Preliminary Nuclear Design Studies for a Small Modular Advanced High Temperature Reactor (SmAHTR)

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INTRODUCTION

A sustained research effort begun at Oak Ridge National Laboratory (ORNL) decades ago has established the laboratory as a world leader in liquid salts and high temperature materials technologies for nuclear reactors, as well as in the advanced nuclear fuels. Building upon this expertise, a Small Modular Advanced High-Temperature Reactor (SmAHTR, pronounced as "smarter") concept is being developed at ORNL to address the R&D objectives for nuclear energy as formulated by the U.S. Department of Energy's Office of Nuclear Energy Research [1]. The SmAHTR concept is designed to be a small high-temperature reactor sized to be transported via truck to support electricity or process heat applications.

The main purpose of the study discussed in the present paper is to determine the neutronics performance of the SmAHTR core including core lifetime, average fuel discharge burnup, and reactivity coefficients for safety analysis.

DESCRIPTION OF THE SmAHTR CORE

The SmAHTR is a 125-MWth Fluoride salt-cooled High-temperature Reactor (FHR) that uses a liquid salt ($LiF-BeF_2$) as primary coolant and a TRISO-based fuel in cylindrical fuel pins. The inlet coolant temperature is designed to be 650°C, while the outlet temperature of the coolant is 700°C.

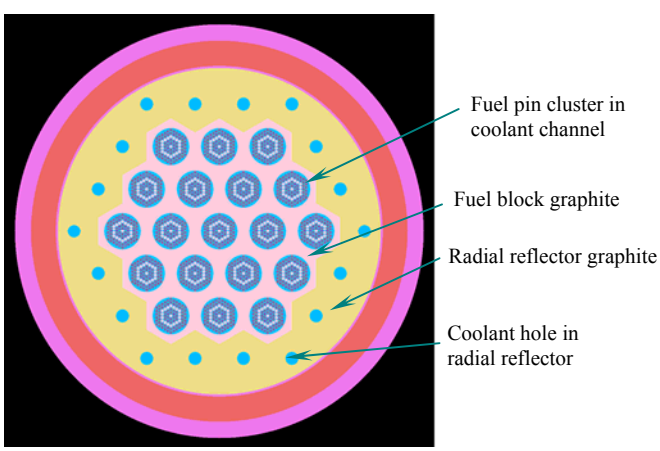

Fig. 1. SmAHTR core cross-sectional view.

The active core (Fig. 1) consists of 19 hexagonal fuel columns with a 22.5 cm apothem (center-to-side distance), each made of 5 stacked fuel blocks for a total core height of 400 cm. The active core is surrounded by a radial graphite reflector with an outer radius of 150 cm. The hexagonal fuel block consists of a fuel cluster that contains 72 fuel pins and 19 graphite pins in a hexagonal array, as illustrated in Fig.2. The fuel cluster is inserted in a circular opening through which the molten fluoride salt coolant flows.

The 2.8-cm diameter fuel pins are made of 19.75% enriched uranium kernels (in $UC_{0.5}O_{1.5}$ form) in TRISO particles, which are in turn embedded in a graphite matrix compact. The packing fraction of the TRISO particles in the fuel compacts is considered to be 40%. With these characteristics, the total mass of heavy metal in the core is 987 kg, while the volumetric power density is set at 9.4 MW/m^3.

The design of the SmAHTR fuel block is based on previous studies performed at ORNL[2] and is optimized for pin power distribution and excess reactivity.

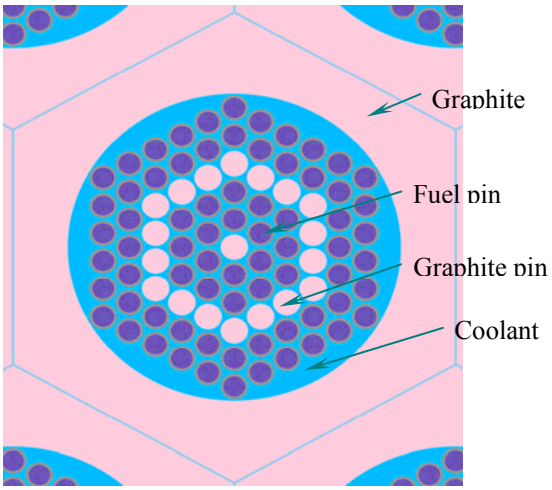

Fig. 2. Detail of the fuel pin cluster in the graphite fuel block.

SmAHTR DEPLETION SIMULATIONS

The evaluation of the lifespan of the SmAHTR core was performed using the 3-dimensional TRITON depletion sequence in SCALE6[3] that uses the Monte Carlo code KENO-VI as a transport solver. Because the depletion simulation uses a multigroup formalism that involves an additional problem-dependent cross sections

evaluation step ("self-shielding"), and therefore additional approximations, the initial state modeled using the multigroup formalism was first validated against a continuous energy KENO-VI calculation. It was found that the multiplication constants for the two simulations are in good agreement: 1.39059 ± 0.00036 for the continuous energy simulation and 1.38982 ± 0.00021 for the multigroup simulation ($\Delta k = 77 \pm 42$ pcm).

The depletion was performed at constant power, using smaller time steps at the beginning of the simulation and increasing the magnitude of these steps as the depletion progresses. The depletion calculations were carried out for a total of 60 time steps, covering a span of ~4.5 years. The core becomes subcritical after slightly more than 2.5 years.

For depletion simulations, the reactor core was divided into 9 zones (3 in the axial direction and 3 in the radial direction).

RESULTS

Figure 3 shows the time behavior in time of the multiplication constant until the excess reactivity of the reactor becomes zero ($k_{eff} = 1.0$).

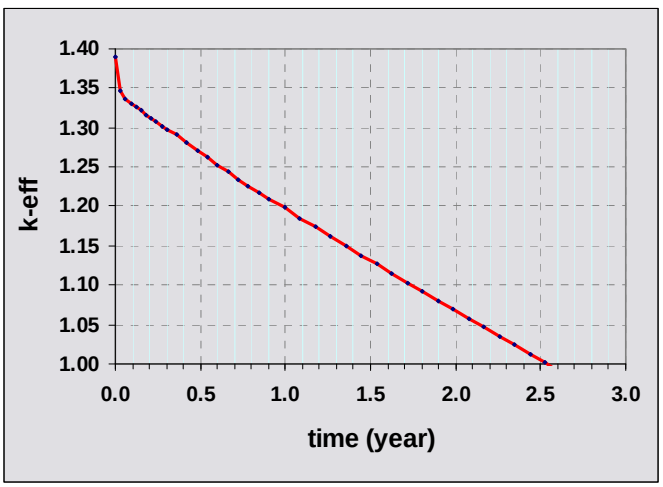

Fig. 3. Time behavior of the multiplication constant for the SmAHTR core.

From the plot in Fig. 3, the life of the core is inferred to be ~2.5 years. The average burnup at discharge is thus calculated to be 116 GWd/MTHM or ~12.5% FIMA (fissions per initial heavy metal atom). The excess reactivity for the fresh core (BOL – beginning of life), calculated as $(k_{eff}-1)/k_{eff}$ is 28,110 pcm.

The fresh core model was also used to calculate the void coefficient of reactivity at BOL. The voided core leads to a negative insertion of reactivity of -2272 pcm.

The isothermal coefficient of reactivity was assessed from the slope of the curve $1/k_{eff}=f(T)$ and, for BOL, is estimated to be -2.58×10^{-5} /°C over the fuel temperature interval [T_{ref}-100, T_{ref}+300]°C, where T_{ref} is the temperature for the reference model: 926.85°C (1200K) for fuel, 675°C for the upflow coolant and graphite moderator, and 650°C for the downflow coolant,. The thermal expansion was not taken into account, which constitutes a conservative hypothesis. The fuel temperature coefficient is -2.21×10^{-5} /°C, the coolant coefficient is -0.19×10^{-5} /°C, while the graphite moderator coefficient is slightly positive: $+0.12\times10^{-5}$ /°C. The coolant's expansion with the temperature was taken into account to calculate the isothermal and the coolant coefficients of reactivity.

The fast neutron flux (E > 100 keV) is estimated to be $(7.8\pm0.5)\times10^{13}$ neutron/cm²/s at the central graphite pin of the central fuel block, and $(2.5\pm0.7)\times10^{9}$ neutron/cm²/s at the inner wall of the reactor vessel, within the acceptable fluence limits for the materials used in those regions.

CONCLUSIONS

The SmAHTR core was modeled and depleted using the ORNL's SCALE code system to assess the cycle length of the core.

The model was also used to estimate certain neutronic parameters of importance for reactor safety (reactivity coefficients), and for materials irradiation fluence estimations.

The model can be further used to estimate the power distribution, additional reactivity effects and kinetic parameters of importance for thermo-hydraulic calculations[4] (e.g., control rod worths delayed neutron fraction, neutron lifetime).

REFERENCES

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