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Designing a heterogeneous subcritical nuclear reactor with thorium-based fuel

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ABSTRACT

Using Monte Carlo methods a heterogeneous subcritical nuclear reactor was designed. The reactor is a 110 cm-side cube with thorium-based molten salt as fuel, the moderator is graphite and the startup neutron source is ^{252}Cf. It has ducts to hold the source, the fuel and for irradiation. In the design the k_{eff} was estimated, varying the ducts features and fuel. For the final design, with Th-based salt, the neutron spectra and the ambient dose equivalent due to neutrons, the neutron amplification and the reactor power, were estimated using fresh and burned fuel conditions.

1. Introduction

Accelerator driven subcritical nuclear reactors have been used for burning nuclear waste particularly plutonium (Degweker, 2001). Also, are used in teaching, allowing the students become familiar with reactor physics concepts under safe conditions (Kamalpour et al., 2014; Vega-Carrillo et al., 2015). In the aim to burn nuclear waste thorium salt thermal reactors, Pb and Na cooled fast reactors have been suggested. In reactors with Th-based fuel fewer amounts of actinides are produced, Pu is not generated, and the fission products have shorter half-lives (Degweker, 2001).

Th is not a fissile material, however ^{232}Th is appropriate to capture thermal neutrons, which disintegrate, producing ^{233}U, a fissile isotope. In general, the fuel of these reactors consists of ^{232}Th and ^{233}U for an initial charge (Waris et al., 2010). However, the startup of Th fueled reactor has been also reported (Degweker, 2001).

One of the six types of nuclear reactors selected as part of the 4th generation nuclear reactor concept (GENE-IV) is the Molten salt reactor (MSR). In order to analyze the transient core, due the duct blockage accident, of a small MSR, numerical models for the diffusion equation have been used (Yamamoto et al., 2006), using two energy groups (fast and thermal) and six groups of delayed neutron precursors. It was found that the safety of the MSR is assured. A parameter related to radiation protection around nuclear reactors is the ambient dose equivalent, H*(10) (Dietze and Menzel, 1994); this is an operative magnitude that can be calculated and measured.

The operation of a power reactor with liquid fuel, unlike reactors with solid fuel, has some advantages, such as eliminating poisons, online-recharge fuel, etc. These benefits were identified during the first MSR program. A eutectic mixture of lithium fluoride and beryllium fluoride called FLIBE, with fertile thorium and U or Pu dissolved in the fluoride molten salt ($^{7}LiF–BeF_2–ThF_4–^{233}UF_4$) in liquid fuel, serves as fuel, as a vehicle for processing fuel, and a heat removing agent (Greaves et al., 2012; Nevinitsa et al., 2014).

The objective of this work was to design a heterogeneous subcritical nuclear reactor with Th-based fuel, graphite moderator, and ^{252}Cf as a neutron source. The design was carried out by using the Monte Carlo method where the k_{eff}, the neutron spectra, and the ambient dose equivalent were estimated inside and outside the reactor.

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2. Materials and methods

The purpose of this research is to design a heterogeneous subcritical nuclear reactor with molten salt based on Th, with graphite as moderator and using a source of californium 252, whose dose levels in the periphery allows its use in teaching and research activities. For that reason, several Monte Carlo simulations were performed with the MCNPX code, version 2.5.0 (Pelowitz, 2005). Four geometric bodies (sphere, cylinder, parallelepiped, and cube) were used in calculations. The value of k_{eff} was calculated in function of the amount of fuel, from these results, the final design was based on cubic geometry.

The reactor was modeled with nine ducts, six are for fuel with Th and ^{235}U, the central duct contains the ^{252}Cf startup neutron source, and the remaining two ducts are for irradiation. In the design, Hastelloy N (Ouyang et al., 2014) was used to model the ducts. Fig. 1 shows the side view, in the x–y plane, of the subcritical reactor model. Here, is shown the duct with the 252Cf source, one of the fuel ducts, and the three sections of one of the ducts for irradiation, labeled as front, center and back. All the ducts are inside the graphite moderator that is surrounded by air.

The ^{252}Cf has a half-life of 2.73 years, it decays by alpha emission or through spontaneous fission (3.2% of decay is by spontaneous fission releasing 3.7 neutrons per fission). The source is cylindrical (0.35 Ø cm × 1 cm), it has 20 μg of ^{252}Cf with a source strength of $4.6 \times 10^7\ s^{-1}$ (Sajo-Bohus et al., 2015). The fuel volume in the 6 ducts is 4241 cm^3.

Fig. 2, shows two 3D images of the reactor, figure (a) shows all the ducts without the moderator, whereas figure (b) include the ducts with the moderator.

With this model, the k_{eff} was calculated, as well as the neutron spectra at various points within and outside the reactor, along the three axes. Using the neutron-fluence-to-H*(10) conversion coefficients form the ICRP 74 (ICRP, 1996), the values of H*(10) in contact with the reactor external surfaces (55 cm from the source), and to 20, 50 and 100 cm from the reactor external surfaces were calculated.

In the first set of calculations, the amount of ducts, the pitch and the diameter was changed, as well as the fuel. In these cases fuel was modeled as 90% of Th plus 10% of ^{235}U, and one case was using the salt $^7LiF–BeF_2–ThF_4–UF_4$ with 70, 18, 3, and 9 mol of ^7LiF, BeF_2, ThF_4 and UF_4 respectively (Serrano-López et al., 2013). For these cases the k_{eff} was estimated.

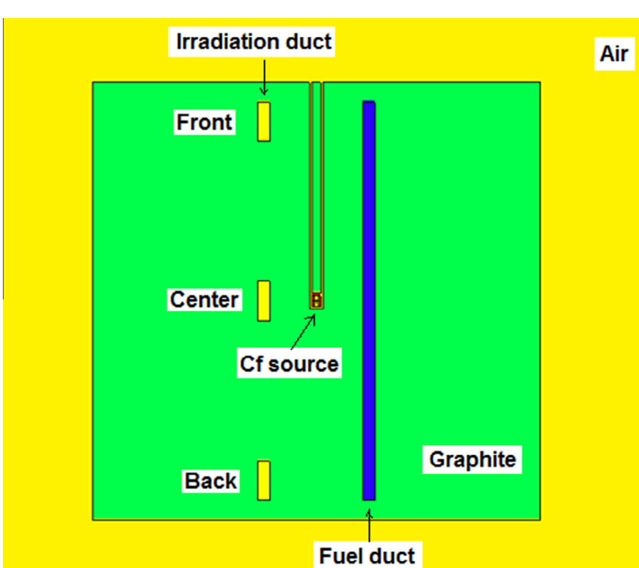

Fig. 1. View of the reactor in the x–y plane.

Another two cases, using 6 ducts 3 cm-diameter with the salt, case 1 was fresh fuel where no ^{233}U was included, just 75% of U (enriched in 3% of ^{235}U) and 25% of Th. Case 2 was burned fuel where neutron capture in Th produces ^{233}U; in this case the ^{233}U was 20%, 5% was Th and 75% of U (enriched in 3% of ^{235}U).

3. Results and discussion

Five cases where the ducts diameter, the amount of ducts, the pitch and the fuel were varied were used to estimate the k_{eff}, these values are shown in Table 1. Simulations were performed with 90% of Th and 10% of ^{235}U. The largest k_{eff} was obtained with 3 cm-diameter ducts, using 6 ducts separated by 10 cm being larger even when 9 ducts were used probably due the neutron moderation. When the fuel in model producing the largest k_{eff} was changed by the Th salt the k_{eff} drops to 0.66 because the amount of ^{235}U was reduced.

In Table 2 the k_{eff} is shown for cases 1 and 2, here the Th salt was used using the reactor model features included in the las row of Table 1.

For case 1 (fresh fuel) the k_{eff} was 0.13, that increases to 0.28 when part of the Th has been converted to ^{233}U (case 2). For burned fuel, the amount of U was maintained (with 3% of ^{235}U), Th was reduced and ^{233}U was produced, increasing the amount of fissile material. Thus, the k_{eff}.becomes 2.15 times larger than the value obtained with the fresh fuel case.

From the value of k_{eff}, the neutron amplification factor and the reactor power (Vega-Carrillo et al., 2015) were calculated for cases 1 and 2. For case 1 the amplification factor and power were 1.15 and 0.088 mW, while for case 2 were 1.39 and 0.229 mW respectively. The amplification factor accounts for the increase of the amount of neutrons per each neutron produced by the startup source, thus for case 1 the amount of neutrons produced by the source is increased in 15%, for case 2 the increase is 39%. For both cases the thermal power is small.

The model with the Th salt was used to estimate the neutron spectra to 4 points allocated to 13, 40, 55 and 155 cm from the source along Y axis, these neutron spectra are shown in Fig. 3.

All the spectra has fission, epithermal and thermal neutrons, as the distance is increased the neutron spectra are reduced. To 13 cm the amplitude of fission neutrons are approximately equal to the amplitude of thermal neutrons, to 155 cm the largest amplitude are due to thermal neutrons due to neutron moderation in graphite.

Fig. 4, shows the neutron spectra in the duct for irradiation (irradiator 1) for the cases 1 (fresh fuel) and 2 (burned fuel). The spectra are in the center and front section of the irradiator. Regardless the fuel, the spectra in the center are the same, however for the front the neutron spectra for case 2 is larger than the spectrum for case 1. In the center the influence of ^{252}Cf is strong, but when the distance is increased the reactor with larger amount of fissile nuclei produces a larger amount of neutrons.

In Fig. 5, the Ambient dose equivalent values calculated from the external face of the subcritical reactor to 155 cm along X-and Y-axis, for cases 1 and 2 are shown. These values are normalized to the source strength (Q). The higher H*(10) values are for case 2 along the Y-axis while the lowest values are for case 1 along the X-axis. Higher doses are when the fuel has more fissile nuclei, and also the values are due the ducts orientation.

The total neutron fluence distribution in the X–Y axis in the middle plane of the reactor is shown in Fig. 6.

There is neutron leakage from the reactor surfaces, being lowest at the corners, this situation is the same regardless the reactor case.