ChinaRxiv

AI translation · View original & related papers at
chinarxiv.org/items/chinaxiv-202306.00249

Simulation of radiation dose distribution and thermal analysis for the bulk shielding of an optimized molten salt reactor Postprint

Authors: ZHANG Zhi-Hong, XIA Xiao-Bin, CAI Jun, WANG Jian-Hua, LI Chang-Yuan, GE Liang-Quan, ZHANG Qing-Xian

Date: 2023-06-18T00:00:00+00:00

Abstract

The Chinese Academy of Science has launched a thorium-based molten-salt reactor (TMSR) research project with a mission to research and develop a fission energy system of the fourth generation. The TMSR project intends to construct a liquid fuel molten-salt reactor (TMSR-LF), which uses fluoride salt as both the fuel and coolant, and a solid fuel molten-salt reactor (TMSR-SF), which uses fluoride salt as coolant and TRISO fuel. An optimized 2 MWth TMSR-LF has been designed to solve major technological challenges in the Th-U fuel cycle. Preliminary conceptual shielding design has also been performed to develop bulk shielding. In this study, the radiation dose and temperature distribution of the shielding bulk due to the core were simulated and analyzed by performing Monte Carlo simulations and computational fluid dynamics (CFD) analysis. The MCNP calculated dose rate and neutron and gamma spectra indicate that the total dose rate due to the core at the external surface of the concrete wall was 1.91 Sv/h in the radial direction, 1.16 Sv/h above and 1.33 Sv/h below the bulk shielding. All the radiation dose rates due to the core were below the design criteria. Thermal analysis results show that the temperature at the outermost surface of the bulk shielding was 333.86 K, which was below the required limit value. The results indicate that the designed bulk shielding satisfies the radiation shielding requirements for the 2 MWth TMSR-LF.

Full Text

Preamble

Simulation of Radiation Dose Distribution and Thermal Analysis for the Bulk Shielding of an Optimized Molten Salt Reactor

ChinaRxiv

Zhang Zhihong[1,2], Xia Xiaobin[1]†, Cai Jun[1], Wang Jianhua[1], Li Changyuan[1], Ge Liangquan[3], and Zhang Qingxian[3]

[1]Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China
[2]University of Chinese Academy of Sciences, Beijing 100049, China
[3]Chengdu University of Technology, Chengdu 610059, China

(Received September 10, 2014; accepted in revised form October 23, 2014; published online August 11, 2015)

The Chinese Academy of Sciences has launched a thorium-based molten-salt reactor (TMSR) research project with a mission to research and develop a fourth-generation fission energy system. The TMSR project intends to construct both a liquid fuel molten-salt reactor (TMSR-LF), which uses fluoride salt as both fuel and coolant, and a solid fuel molten-salt reactor (TMSR-SF), which uses fluoride salt as coolant and TRISO fuel.

An optimized 2 MWth TMSR-LF has been designed to address major technological challenges in the Th-U fuel cycle, and a preliminary conceptual shielding design has been developed for bulk shielding. In this study, the radiation dose and temperature distribution in the shielding bulk due to the core were simulated and analyzed through Monte Carlo simulations and computational fluid dynamics (CFD) analysis. The MCNP-calculated dose rates and neutron and gamma spectra indicate that the total dose rate due to the core at the external surface of the concrete wall was 1.91 µSv/h in the radial direction, 1.16 µSv/h above the bulk shielding, and 1.33 µSv/h below it. All radiation dose rates due to the core were below the design criteria. Thermal analysis results show that the temperature at the outermost surface of the bulk shielding was 333.86 K, which was below the required limit value. These results demonstrate that the designed bulk shielding satisfies the radiation shielding requirements for the 2 MWth TMSR-LF.

Keywords: Molten salt reactor, High temperature, Monte Carlo calculation, Radiation dose, Neutron and gamma spectra, Thermal analysis
DOI: 10.13538/j.1001-8042/nst.26.040603

Introduction

In 2011, the Chinese Academy of Sciences (CAS) launched a thorium-based molten-salt reactor (TMSR) research project with a mission to research and develop a new fission energy system by 2020 [1]. The Molten Salt Reactor (MSR) [2] is a class of nuclear fission reactors that uses molten salt as both fission/fissile material and coolant [3]. Unlike solid-fuel reactors, the fuel is dissolved in fluoride salt, pumped into the reactor core, and the heat is absorbed and removed by the molten salt itself in MSRs [4]. The MSR was selected as one of the candidates for future nuclear energy development by the Generation IV

International Forum (GIF) in 2002 [5] due to its numerous advantages, including good neutron economy, inherent safety, online reprocessing, reduced radioactive waste, and nuclear nonproliferation [6, 7].

Since 2005, GIF has focused on MSR development from thermal spectrum molten salt reactors. Many studies have investigated the radiation safety of MSRs. Elsheikh [8] presented safety characteristics of MSR power plants and assessed them in comparison to other solid-fueled light water reactors. Merk and Konheiser [9] applied the unstructured mesh neutron transport code HELIOS to determine the actual neutron flux and power distribution for an advanced molten salt fast reactor design. Zhu et al. [10] developed a one-dimensional neutron diffusion and transient analysis code to simulate neutron diffusion and transients in an MSR. However, none of these studies accounted for the effect of operating temperature.

The TMSR project intends to construct two types of MSRs: a liquid fuel molten-salt reactor (TMSR-LF) that uses fluoride salt as both fuel and coolant, and a solid fuel molten-salt reactor (TMSR-SF) that uses fluoride salt as coolant and TRISO fuel. A preliminary conceptual design has been completed for a 2 MWth liquid fuel molten-salt reactor, and primary shielding calculations and design have been performed for the 2 MWth TMSR-LF. The primary goal of radiation shielding is to protect personnel from radiation hazards by determining the minimum material thickness required to reduce dose rates to acceptable levels. Another critical consideration is the high operating temperature of the molten salt fuel, which was designed to enter the core of the 2 MWth TMSR-LF at 908.15 K. The high temperature around the core is unsuitable for shielding materials, so a water-cooled thermal shield filled with cooling fins was designed to remove heat and reduce temperature. The performance of this bulk shielding design in attenuating dose rates and reducing temperature must be assessed to confirm radiation shielding safety for the 2 MWth TMSR-LF. It is important to note that only the core of the 2 MWth TMSR-LF was considered as the radiation source; auxiliary components such as primary system pipes, the heat exchanger (IHX), and the pump were not included in the analysis.

The As Low As Reasonably Achievable (ALARA) principle guided radiation protection for the 2 MWth TMSR-LF. Additionally, national standards for radiation source safety [11] and associated guidance were utilized in the shielding design and analysis. Based on these guidelines, shielding analyses were performed for the entire bulk shielding. Monte Carlo simulations were carried out to simulate radiation dose distribution and neutron spectra due to the core for the bulk shielding of the 2 MWth TMSR-LF, and thermal analysis was performed to determine the temperature distribution of the designed bulk shielding for this high-temperature reactor.

II. Method

A. Description of the 2 MWth TMSR-LF and Bulk Shielding

The 2 MWth TMSR-LF considered in this work is a 2 MWth graphite-moderated thermal reactor utilizing solutions of uranium in fluoride carrier salts. A previous parametric study was performed to develop this optimized MSR based on the Molten-Salt Reactor Experiment (MSRE) [12], originally developed at Oak Ridge National Laboratory (ORNL) in the 1960s. Shielding calculations have been carried out and a shielding design was previously presented for the 2 MWth TMSR-LF.

The designed bulk shielding consists of the thermal shield, shield tank, and concrete wall, as shown in [Figure 1: see original paper]. provides information about the 2 MWth TMSR-LF and the bulk shielding. The reactor core, located in the center of the bulk shielding, was a 140 cm diameter by 160 cm high graphite matrix structure composed of 513 graphite stringers mounted in a vertical close-packed array, as shown in [Figure 1: see original paper]. Half-channels were machined in the four faces of each stringer to form flow passages in the assembly. The core of the optimized MSR was surrounded by a graphite lump that serves as both moderator and neutron reflector to achieve good neutron economics. The optimized MSR was designed to be fueled with BeF2-LiF-ZrF4-UF4 (65-29.2-5-0.8 mol%), with uranium atomic fractions of 0.3% ^{234}U, 35% ^{235}U, 64.4% ^{238}U, and 0.3% ^{236}U.

The thermal shield was a water-cooled, cooling fin-filled stainless steel container completely surrounding the reactor core. It was designed to reduce the neutron dose rate by a factor of 10^4 and attenuate the gamma dose rate by a factor of about 10^3 in the horizontal plane. As mentioned, the thermal shield was required to absorb significant heat from the core and was designed so that the temperature difference between its internal and external surfaces would be approximately 150 K.

The fuel pump and heat exchanger were located in a 370 cm diameter by 670 cm high reactor cell. This reactor cell was surrounded by a magnetite concrete shield tank, which in turn sits within an ordinary concrete wall [13]. The reactor cell was not permitted to be entered under any circumstances after reactor operation at power due to high radiation levels. The total dose rate criterion for radiation shielding of the 2 MWth TMSR-LF was 2.5 µSv/h in areas outside the concrete wall. Due to temperature limits of shielding materials, ASME requirements specified that the maximum concrete temperature should not exceed 366 K during operation [14].

B. Method of Analysis

The Monte Carlo code MCNP5 was employed as a calculation tool for numerical radiation transport studies. MCNP is a general-purpose Monte Carlo N-Particle code for neutron, photon, electron, or coupled neutron/photon/electron transport, including capability to calculate eigenvalues for critical systems. For shielding calculations of the 2 MWth TMSR-LF, the fuel was considered static in the core. Total neutrons from the core and fission gamma rays were considered as the radiation source in shielding analysis. Fission gamma rays were determined by source term calculation according to the operation plan. An MCNP model of the core and its bulk shielding was developed following the geometry structure and dimensions given in for Monte Carlo simulations. Neutron and gamma spectra averaged over the core were calculated, then spectra for each shield were simulated and converted to dose rate using flux-to-dose rate conversion factors recommended by ICRP Publication 74 [15]. Data libraries used in these calculations were processed by NJOY based on the most recent ENDF/B-VII nuclear data available.

The computational fluid dynamics (CFD) code FLUENT was applied to thermal analysis for the 2 MWth TMSR-LF and its bulk shielding. A 3D model of the core and shields was created using the 3D mechanical computer-aided design (CAD) program SolidWorks. GAMBIT, a general-purpose preprocessor for CFD analysis, was used to generate meshes and define boundary types for the model. The meshed model was then imported to FLUENT to simulate the temperature field of the 2 MWth TMSR-LF and bulk shielding. Materials such as water and steel were included in the FLUENT database, but other materials such as fuel, graphite, and Hastelloy-N [16] required user definition. Boundary conditions, including fuel flow rate and fuel temperatures at inlet and outlet, also required user definition. Parameters for the CFD simulation are listed in . The heat source was defined based on MCNP5-calculated power density of the core through user-defined functions (UDF) [17–19].

III. Results and Discussion

A. Radiation Dose Distribution

Neutron and gamma spectra, as well as power density averaged over the core, were analyzed first. MCNP5 output was defined by tally specification cards (Fn card). The Monte Carlo program provides seven basic neutron tally types and six standard photon tallies. For measuring neutron and gamma spectra, the tally for flux averaged over a cell (tally F4) was used. For measuring power density, the tally for fission energy deposition averaged over a cell (tally F7) was used. KCODE was employed to specify the source of the 2 MWth TMSR-LF. Simulations ran 15,550 cycles with 10^6 particles per cycle, with 50 cycles skipped before tally accumulation began.

Neutron and gamma spectra averaged over the core are presented in [Figure 2: see original paper]. Neutrons are spread widely from 10^{-9} MeV to about 20 MeV, with the peak in the neutron spectrum occurring between 0.01 eV and 1 eV in the 2 MWth TMSR-LF. Gamma rays have a relatively narrow bandwidth with energy spread from 1 keV to about 20 MeV, showing two sharp peaks around 0.1 MeV.

[Figure 3: see original paper] indicates that power distribution was symmetric in the axial direction of the core, with the peak appearing at the midplane. The power peak did not appear at the center of the radial direction because control rods were located at that position, as seen in [Figure 1: see original paper].

A transport calculation was performed to simulate radiation dose distribution under maximum power conditions shown in [Figure 3: see original paper] to obtain a conservative estimate. For measuring neutron and gamma dose rates, flux-to-dose rate conversion factors written in the Dose Energy Card (DE) and Dose Function Card (DF) were used. Figure 4: see original paper presents MCNP5 calculation results for radiation dose rate in the radial direction along the core centerline. The 80 cm thickness of magnetite concrete was sufficient to reduce neutron dose rate to 1 μSv/h; however, an additional 80 cm of ordinary concrete was required to attenuate photon dose rate. Total dose rates due to the core at the external surface of the concrete wall were 1.91 μSv/h. Figure 4: see original paper shows radiation dose rate in the axial direction approximately 40 cm from the core centerline, where the power peak appears. Total dose rates due to the core were 1.16 μSv/h above and 1.33 μSv/h below the bulk shielding.

[Figure 4: see original paper] also shows that neutron dose rate due to the core was at the 10^5 level at the thermal shield internal surface and at the 10^2 level at its external surface. Photon dose rate due to the core was at the 10^4 level at the thermal shield internal surface and at the 10 level at its external surface. The attenuation factors for both neutron and photon dose rates meet design objectives.

[Figure 5: see original paper] shows neutron and gamma spectra outside each shield in the radial direction along the core centerline, demonstrating that the thermal shield provides extraordinary attenuation of both neutron and gamma flux.

B. Temperature Distribution

As mentioned, molten salt was designed to be pumped into the core at 908.15 K with a flow rate of 171.86 kg/s. Thermal analysis was performed to determine temperature distribution throughout the system using the CFD code FLUENT.

The porous model [20, 21] was employed to simulate the core because fuel passage dimensions were relatively small compared to core dimensions. Power density distribution shown in [Figure 3: see original paper] was loaded into the CFD model as a heat source via UDF compiled in the C programming language.

[Figure 6: see original paper] presents the temperature profile of the bulk shielding in the radial direction along the core centerline, with cooling water flow rate in the thermal shield at 6.3 kg/s. Temperature difference between internal and external surfaces of the thermal shield was approximately 150 K, meeting design objectives. Temperature at the internal surface of the concrete wall was 333.86 K, which was below the required limit value.

IV. Conclusion

Radiation dose distribution and temperature distribution for the 2 MWth TMSR-LF proposed in the TMSR project were simulated and analyzed through Monte Carlo simulations and CFD analysis. Neutron and gamma spectra from each shield were first calculated by MCNP5, then total radiation dose distribution was determined based on these spectra using flux-to-dose rate conversion factors. Results indicate that total dose rates due to the core at the external surface of the concrete wall were 1.91 µSv/h in the radial direction, 1.16 µSv/h above, and 1.33 µSv/h below the bulk shielding. Neutron and gamma spectra show that the thermal shield, composed of stainless steel fins and cooling water, provides extraordinary attenuation of neutron and gamma flux.

Thermal analysis was also performed using FLUENT. The porous model simulated the core, and power density calculated by MCNP5 was loaded into the CFD model as a heat source through UDF. Results show that the thermal shield could reduce temperature sharply, and concrete temperature was below design objectives.

The present analysis demonstrates that the designed bulk shielding satisfies the shielding requirements of the 2 MWth TMSR-LF.

References

[1] Jiang M H, Xu H J and Dai Z M. Advanced fission energy program-TMSR nuclear energy system. Bull Chin Acad Sci, 2012, 03: 366–374. DOI: 10.3969/j.issn.1000-3045.2012.03.012

[2] MacPherson H G. The molten salt reactors adventure. Nucl Sci Eng, 1985, 90: 374–380.

[3] Merle L E, Heuer D, Allibert M, et al. Introduction to the physics of molten salt reactors. Corsica (France): Springer Press, 2008, 501–521.

[4] Ergen W K, Callihan A D and Mills C B. The aircraft reactor experiment-physics. Nucl Sci Eng, 1957, 2: 826–840.

[5] DOE. A technology roadmap for Generation IV nuclear energy systems. Nuclear Energy Research Advisory Committee and the Generation IV International Forum, GIF-002-00, 2002.

[6] Nuttin A, Heuer D, Billebaud A, et al. Potential of thorium molten salt reactors: Detailed calculations and concept evolutions in view of a large nuclear energy production. Prog Nucl Energ, 2005, 46: 77–99. DOI: 10.1016/j.pnucene.2004.11.001

[7] Moir R W and Teller E. Thorium-fueled underground power plant based on molten salt technology. Nucl Technol, 2005, 151: 334–340.

[8] Elsheikh B M. Safety assessment of molten salt reactors in comparison with light water reactors. J Radiat Res Appl Sci, 2013, 6: 63–70. DOI: 10.1016/j.jrras.2013.10.008

[9] Merk B and Konheiser J. Neutron shielding studies on an advanced molten salt fast reactor design. Ann Nucl Energy, 2014, 64: 441–448. DOI: 10.1016/j.anucene.2013.07.014

[10] Zhu L, Pu P, Du S, et al. Simulation of neutron diffusion and transient analysis of MSR. Nucl Sci Tech, 2014, 25: 020601. DOI: 10.13538/j.1001-8042/nst.25.020601

[11] Basic standards for protection against ionizing radiation and for the safety of radiation sources. GB18871, 2002.

[12] Robertson R C. MSRE design and operations report part I: Description of reactor design. ORNL USA, ORNL-TM-0728, 1965.

[13] McConn R J, Gesh C J, Pagh R T, et al. PIET-43741-TM-963, PNNL-15870 Rev. 1, Compendium of material composition data for radiation transport modeling. USA: Pacific Northwest National Laboratory, 2011.

[14] ASME. BPVC, Section III-Rules for construction of nuclear facility components. Division 2-Code for concrete containments. ASME, 2013.

[15] ICPR. Conversion coefficients for use in radiological protection against external radiation. ICRP Publication 74, 1996.

[16] Haynes International I. HASTELLOY® N alloy, Indiana USA, 2008.

[17] He W F, Dai Y P, Wang J F, et al. Performance prediction of an air-cooled steam condenser using UDF method. Appl Therm Eng, 2013, 50: 1339–1350. DOI: 10.1016/j.applthermaleng.2012.06.020

[18] Liu Y, Li W Z and Quan S L. A self-standing two-fluid CFD model for vertical upward two-phase annular flow. Nucl Eng Des, 2011, 241: 1636–1642. DOI: 10.1016/j.nucengdes.2011.01.037

[19] Ye F, Xiao J, Hu B, et al. Implementation for model of adsorptive hydrogen storage using UDF in fluent. Phys Procedia, 2012, 24: 793–800. DOI: 10.1016/j.phpro.2012.02.118

[20] Pilehvar A F, Aghaie M, Esteki M H, et al. Evaluation of compressible flow in spherical fueled reactors using the porous media model. Ann Nucl Energy, 2013, 57: 185–194. DOI: 10.1016/j.anucene.2013.01.062

ChinaRxiv 中X

[21] Short M P, Hussey D, Kendrick B K, et al. Multiphysics modeling of porous CRUD deposits in nuclear reactors. J Nucl Mater, 2013, 443: 579-587. DOI: 10.1016/j.jnucmat.2013.08.014

Note: Figure translations are in progress. See original paper for figures.

Source: ChinaXiv —Machine translation. Verify with original.