CORE PHYSICS CHARACTERISTICS AND ISSUES FOR THE ADVANCED HIGH-TEMPERATURE REACTOR (AHTR)

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Abstract

Graphite-coated nuclear fuel has traditionally been used for high-temperature reactors that are cooled with high-pressure helium gas. A next-generation, high-temperature reactor concept is being developed that utilizes the same graphite-coated fuel and graphite moderator as helium-cooled reactors, but uses low-pressure liquid fluoride salt as the primary coolant. The Advanced High-Temperature Reactor (AHTR) combines attractive features of gas-cooled reactor fuel, liquid-salt reactor coolant, and liquid-metal-cooled facility design to yield a reactor concept with exceptional safety and economic features. A physics viability study for the AHTR was completed recently, which concluded that there are no fundamental physics barriers with the concept. The study also identified a number of engineering challenges and future research and development needs. As part of the viability study, an initial analysis of the reactor core physics performance was performed, including coolant reactivity effects, fuel burnup cycle length, and transient behaviors. Considerable attention was given to the coolant void reactivity feedback effects, which are highly dependent on salt composition and core geometry. The coolant void coefficient ranges from negative for lighter element salts such as lithium fluoride and beryllium fluoride, to substantially positive for heavier element salts such as sodium fluoride and zirconium fluoride. Due to the large thermal inertial of the graphite-moderated core and coolant pool, and the excellent passive decay heat removal characteristics of the system, temperature transients appear to be very slow and relatively benign. Detailed results of the physics analyses and future development needs will be presented.

Introduction

The production of hydrogen (H_2) by thermochemical processes and the highly efficient production of electricity require significant amounts of energy delivered at very high temperatures. Hydrogen production may require that heat be provided to chemical reagents at temperatures near 850°C. Similar temperatures can produce electricity at efficiencies exceeding 50%, substantially greater than current nuclear plants. In order to provide these temperatures, the reactor coolant exit temperature must exceed 850°C sufficiently to account for temperature drops in the intermediate heat transfer loop from the reactor to the turbines or the H_2 production plant. For this reason, work is under way to develop reactors with coolant exit temperatures of 1000°C. Specifically, the Next Generation Nuclear Power (NGNP) plant project, which is being directed by the U.S. Department of Energy (DOE) Office of Nuclear Energy, Science and Technology (NE), specifies a reactor core outlet temperature of 1000°C as a top level functional requirement.[1]

Historically, helium has been proposed as the coolant of choice for very high-temperature reactors. Consequently, the leading advanced reactor concept being developed by DOE within the Generation IV program is the helium-cooled Very High-Temperature Reactor (VHTR).[2] An alternative option is to use a liquid fluoride salt as the coolant with the same fuel type that has been developed and demonstrated in gas-cooled reactors. The superior heat capacity and transport characteristics of liquids compared with gases enable delivery of high-temperature heat at a near uniform temperature with lower reactor fuel and component temperatures. A new concept, designated the Advanced High-Temperature Reactor (AHTR),[3] is being developed that uses a combination of existing technologies: (1) high-temperature, low-pressure liquid-fluoride-salt coolant, (2) coated-particle graphite-matrix fuel developed for high-temperature gas-cooled reactors, (3) passive safety systems developed for modular liquid-metal-cooled reactors, and (4) a high-efficiency Brayton power cycle for electricity production.

The primary objective of developing the AHTR is to provide an alternative to gas-cooled reactors for high-temperature applications, especially for efficient production of electricity and thermochemical production of hydrogen. In addition to the high production efficiencies of electrical power and hydrogen afforded by the high-coolant temperature, the improved ability of the liquid coolant to hold and transport heat at low pressures results in several significant advantages over gas-cooled systems—higher power output for a similar-sized reactor vessel and containment, reduced reactor vessel thickness, cooler peak fuel temperatures for normal operation and transients, better retention of fission products released from failed fuel particles, and reduced plant footprint. All these factors translate ultimately to significantly improved economics and potential safety advantages.

Although a complete "point design" has not been developed yet for the AHTR, initial core neutronics, thermal hydraulics, decay heat removal, and power conversion analyses have been performed to assess the viability of the design and establish pre-conceptual design parameters.[4] This paper summarizes the neutronics analysis. A brief description of the pre-conceptual AHTR plant design is given below followed by a detailed description of the neutronics analyses and their results.

Pre-Conceptual Plant Design

The current AHTR pre-conceptual design uses a core outlet temperature of 1000°C for the liquid salt in order to respond to the functional requirement specified for the NGNP. This option is sometimes referred to as the AHTR–VT (very high temperature) concept. Because of the liquid coolant in the AHTR, it can deliver heat to the secondary systems with a much smaller temperature drop than for helium-cooled reactors, so a 1000°C outlet temperature is higher than required for the AHTR to deliver the desired 850°C heat to the hydrogen plant or the power conversion units. For this reason, lower temperature versions of the AHTR are being considered also.

Figure 1 shows the general plant layout with the reactor containment building in the center, the turbine building on the left, and the spent fuel storage building on the right. The AHTR plant layout is very similar to the liquid-sodium-cooled S-PRISM plant developed by General Electric[6] because they both share the same feature of low-pressure, liquid coolant. Table 1 provides a list of key design parameters associated with the current pre-conceptual plant design.

The AHTR uses the same coated-particle, graphite-matrix fuel as used in all helium-cooled reactors, including the Gas-Turbine Modular Helium Reactor (GT-MHR) being developed by General Atomics.[5] The coated particles are incorporated into a graphite-matrix fuel compact, which is loaded into a hexagonal graphite-matrix fuel block identical to those used in the Fort St. Vrain reactor. A total of 324 columns of fuel blocks are assembled into an annular geometry with nonfueled graphite reflector blocks filling the interior portion of the annulus and the region between the outer diameter of the core and the reactor vessel. Figure 2 provides a plan view of the core and reflector geometry. The core, inner reflector, and outer reflector blocks are stacked 10 blocks high with an additional layer of nonfueled graphite blocks at the top and bottom of the assembly to form axial reflectors. An elevation view schematic of the reactor core, internals, and vessels is given in Fig. 3.

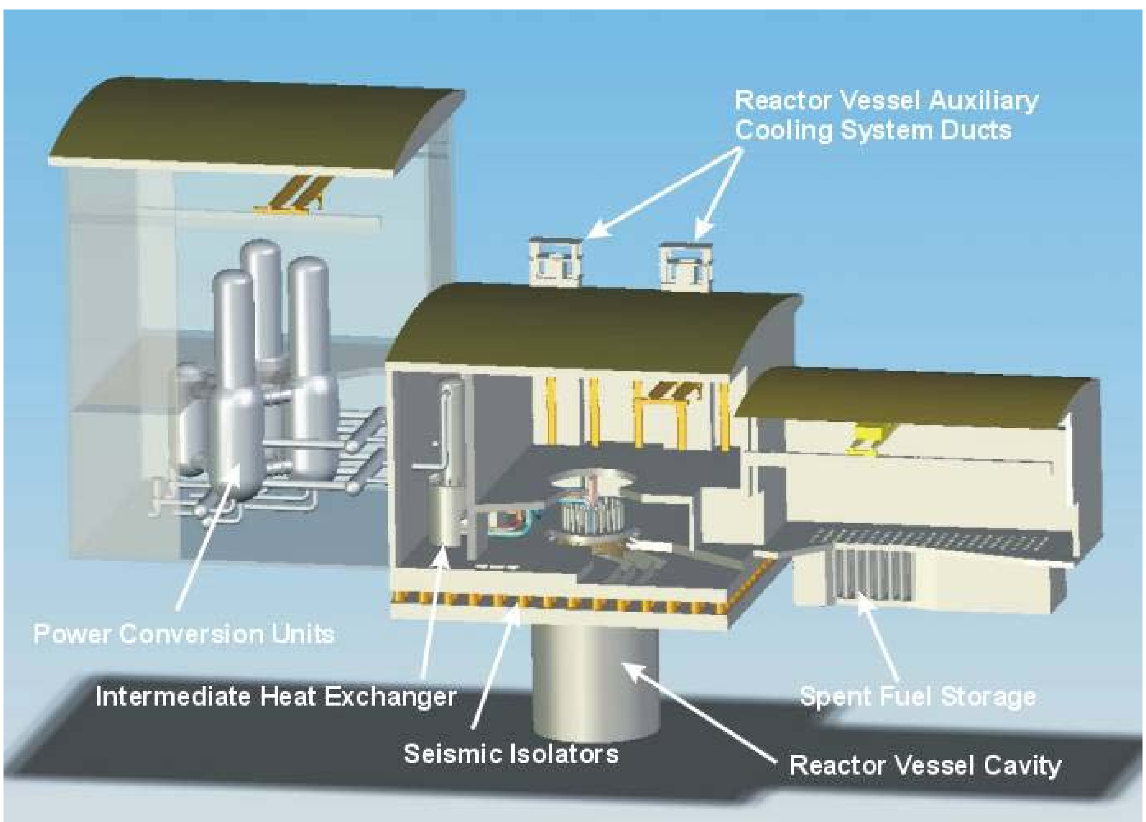

Fig. 1. Conceptual layout of AHTR plant.

Table 1. AHTR pre-conceptual design parameters

Power level	2400 MW(t)	Electrical output	1300 MW(e)
Core inlet/outlet temperature	900°C/1000°C	Power cycle	3-stage multi-reheat Brayton
Coolant (alternate)	Li_2BeF_4 (NaF-ZrF_4)	Power cycle working fluid	Nitrogen (helium longer-term option)
Mass flow rate	12,070 kg/s (20% core bypass)	Core inlet pressure outlet pressure	0.230 MPa 0.101 MPa
Volumetric flow rate	5.54 m^3/s	Pressure drop	0.129 MPa
Channel diameter	0.95 cm	Core shape	Annular
Fraction (core)	6.56%	Core outer diameter	7.8 m
Velocity	2.32 m/s (7.6 ft/s)	Core annulus	2.3 m
Fuel kernel	Uranium carbide/oxide	Core height	7.9 m
Enrichment	10.36 wt % ^{235}U	Pumping power	716 kW
Form	Prismatic	Power density	8.3 MW/m^3
Block. diameter	0.36 m (across flats)	Reflector (outer)	138 columns
Block height	0.79 m	Reflector (inner)	55 columns
Columns	324	Vessel diameter	9.2 m
Mean temperature	1050°C	Vessel height	19.5 m
Peak temperature	1168°C	Vessel thickness	10.0 cm

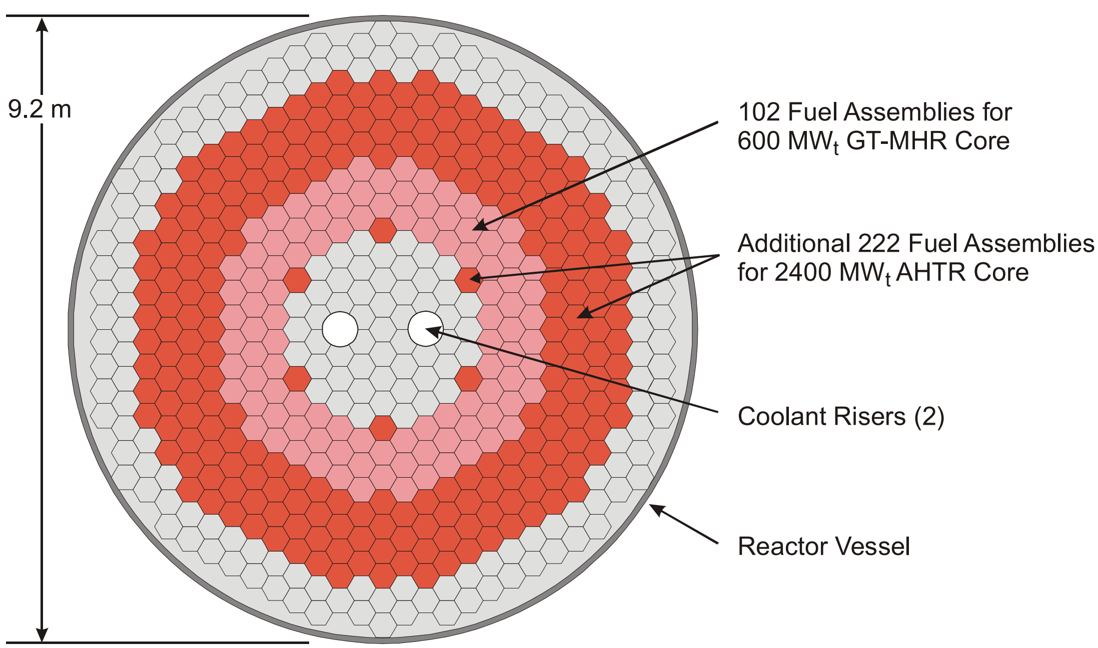

Fig. 2. Plan view of AHTR core showing 324 columns of fuel assemblies.

The primary reactor vessel has a diameter of 9.2 m and a height of 19.5 m. The vessel thickness is 10 cm, which is significantly less than the 22-cm thickness of the VHTR vessel because of the low pressure of the liquid-salt coolant in the AHTR compared with the high-pressure helium of the VHTR. The guard vessel separates the reactor vessel from the reactor cavity cooling system. It is 2.5-cm thick and is separated from the reactor vessel by a 20-cm-thick argon gap. The reference primary coolant is a liquid fluoride salt containing lithium and beryllium (Li_2BeF_4—referred to as “Flibe”). It has a melting point of 459°C, a boiling point of 1430°C, and a density of 1.94 g/cm^3. The heat capacity of Flibe is 4540 kJ/m^3, which is similar to that of water, more than four times that of sodium, and more than 200 times that of helium (at typical reactor conditions). This enables several design performance improvements relative to gas-cooled systems. There is considerable experience with Flibe in nuclear systems: it was used in both the primary and secondary loops of the Molten Salt Reactor Experiment (MSRE) and related test loops.[7]

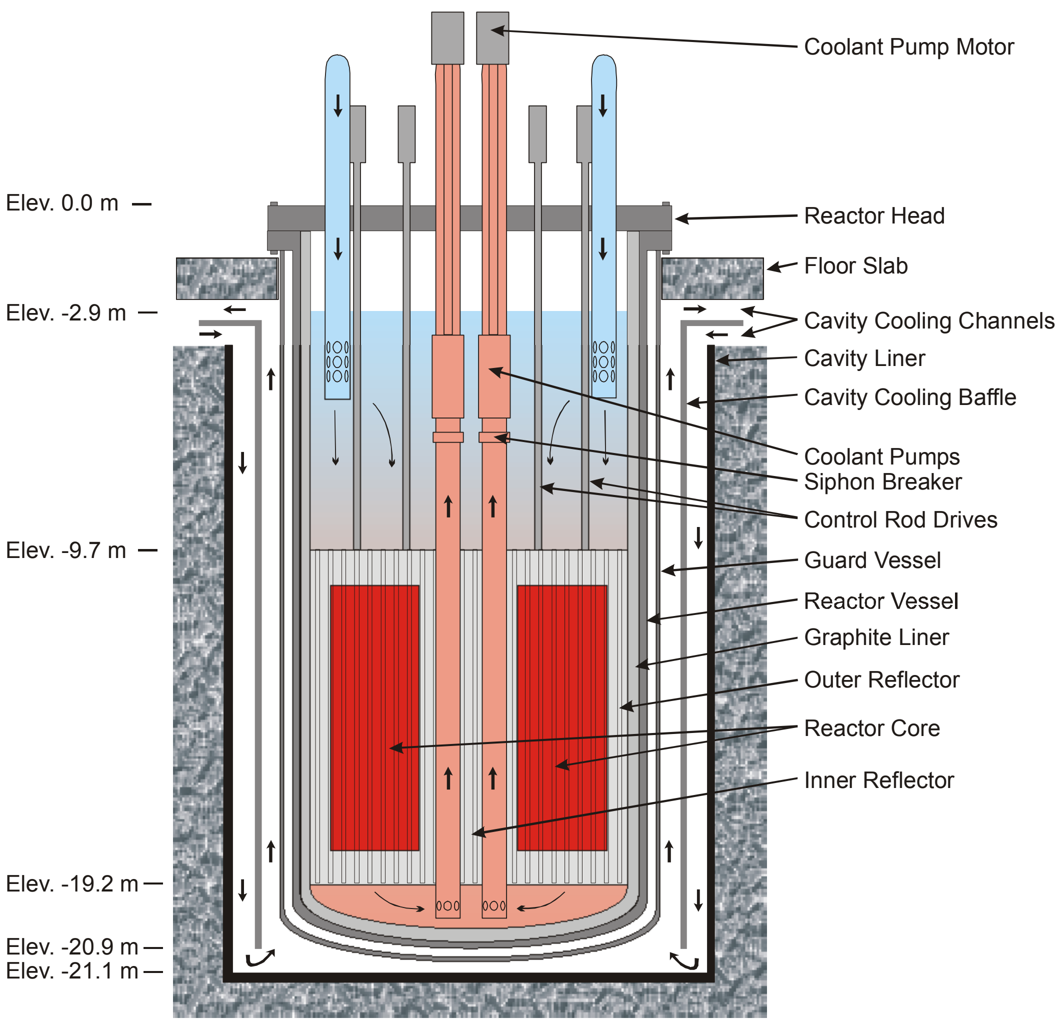

Fig. 3. Elevation view of AHTR core, vessels, and internals.

Physics Analysis

The AHTR core physics behavior is similar to that of the VHTR because they share the same fuel and moderator. A key characteristic of this type of thermal reactor system is a strong temperature feedback effect due to the Doppler broadening of the uranium and generated plutonium resonances that occurs at elevated temperatures. As the temperature of the fuel increases, the parasitic absorption of neutrons by the fertile component of the fuel increases, which reduces the total reactivity of the system and reduces the power level. The AHTR temperature coefficient was estimated as –\$0.01/°C (or 6×10^{-5} k/°C), which is in good agreement with the point design analysis reported for the gas-cooled prismatic VHTR design.[2]

An anticipated difference between the ATHR and helium-cooled reactors is the coolant void coefficient of reactivity, since the nuclear macroscopic cross sections for liquid salts are larger than those for helium. The void coefficient corresponds to the amount of reactivity that is added or subtracted by complete removal of the coolant. Since initial AHTR calculations indicated that the void coefficient can be positive or negative depending on the precise design of the core, the focus of the physics analysis was to characterize this effect more carefully.

Several neutronics calculations were performed to evaluate the coolant void coefficient and to understand its sensitivity to various core parameters. Both ORNL and SNL participated in the physics analysis using slightly different models and assumptions. The results are in good agreement, however. The Monte Carlo N-Particle (MCNP) (version 4C2) code was used for most of the neutronics analyses by both organizations.

The initial SNL analysis used the core model depicted in Fig. 4.[8] A 2.54-cm-diam fuel compact was surrounded by six 0.8-cm-diam coolant channels in a hexagonal array with a 3.41-cm pitch. The fuel and coolant channels were contained in an annular core region (i.e., the individual hexagonal fuel assemblies were not modeled explicitly). This yielded a 10% coolant volume fraction and a 50% fuel-compact volume fraction. The computed void coefficients for total core voiding are given in Table 2 for several salt compositions. For salts containing lithium, it was assumed that the lithium contained pure ^{7}Li isotope, i.e. no ^{6}Li isotope.

Table 2. Void coefficient of reactivity for different salt compositions (initial SNL model)

Salt	Total void reactivity effect (\$)		Salt	Total void reactivity effect (\$)
BeF_2	–1.46		NaF/BeF_2 (57/43)	+1.82
LiF/BeF_2 (66/34)	–0.47		ZrF_4	+1.41
MgF_2/BeF_2 (50/50)	–0.49		NaF/ZrF_4 (25/75)	+1.88
LiF (Li-7)	+0.16		NaF/ZrF_4 (50/50)	+2.64
ZrF_4/BeF_2 (50/50)	+0.43		NaF/ZrF_4 (75/25)	+3.83
ZrF_4/LiF (52/48)	+1.25		NaF	+7.05

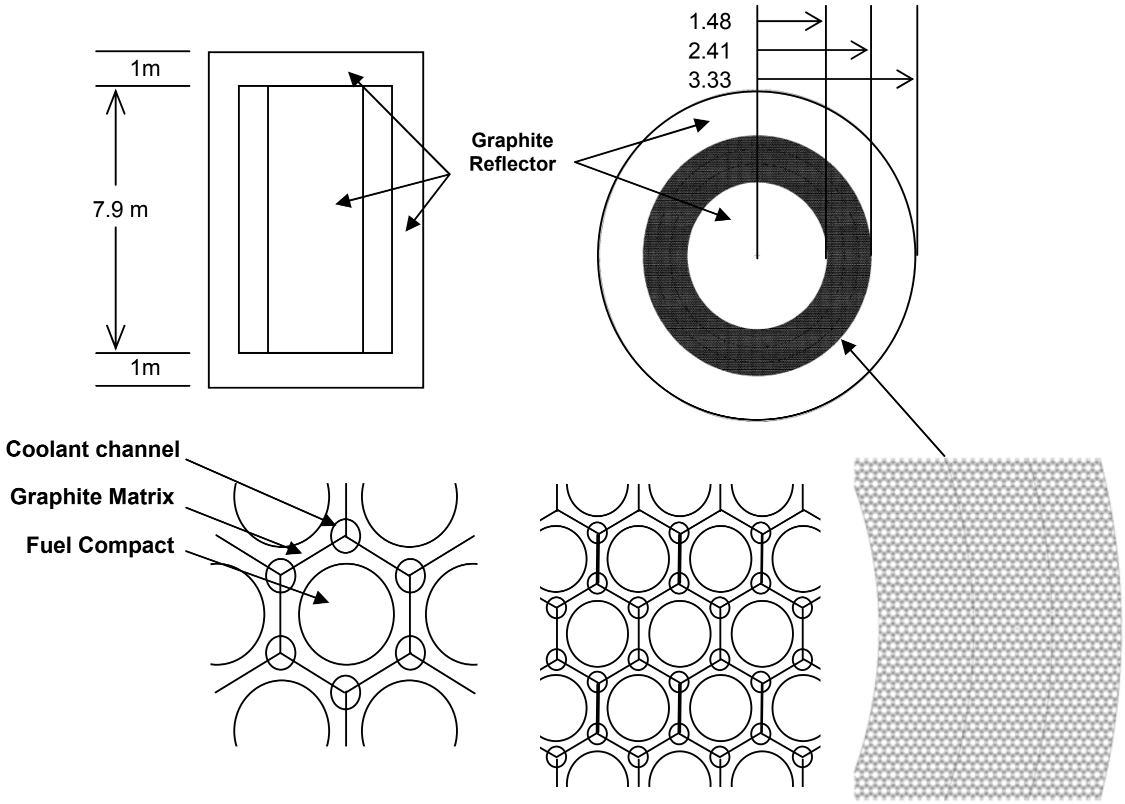

Fig. 4. Reactor core model used in initial SNL analysis.

The ORNL analysis of the void coefficient used a core geometry that modeled individual prismatic fuel elements as depicted in Fig. 2. The fuel assemblies were formed into a 102-column annular core corresponding to the GT–MHR core model, but with liquid salt (Flibe) coolant. The core contained 78 fuel columns and 24 control columns. The focus of the analysis was to explore options for reducing the void coefficient through the use of burnable poisons (BPs) placed either in discrete rods within the fuel assembly or distributed in the graphite blocks. Because the presence of the BP lowers the overall reactivity of the core, the fuel enrichment was increased to 14 wt % ^{235}U for all cases.

Figure 5 shows the result of replacing 14 fuel rods per fuel element with europium oxide rods with varying amounts of erbium loading. Increasing the amount of thermal absorber tends to harden the neutron energy spectrum, which reduces the impact of removing the Flibe coolant. In these calculations, the lithium in the coolant contained 0.01% ^{6}Li. The impact of the 0.01% ^{6}Li was investigated by removing all ^{6}Li from one of the cases shown in Fig. 5, specifically the case with 15 g erbium per BP rod. For this case, the whole core void coefficient dropped from \$0.98 to \$0.04. However, it is difficult to produce lithium with higher than 99.99% ^{7}Li enrichment. Reducing the ^{6}Li content by an additional factor of 10 would be highly desirable, but further reduction would not be helpful because of transmutation effects in the reactor that produce ^{6}Li from beryllium during normal operation.

The impact on the void coefficient of distributing a BP uniformly within the prismatic graphite that forms the fuel assemblies and reflector blocks was also studied. Again using the case of 14 erbium rods per fuel assembly with 15 g erbium per rod, adding 5.0 weight-ppm natural boron to the graphite was observed to drop the void coefficient from \$0.98 to \$0.80. The analysis demonstrates that the positive void coefficient can be reduced by increasing the BP loading in the core. Disadvantages of doing this are increased fuel cycle costs and an increasing void coefficient with burnup due to burnout of the poison.

Updated neutronics analyses were performed at SNL to better match the reference AHTR geometry and fuel/coolant fractions. Two geometric configurations were used: (1) the reference AHTR configuration with a separate coolant channel surrounded by six fuel element channels on a 1.9-mm triangular pitch and (2) a configuration in which each coolant channel is surrounded by an annular fuel compact. The coolant channel diameter was 0.953 cm and the coolant fraction was 7.6% for all of the calculations. The fuel fraction varied in the calculations, with the standard configuration having a fuel fraction of 26.9%. The standard AHTR configuration and revised annular fuel model are shown in Fig. 6.

Two salts were used in the analysis: Flibe—66% LiF and 34% BeF_2 with a density of 1.82 g/cm^3, and 50% NaF and 50% ZrF_4 with a density of 2.906 g/cm^3. The Flibe analyses were performed for two cases—either 0.01% ^{6}Li or pure ^{7}Li. Uranium enrichments were varied from 5 to 20%, with 10% enrichment set as the standard.

The void coefficient results for the updated reference AHTR model are shown in Fig. 7. For a coolant fraction of 7.6% and an enrichment of 10%, a negative void reactivity effect can be attained for a fuel fraction greater than ~0.25 for pure ^{7}Li Flibe and greater than ~0.5 for Flibe with 0.01% ^{6}Li content. Increasing the fuel enrichment allows for slightly lower void reactivity effects, as also shown in Fig. 7.

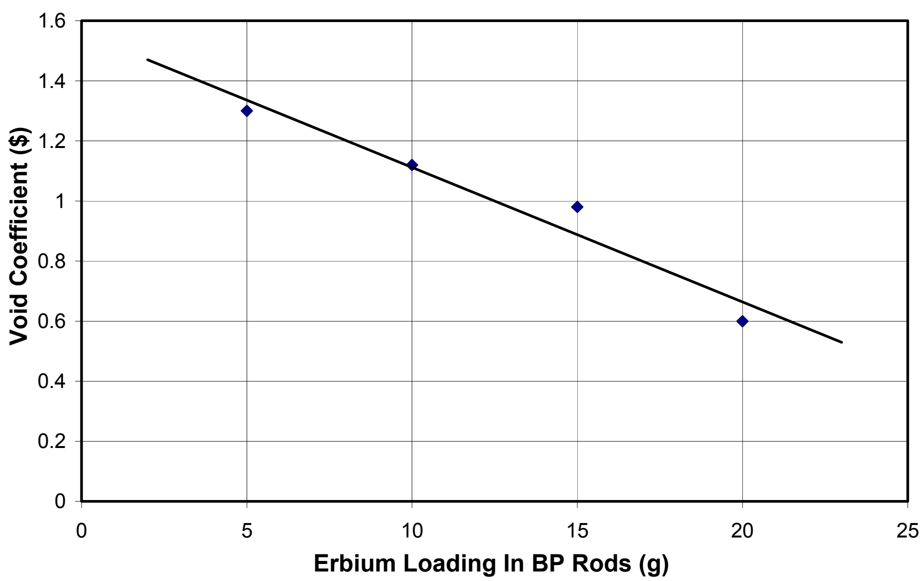

Fig. 5. Sensitivity of void coefficient (whole core voiding) on erbium loading in BP rods.

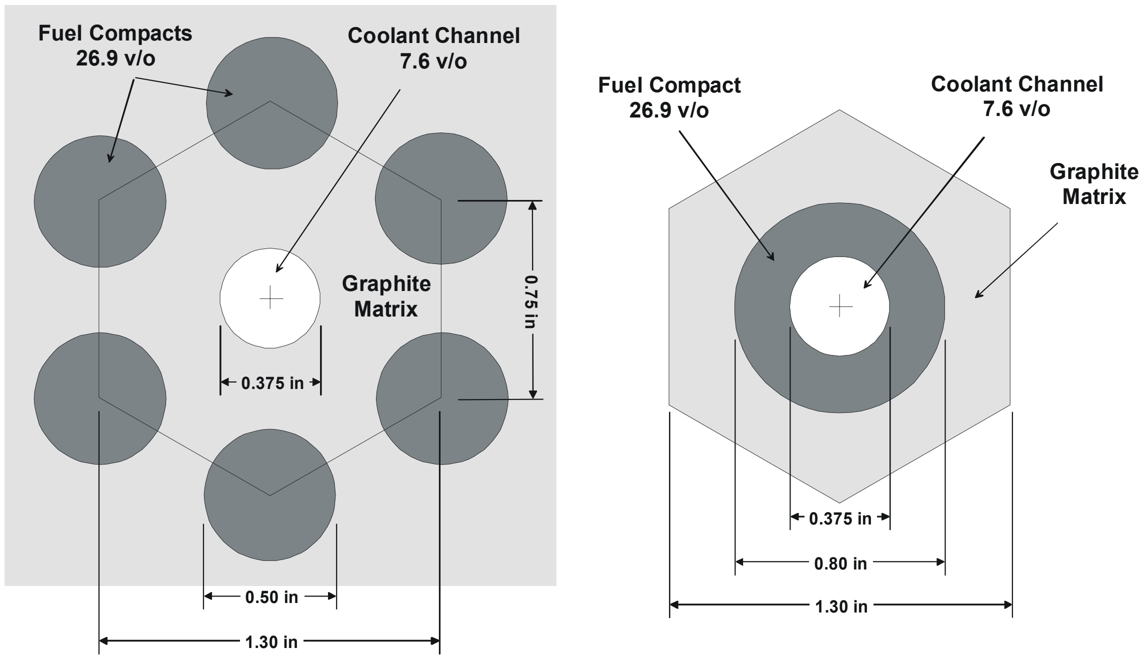

Fig. 6. Model of reference AHTR fuel/coolant geometry (left) and an alternative annular fuel geometry (right).

Calculations for the standard configuration at 10% ^{235}U enrichment with natural boron in the fuel were made to determine BP effects on reactivity. At a fuel fraction of 0.268, no change in the reactivity effects were found for the Flibe (0.01% ^{6}Li) or the NaF-ZrF$_4$ salts. A concentration of 0.00007 g/cm^3 of ^{10}B in the fuel compact was sufficient to make k_{eff} near 1.0. This value is about 14 times more than the quantity of ^{6}Li in the core.

Arranging the configuration in an annular geometry with the same coolant fraction and fuel fraction had little impact on decreasing the void reactivity effect. In fact, slightly more positive effects are found, although the results were within two standard deviations of each other. It is unclear as to why the effects could be slightly more positive.

Several design options exist that can help to lower the coolant void coefficient, and these will need to be explored further. Fortunately, the intrinsic characteristics of the prismatic core design allow the volume fraction of the fuel, coolant, and moderator to be independently varied. The use of discrete or distributed BPs is expected to reduce the void coefficient, as well as geometry changes. Earlier versions of the Canadian power reactors (heavy water moderated) and U.S. production reactors (graphite moderated) had positive coolant void coefficients. With more advanced fuel designs, the Hanford N-Reactor was able to achieve a negative void coefficient, and the advanced CANDU design is projected also to have a negative void coefficient. Similar design approaches are being evaluated for the AHTR.

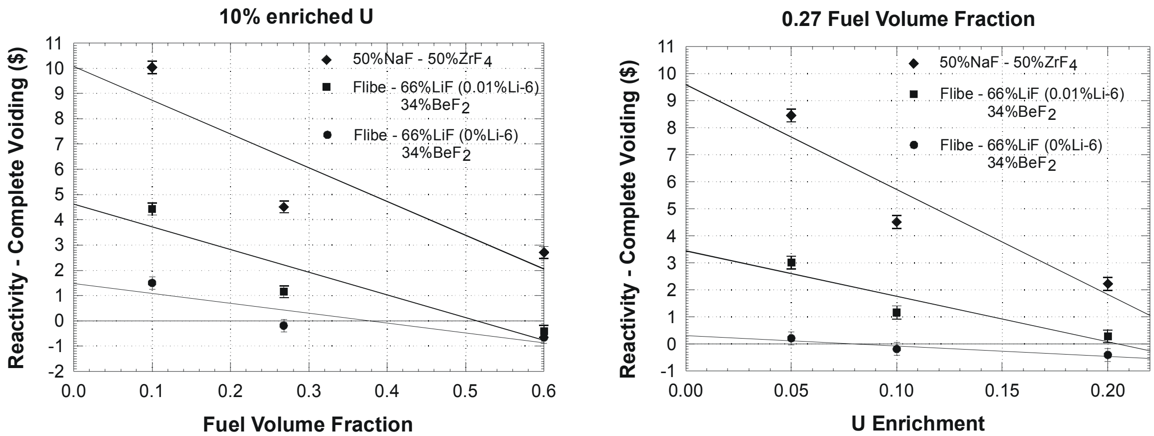

Fig. 7. Sensitivity of void coefficient to fuel fraction (left) and uranium enrichment (right).

Even though a positive void coefficient was predicted for whole-core voiding in some of the designs studied, it was expected that the large negative temperature feedback would act to mitigate transients in the reactor core and would limit power excursions. Figure 8 shows a result for the transient behavior of the core for the case of an instantaneous reactivity addition of +\$0.4. This would be similar to the effect of voiding ~20% of a core cooled with NaF-ZrF_4 salt, which has the largest positive void coefficient of the salts considered. The core temperature feedback balances the reactivity addition, and a new steady-state core temperature is achieved. The assumption here is that the core continues to be cooled at a rate that removes 2400 MW of thermal power. The large negative Doppler coefficient (\$–0.01/°C) combined with large margins to fuel failure allows the reactor core to survive such transients without the need for an active core protection system, and the large heat capacity of the core and the coolant inventory result in a relatively slow transient.

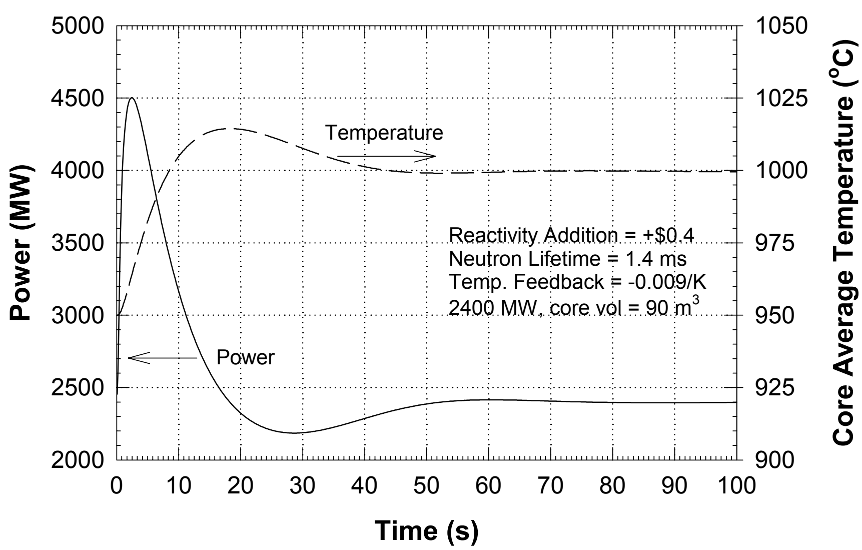

Fig. 8. Thermal power and average core temperature following a \$0.4 reactivity insertion.

Fuel Cycle Analysis

Burnup calculations were performed for the reactor for the standard 2400 MW(t) configuration with 10% enriched and 20% enriched fuel. The calculations were made by having an inner, middle, and outer core with equal volumes. The first curve, starting at t=0, represents the initial core loading with fresh 10% or 20% enriched fuel. When k_{eff} approached 1.0, the fuel was shuffled, with fresh fuel placed in the outer core region, the outer region moved to the middle region, and the middle region moved to the inner region. This was repeated several times so that an "equilibrium" state resulted. Anomalies in the shapes of the curves are due to the rather coarse statistics used in the MCNP calculations.

The results indicate that the 290-m^3 core at 2400 MW(t) would have a burnup cycle of ~330 days (990 days total core lifetime) for 10% enriched fuel and ~510 days (1530 days total core lifetime) for 20% enriched fuel. Figure 9 shows the reactivity as a function of burnup for the case of 20% enriched fuel. These fuel reloading times are similar to those of current LWRs.

Void reactivity effect calculations were performed for the cores near the end of the burnup cycle to determine the effect of the lower ^{235}U content and larger fission product inventory. The results showed that the void reactivity was about the same value as for the fresh core configurations.

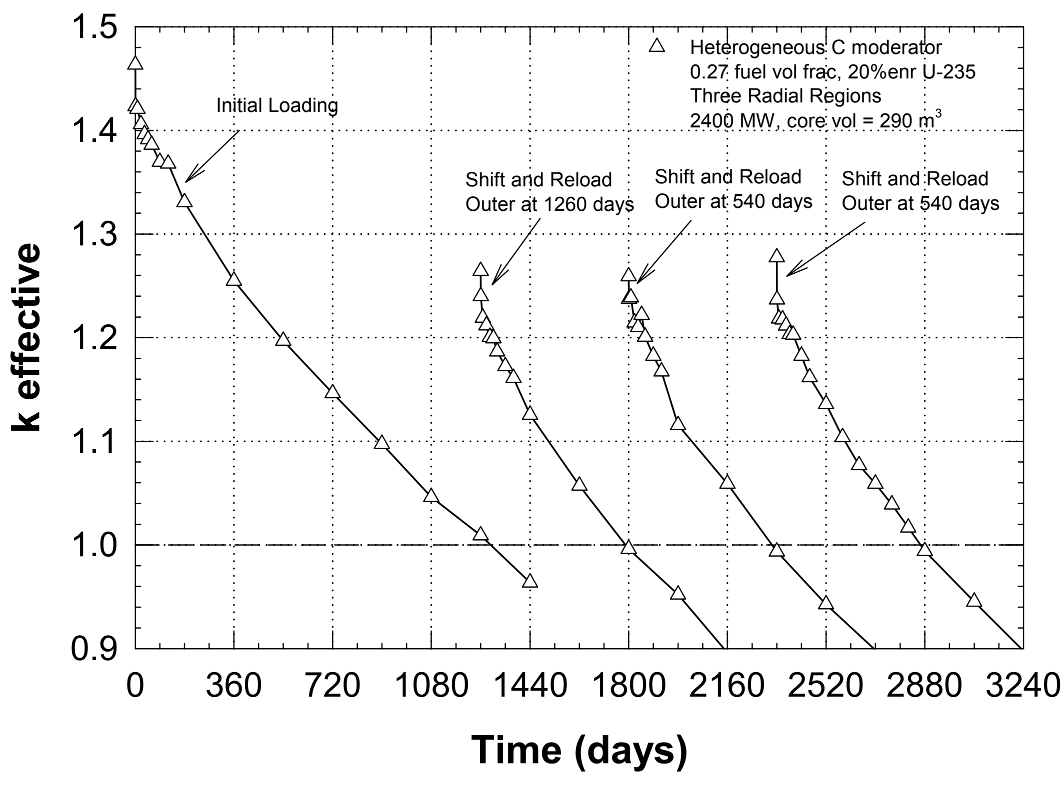

Fig. 9. Fuel burnup predictions for 20% ^{235}U enriched core.

Conclusions

As a new member to the family of high-temperature reactors, the AHTR is defined by two characteristics: (1) a high-temperature fuel and (2) a low-pressure liquid coolant. Our studies indicate that a reactor with these characteristics has the potential for significantly improved economics for the production of electricity and hydrogen while meeting the top-level functional requirements of the NGNP project. The initial viability study for the AHTR is encouraging. There appears to be no fundamental barriers do developing a large commercial system. The neutronics performance of the AHTR appears to be very similar to helium-cooled systems even at substantially high powers (2400 MW versus 600 MW). The coolant void coefficient is the physics parameter of most concern and can vary widely depending on the coolant salt composition and other core parameters. Significant analysis, testing, and engineering design work will be required before the performance characteristics of and potential for the AHTR can be realized. Future physics analyses will include: establishing a new core design, designing a reactivity control system, investigating alternative fuel assembly designs, optimizing power density and peaking factors, and assessing the behavior of the system to additional reactor transients.

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