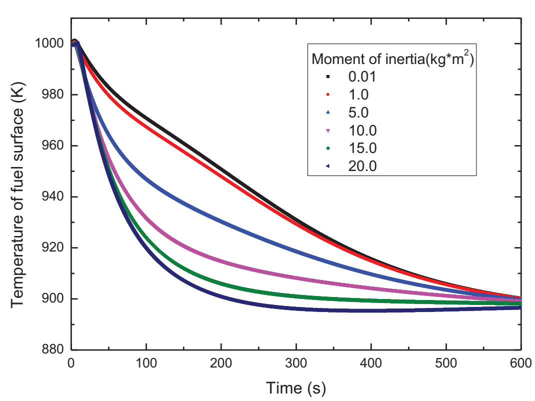

Figure 8. Peak temperature of Fuel surface

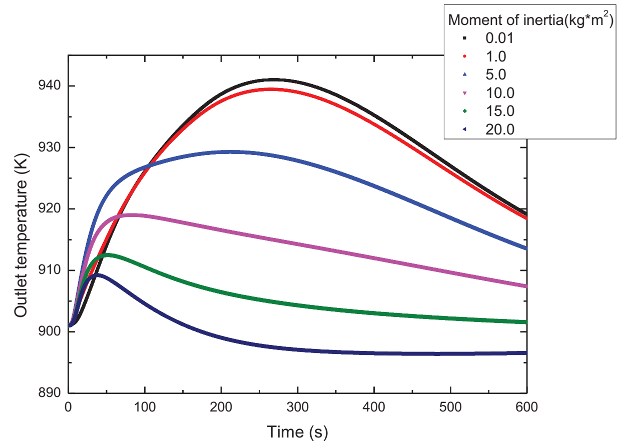

Figure 9. Core outlet coolant temperature

4.2. Loss of coolant flow in long time

In the analysis, core outlet coolant temperature and reactor vessel peak temperature are calculated. The residual heat removal system is started up after 600 sec when coolant downward flow slowing down is tripped. Fig. 10 and Fig. 11 show the changes in core outlet coolant temperature and reactor vessel peak temperature, respectively during the transient that follows the loss of coolant flow. As can be seen, the core outlet coolant temperature and reactor vessel temperature don’t exceed 950K. During the long time of loss of coolant flow, the passive residual heat removal system will transfer heat energy into finalized heat sink and maintain the core in a safe shutdown condition.

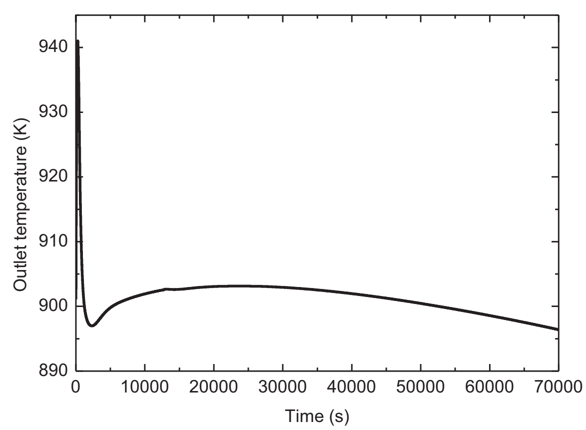

Figure 10. Core outlet coolant temperature

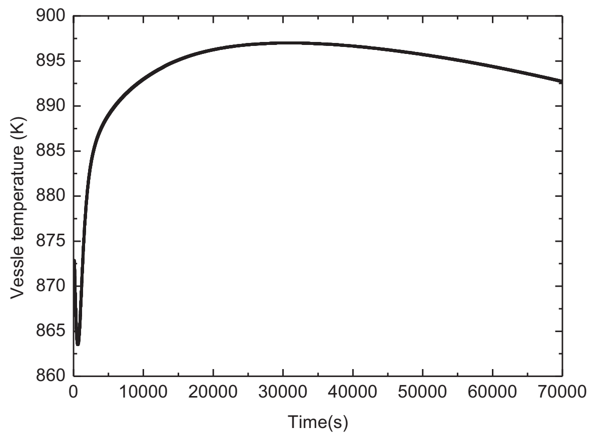

Figure 11. reactor vessel peak temperature

4.3. Summary

Loss of coolant flow transients at full power condition have been analyzed with variable moment of inertia for the pump in the pebble bed molten salt reactor. The simulation results show that the max temperature of fuel surface is not sensible to moment of inertia of the main pump, but the core outlet coolant temperature is sensible to moment of inertia. Moreover for all analyzed conditions the highest temperature of fuel and fuel surface reached in these transients are far below 1200K. The highest core outlet coolant temperature and the highest reactor vessel temperature are below 950K in short time and long time analysis.

5. CONCLUSIONS

This paper aims at the modeling and transient analysis of TMSR. The nodalization of system has been brought out with one dimensional system code REALP5/MOD4.0, which includes the recommended properties of FLiBe, FLiNaK and heat transfer correlation.

Transient analysis results show that the highest temperature of fuel sphere elements is 300 K less than limited value and the core outlet coolant peak temperature is 30K less than limited temperature. The safety margin of core outlet coolant temperature is less than fuel temperature. The reliability of TMSR system, as well as its response during transients and accidents, will depend greatly upon the structure materials that are selected.

In preliminary analysis, the obtained results show that the main pump without moment of inertia is a feasible method according to the simulation results and the current safety system design is able to provide effective cooling to the reactor in the loss of coolant flow accident.

REFERENCES

1. T. Allen, S. Ball, E. Blandford, et al , ‘Preliminary FHR subsystems definition, functional requirement definition, and LBE identification white paper’, *University of California at Berkeley Reports* , **Vol. UCBTH-12-001**, (2013).
2. A. Griveau, ‘Modeling and Transient Analysis for the Pebble Bed Advanced HighTemperature Reacto r (PB-AHTR)’, *University of California at Berkeley Reports,* **Vol. UCBTH-07-001** (2007).

3. Y. Yang, Y. Fu, et al , ‘Reactivity-insertion-transient Analysis of a Fluoride Salt Cooled High Temperature Reactor’, *Proceedings of International Topical Meeting on Advances in Reactor Physics 2014 (PHYSOR 2014)*, Kyoto, Japan, September28 –October 3, on CD-ROM (2014) .
4. ‘Pre-conceptual Design of a 2MW Pebble bed Fluoride Salt Coolant High Temperature Test Reactor’, *Shanghai Institute of Applied Physics Reports,* (2014)
5. J. A. LANE, H.G. Mac Pherson, F. Maslan, *Fluid Fuel Reactor*, Chapter 12, Addison-Weley, Massac husetts, USA (1958).
6. D. F. Williams, L. M. Toth, K. T. Clarno, ‘Assessment of Candidate Molten Salt Coolants for the NG NP/NHI Heat-Transfer Loop’, *Oak Ridge Natl. Lab. Reports*, **Vol.ORNL/TM-2006/12** (2006)
7. J. He, Y. Zou, et al, ‘Fabrication of Spherical Fuel Element for 10MW High Temperature Gas-cooled Reactor ’, *Atomic Energy Science and Technology*, **Vol.37, Suppl.** , pp 40-44. (2003).
8. C. Tang, Y. Tang, J. Zhu, et al, ‘Design and manufacture of the fuel element for the 10MW high tempe rature gas-cooled reactor’, *Nuclear Engineering and Design*, **Vol.218, Issues1-3**, pp 91-102, (2002).
9. ‘Relap5/MOD3.3 Code Manual Volume I:Code Structure, System Models, and solution Methods’, Information Systems Laboratories, Inc. Rockville, Maryland Idaho Falls, USA (2001)
10. ‘Relap5/MOD3.3 Code Manual Volume II: User’s Guide And Input Requirements’, Information Systems Laboratories, Inc. Rockville, Maryland Idaho Falls, USA (2001)
11. ‘Relap5/MOD3.3 Code Manual Volume IV: Models and Correlations,’, Information Systems Laboratories, Inc. Rockville, Maryland Idaho Falls, USA (2001)
12. www.relap.com
13. M. Kaviany, *Principles of Heat Transfer in Porous Media*, Chapter 2-3, Springer, New York , USA (1999).
14. N. Wakao, S. Kaguei, T. Funazkri, ‘Effect of fluid dispersion coefficients on particle-to-fluid heat tran sfer coefficients in packed beds: Correlation of Nusselt numbers ’, *Chem. Eng. Sci.*, **Vol. 34**, no. 3, pp. 325–336, (1979).
15. N. Wakao, T. Funazkri, ‘Effect of fluid dispersion coefficients on particle-to-fluid heat transfer coeffic ients in packed beds: Correlation of sherwood numbers ’, *Chem. Eng. Sci.*, **Vol. 34**, no. 3, pp. 325–33 6, (1979).
16. Zheng Yan-Hua, Shi Lei, Chen Fu-Bing, ‘Coated Fuel Particle Temperature Analysis of the Pebble-bed Modular High Temperature Gas-cooled Reactor’, *Progress Report on China Nuclear Science & Technology* , **Vol.2**, (2011).
17. T. Allen, E. Blandford, G. P. Cao, et al , ‘FHR Methods and Experiments Program White Paper’, *University of California at Berkeley Reports* , **Vol. UCBTH-12-002,** (2013) .