Annals of Nuclear Energy 55 (2013) 71–74

Contents lists available at SciVerse ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Neutronic design of a very high temperature reactor core with low graphite content

Dionisio Telésforo-Reyes, Juan-Luis François *, Cecilia Martín-del-Campo, Arturo G. Reinking-Cejudo

Departamento de Sistemas Energéticos, Facultad de Ingeniería, Universidad Nacional Autónoma de México, Paseo Cuauhnáhuac 8532, Jiutepec, Mor., CP 62550, Mexico

ARTICLE INFO

Article history:
Received 25 November 2011
Received in revised form 22 October 2012
Accepted 29 November 2012
Available online 19 January 2013

Keywords:
VHTR
GT-MHR
MCNPX
Radioactive graphite
Neutronic analysis

ABSTRACT

A relevant issue concerning the deployment of the very high temperature reactors is the accumulation of radioactive graphite and its final disposal. One way to reduce the amount of graphite in this type of reactors is to replace the reflector with other material. In this study the Gas Turbine Modular Helium Reactor was investigated, and most of the reflector was replaced with Ferritic–Martensitic steel. The Monte Carlo particle transport code MCNPX 2.6 was used. A fully detailed three dimensional core model was set up in order to evaluate the neutronic response of the modified core design. Results show that the proposed core performs well in terms of the neutron multiplication factor and reactivity coefficients. Considering a reactor life time of 60 years, and taking into account 6 years period for the replacement of the removable reflector, more than one thousand cubic meters of irradiated graphite are saved, which represents estimated economic saving of USD $237,541,000 and 7.73×10^{15} Bq in radioactivity.

1. Introduction

Nuclear power is one of the cleanest technologies for electricity production. With very few greenhouse gases emissions in all the technology life cycle, it is, nevertheless, questioned partly due to radioactive wastes concerns. It is a mature technology; since Generation I reactors connected into the grid in the 1950s, until the new Generation III+ reactors, several technological improvements have been achieved. Most of the reactors in operation worldwide are based on the (light or heavy) water technology. New concepts are under development: the Generation IV reactors. One of these concepts is the Very High Temperature Reactor (VHTR), based on the gas coolant technology, and graphite as moderator and reflector. With these reactors, output temperatures close to 1000 °C can be achieved, to be used for electricity production and/or heat applications like hydrogen production.

One of the VHTR reference design is based on helium coolant and prismatic fuel blocks (GIF, 2011), in which fissile particles (TRISO) are embedded in a graphite matrix. Graphite is also used as neutron reflector. Therefore, this type of reactor uses large amounts of graphite, which under irradiation becomes radioactive, and this is considered as an intermediate level waste. In the deployment of VHTRs, the huge amount of radioactive graphite could become an important waste management issue in the future (Bourdeloie and Marimbeau, 2004).

Graphite has some characteristics that make it a very special waste form. The stored "Wigner energy", the possibilities of conventional fire and dust explosion, and the problems associated with isotope inventories of carbon-14 and chlorine-36, are some of the issues. For disposal there are the possibilities of conventional burial, oxidation to the gas phase and release as carbon dioxide (with radionuclide retention as appropriate) or recycling into new graphite or carbon products (Wood, 2006). Nevertheless, graphite recycling is the most technically challenging, because of the high specifications of graphite required for the fabrication of graphite-based components for new reactors. In fact, recycling of graphite is in an incipient stage of development, therefore it is not considered as an option in this study.

Based on this, the purpose of this work is to propose an alternate design of a VHTR with a reduced amount of graphite as reflector, however, with the same technological concept of the Gas Turbine Modular Helium Reactor (GT-MHR), it means: the same prismatic fuel concept with graphite as moderator and helium as coolant. At this stage of the investigation only the neutronic response is analyzed. Thermal–hydraulic issues, which are very important in this type of reactors – like decay heat removal during a loss of coolant –, are out of the scope of this paper. They must be addressed taking into account that the high thermal inertia capability of the graphite will be reduced, and it should be replaced by an active or a passive system, as it is being studied for gas cooled fast reactors (Cheng and Wei, 2009; Epiney et al., 2012).

* Corresponding author. Tel.: +52 777 319 4101; fax: +52 55 5616 1855.
E-mail addresses: dionitel@yahoo.com.mx (D. Telésforo-Reyes), juan.louis.francois@gmail.com (J.-L. François), cecilia.martin.del.campo@gmail.com (C. Martín-del-Campo), reinking@unam.mx (A.G. Reinking-Cejudo).

0306-4549/$ - see front matter © 2012 Elsevier Ltd. All rights reserved.
http://dx.doi.org/10.1016/j.anucene.2012.11.032

2. Reactor core description

The selected design is based on the Gas Turbine Modular Helium Reactor (LaBar, 2002), from which sufficient information is available from the open literature. Talamo's work (Talamo and Gudowski, 2005) was chosen because it is one which is very well documented and gave us the data needed to be able to reproduce some of his results and confirm the validity of our MCNPX model. Fig. 1 shows the horizontal cross section of the GT-MHR core. In this design, the fuel in the core is distributed in three concentric radial regions, depending on the uranium enrichment. From the outside to the inside of the core the following regions are identified (see Fig. 1): (a) the periphery, formed by the reactor grade graphite (H451; density = 1.74 g/cm^3) permanent reflector; (b) the lateral removable reflector, (c) the outer fuel ring (OR), (d) the central fuel ring (CR), (e) the inner fuel ring (IR), and (f) the central removable reflector. The removable reflectors are composed of hexagonal blocks of H451 graphite. Each fuel ring is composed by 36 radial blocks and 10 axial blocks, which represents a total of 1080 hexagonal fuel blocks. Fig. 2 shows a cross section of a hexagonal fuel block, which contains 216 fuel channels and 108 coolant channels. Each fuel channel has fuel compacts of 1.3 cm diameter and 4.9 cm length. Each fuel compact has thousands of TRISO micro-particles embedded in a cylindrical graphite matrix with a packing fraction of 37.5%. The total channel active length is 7.93 m. A TRISO particle is composed of a kernel with the fissile material, surrounded by four layers of different graphite compounds. Table 1 shows the TRISO's composition and dimensions used in this work.

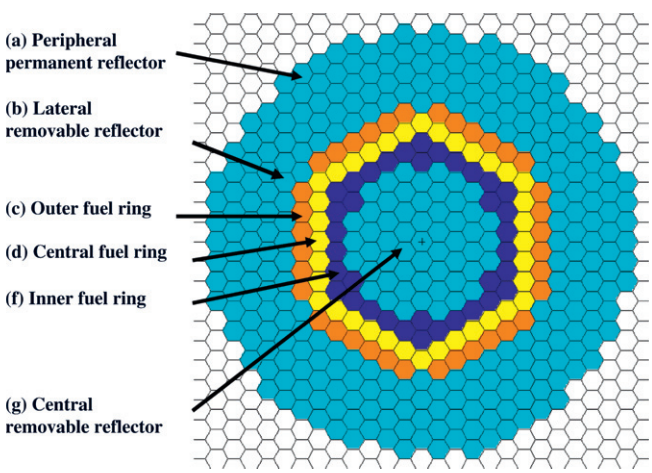

Fig. 1. Cross section of the GT-MHR core (MCNPX model).

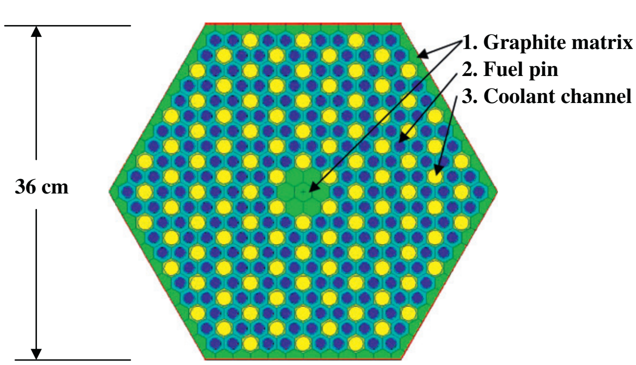

Fig. 2. Cross section of the hexagonal fuel block.

Table 1
TRISO composition and dimensions (Talamo and Gudowski, 2005).

Material	Atomic percentage	Density (g/cm^3)	Dimension (μm)
TRISO kernel	U-235–Th-232 (37.04%); O-16 (62.96%)	10.2	Radius: 150
TRISO porous graphite	C (100%)	1.0	Width: 150
TRISO inner pyrocarbon	C (100%)	1.85	Width: 35
TRISO zirconium carbide	Zr (50%); C (50%)	6.56	Width: 35
TRISO outer pyrocarbon	C (100%)	1.85	Width: 40

3. Core simulations

MCNPX (Monte Carlo N-Particle Transport Code) version 2.6 (Pelowitz, 2008) was used to perform our study. The first step was to set up a MCNPX model of the GT-MHR core, and validate our results against data obtained from the open literature. Afterwards, the GT-MHR core was modified to reduce the amount of graphite in the reflector.

3.1. GT-MHR core

A fully detailed three dimensional core model was set up, where each TRISO particle was explicitly modeled (Telésforo and François, 2009), based on Talamo and Gudowski work (2005). Fig. 3 shows a detailed cross section view of the hexagonal fuel block, where the fuel and helium channels are depicted. The TRISO particles into the fuel compact can be also observed. The ^{235}U enrichment of each fuel ring in the core is the following: IR = 3.6%, CR = 4.4% and OR = 3.9%.

In order to verify our MCNPX model (case: GT-MHR_T&F), the results obtained in terms of the effective neutron multiplication factor (*K*-eff) versus average fuel burn-up were compared against Talamós results. We obtained a very good agreement as can be seen in Fig. 4. The average of the absolute value of the *K*-eff differences between Talamo and our model is 0.7% *dK*, with a standard deviation of 0.5%. Even if this difference is small, it can be explained because different MCNP versions, different cross section libraries and different burn-up simulation codes were used. Talamo used MCNP-4c3 version 5 and MCB-1c2 (burn-up simulator) coupled to a nuclear data library based on JEF2.2 extended with JENDL-3.2, ENDF/B-6.8, DCL-200 and EAF-99; meanwhile we used MCNPX version 2.6, which includes CINDER90 for burn-up simulation, and we used ENDF/B-VII based cross section libraries. Concerning our MCNPX simulations, we ran a total of one million neutron histories (4000 neutrons × 250 cycles), which gave us a

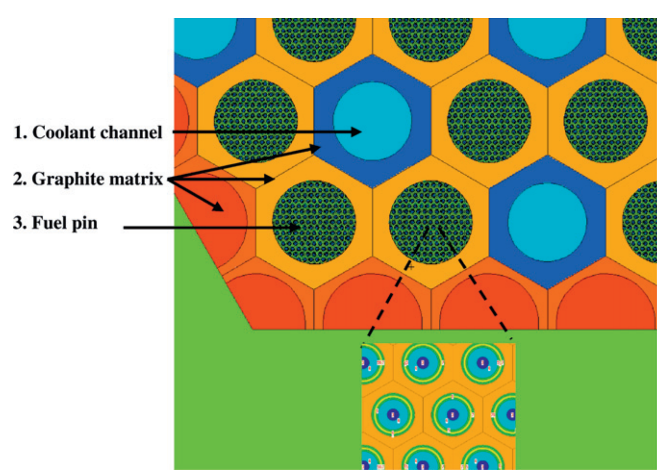

Fig. 3. Detail of the hexagonal fuel block.