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Physics Analysis of the LS-VHTR: Salt Coolant and Fuel Block Design

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Abstract

The Liquid-Salt–Cooled Very High Temperature Reactor (LS-VHTR), also known as the Advanced High Temperature Reactor (AHTR), is a new, large [>2400 MW(t)], passively safe, high-temperature reactor concept. It uses a graphite-matrix coated-particle fuel and a graphite moderator similar to the fuel used in modular high-temperature gas-cooled reactors, but with a clean liquid-fluoride salt coolant. The neutronics properties of various salt coolant options are considered with respect to their coefficients of reactivity for various reactor configurations. In addition, several variations on the basic graphite block design of the AHTR are considered that would simplify refueling. The results show that the coolant coefficients of reactivity are negative or very small relative to other reactivity feedbacks, such as the fuel Doppler feedback. This allows several salt-coolants, even some with a positive coolant density coefficient, to be considered for use in the AHTR. In addition, parametric studies of assembly-type clustered-rod configurations show that there is minimal impact on the reactivity coefficients and multiplication factors with appropriate cluster design choices.

KEYWORDS: AHTR, Gen-IV, liquid salt, VHTR, reactivity coefficients

1. Introduction

In 2000, a new concept was proposed for a reactor system that uses high-temperature fuel and high-temperature liquid coolant to address a growing interest in developing nuclear systems that are capable of generating electricity with much higher efficiency than current plants and are also capable of large-scale production of hydrogen. The concept designated the Advanced High-Temperature Reactor (AHTR) [1] uses graphite-coated particle fuel and liquid fluoride salt coolant to create a reactor concept with very favorable economics and safety characteristics. The concept was not sufficiently developed to be considered seriously during the Generation IV Roadmap program conducted by the U.S. Department of Energy (DOE); however, it continues to attract interest from the research and industrial communities. In 2004, the DOE funded an initial viability study for the AHTR as a variant of the Very High-Temperature Reactor (VHTR), which is the leading candidate to be constructed as part of the Next Generation Nuclear Plant (NGNP) project [2]. Therefore, the AHTR concept has come to be known as the liquid-salt-cooled VHTR, or LS-VHTR.

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1.1 The 2005 AHTR Baseline Design

The 2005 effort for the AHTR was focused on developing a new baseline core design that is more optimized for liquid coolant and that satisfies the primary safety and high-temperature applications [3]. The AHTR is an adaptation of the gas-cooled prismatic VHTR design with a replacement of the coolant, a larger core, and some changes in the coolant-hole sizes in the fuel block. In addition, the inner graphite reflector was removed because a loss-of-forced-cooling simulation for the AHTR showed that significant natural circulation of the liquid salt coolant occurred during the transient and provided effective heat transfer to the vessel. Thus the inner reflector is not required, and removing it improves the overall neutron economy of the AHTR by reducing the neutron leakage from the core. It also eliminates the problem of severe power peaking near the inner reflector-core interface, as observed in the gas-cooled VHTR.

The neutronic performance of the core, particularly the reactivity response of the core to voiding of the liquid coolant, is highly dependent on the isotopic and elemental composition of the salt and on the core composition and geometry. A large number of material and design options were considered in an attempt at achieving an optimized core configuration. A parallel task evaluated and compared thermo-physical properties and chemical behaviors of several candidate salts for this application [4]. The baseline salt LiF-BeF_2, referred to as "Flibe," has been used in previous nuclear applications [5] and is the most neutronically favorable of the candidate salts.

Fig. 1 shows the baseline fuel block design used for this study. Several alternative designs were considered as part of the study, including changing the fuel and coolant channel diameters, the pitch between the channels, the number of fuel pins and coolant channels, and a more heterogeneous clustering of the fuel pins. The standard hexagonal fuel block consists of TRISO particle fuel (25% packing fraction, 15% enriched), 1.27-cm-diameter fuel channels, 108 coolant channels, and 216 fuel channels. Because of the superior heat transfer properties of the salt compared with helium, the coolant channels were reduced to 0.935 cm in diameter (7% of the block volume). A ^{7}Li enrichment of 99.995% was used for lithium-containing coolants (unless noted otherwise). The 2004 annual core design and the 2005 core design are shown in Fig. 2, and full design details for the 2005 baseline design are available in Ref. 3.

1.2 Overview of Design Goals

There have been three primary drivers of recent design studies of the AHTR. First, liquid salts are known to contribute significantly to a parasitic neutron capture, and, if coolant is removed, would generally lead to a positive reactivity insertion. Second, refueling a 2400-MWt reactor that consists of 2650 individually fueled graphite blocks is a daunting task; matching light water reactor refueling outage times would be challenging. Third, the substatial cost of the preferred salt—LiF-BeF_2—creates an initial financial burden due to large amounts of enriched lithium (^{7}Li) and handling issues for beryllium. Neutronics analysis has been performed to evaluate the reactivity coefficients in the AHTR to address the first issue. An evaluation of the refueling options based on previous experience is currently under way and is outside the scope of this work. However, one consideration is a redesign of the fuel block to allow easier removal of the fuel with an assembly or cluster-type design. Finally, an evaluation of salts has been performed

Figure 1: Baseline AHTR fuel block.

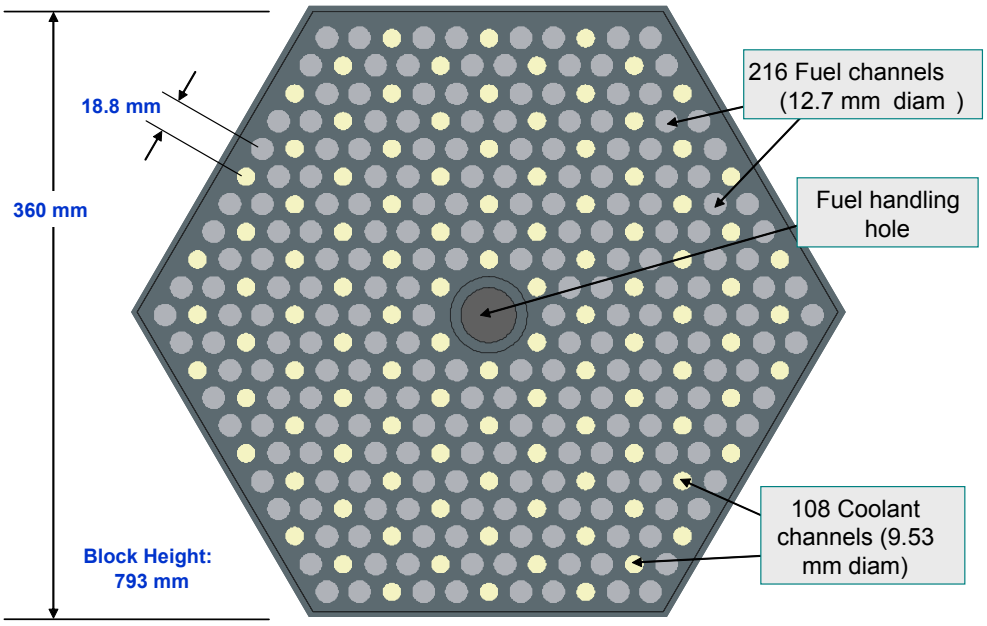
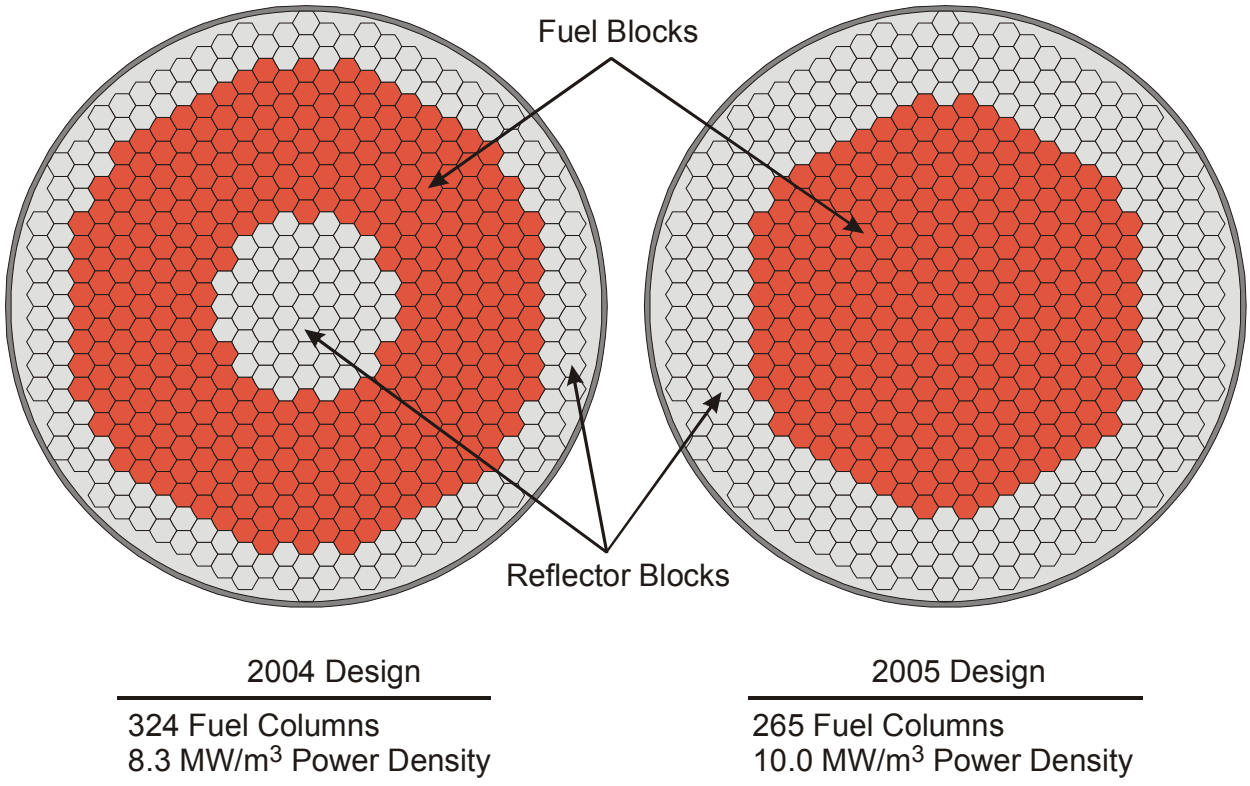


Figure 2: AHTR core and reflector pattern from 2004 (left) and 2005 designs (right)



to address neutronics issues as well as activation and thermal properties [4]. The information presented in this paper will focus on the reactivity coefficients and the fuel block redesign activities. However, the issues mentioned continue to drive design modifications to the AHTR, as will be discussed in Section 5. Note that in addition to neutronics analyses performed at Oak Ridge National Laboratory (ORNL), additional neutronics analyses were performed at Idaho and Argonne national laboratories to further evaluate the reactivity coefficients, determine power distributions to support the thermal-hydraulics analyses, and select the number of fuel batches and enrichment to ensure 18 months of operation [3]. The results of these studies are available in Ref. 3.

2. Description of Analysis Tools

The analysis at ORNL was performed with tools contained within the SCALE 5.1 system [6]. The SCALE system consists of data libraries, cross section processing codes, radiation transport codes, and fuel depletion and activation capabilities that are used in a modular manner to perform reactor, criticality, and shielding analyses. The lattice physics and core calculations were performed using the TRITON depletion sequence, which combines the resonance processing methods (BONAMI/CENTRM), neutron transport (NEWT or KENO-VI), and depletion and activation (ORIGEN-S) tools.

All analyses were performed with the 238-group ENDF/B-VI cross section library that will be available in SCALE 5.1. The resonance processing procedure uses the Bondarenko methodology (in the BONAMI code) for the unresolved resonance region and a continuous energy methodology for the resolved resonance region (CENTRM). The CENTRM resonance processing tool performs a near-continuous energy (pointwise), one-dimensional, discrete-ordinates (S_8, P_3) transport calculation for each pincell type in the problem. This leads to a highly accurate, problem-dependent flux spectrum incorporating resonance interference effects that is used as a weighting function for the cross sections from ENDF/B-VI to create a problem-dependent 238-group cross section library using the PMC code.

Recent additions to TRITON include the ability to perform resonance processing of cross sections due to doubly heterogeneous fuel. This utilizes two CENTRM/PMC calculations in series to determine the spectrum of a particle and weight the particle cross sections and then determine the spectrum in the compact/pebble and re-weight the cross sections accordingly. This is necessary for a proper analysis of the AHTR, which uses particle fuel within compacts.

3. Analysis of Reactivity Coefficients in the AHTR

The thermal-spectrum AHTR core consists of three constituents: TRISO fuel, graphite moderator, and molten-salt coolant. A distinct feature of the AHTR is that a major component of parasitic neutron capture, and a significant amount of moderation, can reside in the liquid-salt coolant. The relationship between capture and moderation is especially significant during transients or accident conditions, e.g., when coolant is removed from the core by a temperature-driven density change, a coolant void is postulated, a gas bubble is trapped in the system, or a breach of the primary circuit occurs. For these scenarios, the increase in reactivity due to coolant density reduction should be minimized or mitigated when the transient or accident occurs. An

assessment of the coolant density reactivity coefficient and the total system temperature reactivity coefficients were evaluated for various salts.

3.1 Assessment of Coolant Density Reactivity Coefficient of Various Salts

Using the analysis methods described in Section 2, the coolant reactivity coefficients were evaluated for a range of salts and over a range of temperatures. Table 1 displays the coolant density reactivity coefficient (reactivity change due to coolant expansion upon heating) and the coolant void reactivity (CVR) (the change in reactivity due to a 100% voiding of the coolant) for a variety of salt coolants in the AHTR design with Er_2O_3 poison. All salts except LiF-BeF_2, which uses enriched lithium (^{7}Li) to minimize the parasitic absorption, contribute to a positive coolant density reactivity coefficient and CVR. Positive voiding or coolant density reactivity coefficients are often characterized as "forbidden" zones for reactors, but it is important to look at the entire reactivity response of the core before passing judgment on the reactor response.

3.2 Assessment of the Total Reactivity Coefficients for Various Salts

It is necessary to consider that the AHTR is a pool-type reactor operating at near-atmospheric pressure with a margin to boiling for the coolant of ~500°C. Thus there can be no depressurization that leads to a sudden loss of coolant. Off-normal conditions that result in a decrease of coolant in the core are either caused by or accompanied by a temperature change. Therefore, the total temperature coefficient should also be considered in the choice of salt coolant. A loss of forced circulation (e.g., break in a major coolant pipe) will result in an increased coolant temperature, but note that the fuel temperature will rise more quickly. A sudden reactivity insertion will cause a rise in coolant temperature, but this rise will lag behind the rapid rise in fuel temperature. Therefore, the relative magnitude and sign of the coolant temperature plus the coolant density coefficient (or total coolant temperature coefficient) should be considered with respect to the total non-coolant (fuel plus graphite) temperature coefficient.

Table 1: Coolant reactivity coefficients for various salts

Salt	Composition (mol%)	Coolant density coefficient (dollars per 100°C)	Coolant void reactivity (dollars)
LiF-BeF_2	67-33	–$0.01	–$0.11
LiF-ZrF_4	51-49	$0.04	$1.40
NaF-BeF_2	57-43	$0.06	$2.45
LiF-NaF-ZrF_4	42-29-29	$0.06	$2.04
LiF-NaF-ZrF_4	26-37-37	$0.09	$2.89
NaF-ZrF_4	59.5-40.5	$0.11	$3.44
NaF-RbF-ZrF_4	33-23.5-43.5	$0.15	$4.91
RbF-ZrF_4	58-42	$0.18	$6.10
KF-ZrF_4	58-42	$0.27	$7.92

Table 2: Reactivity coefficients for the AHTR with and without Er_2O_3 poison

Salt	Composition mol%	Coefficients of Reactivity (Dollars per 100 °C) Coolant Temp	Coolant Density	Coolant Total	Non-Coolant Fuel	Non-Coolant Graphite	Non-Coolant Total
coefficients without Er_2O_3 poison present							
$LiF-BeF_2$	67-33	-$0.01	$0.01	$0.00	-$0.46	-$0.12	-$0.58
$LiF-ZrF_4$	51-49	-$0.01	$0.04	$0.03	-$0.64	$0.03	-$0.61
$NaF-BeF_2$	57-43	$0.00	$0.06	$0.07	-$0.41	$0.02	-$0.39
$LiF-NaF-ZrF_4$	42-29-29	-$0.01	$0.06	$0.05	-$0.47	-$0.03	-$0.50
$LiF-NaF-ZrF_4$	26-37-37	$0.00	$0.09	$0.09	-$0.41	$0.00	-$0.41
$NaF-ZrF_4$	59.5-40.5	$0.00	$0.11	$0.11	-$0.39	$0.05	-$0.35
$NaF-RbF-ZrF_4$	33-23.5-43.5	$0.00	$0.14	$0.13	-$0.37	$0.12	-$0.25
$RbF-ZrF_4$	58-42	-$0.02	$0.17	$0.15	-$0.50	$0.07	-$0.43
$KF-ZrF_4$(	58-42	-$0.01	$0.27	$0.26	-$0.57	$0.05	-$0.52
coefficients with Er_2O_3 poison present							
$LiF-BeF_2$	67-33	-$0.09	$0.00	-$0.09	-$0.92	-$1.54	-$2.45
$LiF-ZrF_4$	51-49	-$0.03	$0.04	$0.01	-$0.64	-$1.42	-$2.08
$NaF-BeF_2$	57-43	-$0.08	$0.06	-$0.01	-$0.86	-$1.40	-$2.25
$LiF-NaF-ZrF_4$	42-29-29	-$0.03	$0.06	$0.03	-$0.47	-$1.38	-$1.85
$LiF-NaF-ZrF_4$	26-37-37	-$0.05	$0.09	$0.04	-$0.87	-$1.41	-$2.27
$NaF-ZrF_4$	59.5-40.5	-$0.05	$0.11	$0.06	-$0.85	-$1.37	-$2.21
$NaF-RbF-ZrF_4$	33-23.5-43.5	-$0.05	$0.15	$0.11	-$0.82	-$1.29	-$2.10
$RbF-ZrF_4$	58-42	-$0.04	$0.19	$0.15	-$0.50	-$1.31	-$1.81
$KF-ZrF_4$	58-42	-$0.02	$0.27	$0.25	-$0.57	-$1.33	-$1.90

In Table 2, a group of temperature coefficients are shown for an AHTR fuel block for conditions with and without Er_2O_3 poison in the graphite moderator. Erbium is a resonance absorber and therefore can be used to enhance the negative temperature feedback at the cost of increased absorption and therefore increased fuel enrichment. The total coolant temperature reactivity coefficient is far smaller in magnitude than the non-coolant (fuel, moderator, and poison) negative temperature coefficient. As the results in Table 2 show, this effect is more pronounced when erbium poison is added to the core because a small change in temperature substantially increases the fraction of neutrons in the 0.3- eV resonance of ^{167}Er.

3.3 Sensitivity of Other Design Parameters

Therefore, if the possibility of a complete voiding without a temperature change can be removed from consideration, the significance of a coolant density change must be considered in conjunction with all other coefficients. For any scenario driven by a rapid reactivity insertion, coolants that exhibit small negative temperature coefficients do not control the system response

because the total response is dominated by the negative reactivity effects in the closely-coupled fuel. However, for salt coolants with a positive total coolant temperature coefficient, a coupled neutronics/thermal-hydraulics analysis must be performed to understand the net reactivity effect. In this regard, it is useful to define a new parameter, the coolant safety ratio, which is the ratio of the magnitude of a positive (total) coolant temperature coefficient and the total non-coolant temperature coefficient. For instance, a coolant safety ratio of 1.9% implies that the fuel and graphite temperature must increase by only 1.9°C to offset a 100°C increase in coolant temperature.

These safety-related parameters for the leading coolant candidates are shown in Table 3. The design basis for the AHTR includes a two-batch core with a 1.5-year cycle, and a burnup of 150 MWd/kgU. Therefore, because of differences in parasitic capture of the salt, the enrichment levels were varied to reach these design specifications. The following parameters were varied in our calculations in order to explore the effects on reactivity coefficients: (a) coolant volume fraction (7%, 15%), (b) Er_2O_3 poison level in the fuel-compact matrix (0, 5 mg/cm^3), and (c) ^{7}Li enrichment level (99.995%, 99.9%). The calculations were also performed with the simplifying assumption that the temperature rise was uniformly distributed over all materials in the core (fuel-coolant-moderator). This assumption is likely to cause an exaggeration of the positive reactivity contributions arising from the salt coolant.

Table 3: Reactivity coefficients for various salts with changes in design parameters

Salt	Composition (mol%)	^{235}U Enrichment (wt%)	Coolant Void Ratio (Dollars)	Total Coolant Coefficient (Dollars per 100 °C)	Salt Coolant Safety Ratio (%)	Total Thermal Coefficient (Dollars per 100 °C)
coefficients <u>without</u> Er_2O_3 poison, 7% coolant fraction						
$LiF-BeF_2$	67-33	14.1	\$0.28	\$0.00	-0.1%	-\$0.58
$NaF-BeF_2$	57-43	15.4	\$2.71	\$0.07	17.0%	-\$0.32
$LiF-NaF-ZrF_4$	26-37-37	15.5	\$2.83	\$0.09	21.5%	-\$0.32
$NaF-ZrF_4$	59.5-40.5	15.8	\$3.35	\$0.11	30.5%	-\$0.24
$NaF-RbF-ZrF_4$	33-23.5-43.5	16.5	\$4.39	\$0.13	53.8%	-\$0.11
coefficients <u>with</u> Er_2O_3 poison, 7% coolant fraction						
$LiF-BeF_2$	67-33	14.3	-\$0.11	-\$0.09	-3.7%	-\$2.54
$NaF-BeF_2$	57-43	15.6	\$2.45	-\$0.01	-0.6%	-\$2.26
$LiF-NaF-ZrF_4$	26-37-37	15.8	\$2.89	\$0.04	1.9%	-\$2.23
$NaF-ZrF_4$	59.5-40.5	16.1	\$3.44	\$0.06	2.9%	-\$2.14
$NaF-RbF-ZrF_4$	33-23.5-43.5	16.9	\$4.91	\$0.11	5.1%	-\$2.00
coefficients <u>with</u> Er_2O_3 poison, 7% coolant fraction, 99.9% ^{7}Li						
$LiF-BeF_2$	67-33	19.2	\$9.56	\$0.17	9.4%	-\$1.62
$LiF-NaF-ZrF_4$	26-37-37	16.9	\$4.99	\$0.12	5.1%	-\$2.16
coefficients <u>with</u> Er_2O_3 poison, 15% coolant fraction						
$LiF-BeF_2$	67-33	15.5	-\$0.64	-\$0.19	-8.8%	-\$2.40
$NaF-BeF_2$	57-43	18.0	\$4.63	-\$0.04	-2.2%	-\$1.81
$LiF-NaF-ZrF_4$	26-37-37	18.7	\$5.83	\$0.08	4.2%	-\$1.78
$NaF-ZrF_4$	59.5-40.5	19.3	\$6.98	\$0.12	7.2%	-\$1.57
$NaF-RbF-ZrF_4$	33-23.5-43.5	21.2	\$10.41	\$0.21	15.0%	-\$1.21

A careful comparison of the results in Table 3 reveals that the reactivity coefficients that affect safety, other than CVR, are more dependent upon the coolant fraction and poison content than on the choice of salt coolant. Therefore, each of the salt coolant options can provide adequate protection during a temperature transient if coupled with a properly designed fuel block. When a coolant /fuel-block combination has a positive total coolant temperature coefficient, a coupled neutronics/thermal-hydraulics assessment should be performed to determine the significance of the positive coefficient. It is also apparent that the lithium enrichment is significant for the LiF-BeF_2 coolant, but not for the LiF-NaF-ZrF_4 coolant, because of the lower relative absorption in lithium compared with the other constituents of the salt (sodium and zirconium). Therefore, to achieve the optimum neutronic performance, the LiF-NaF-ZrF_4 coolant does not have to use extremely high ^{7}Li enrichment (as is typically assumed).

4. Assembly-Type Fuel Loading

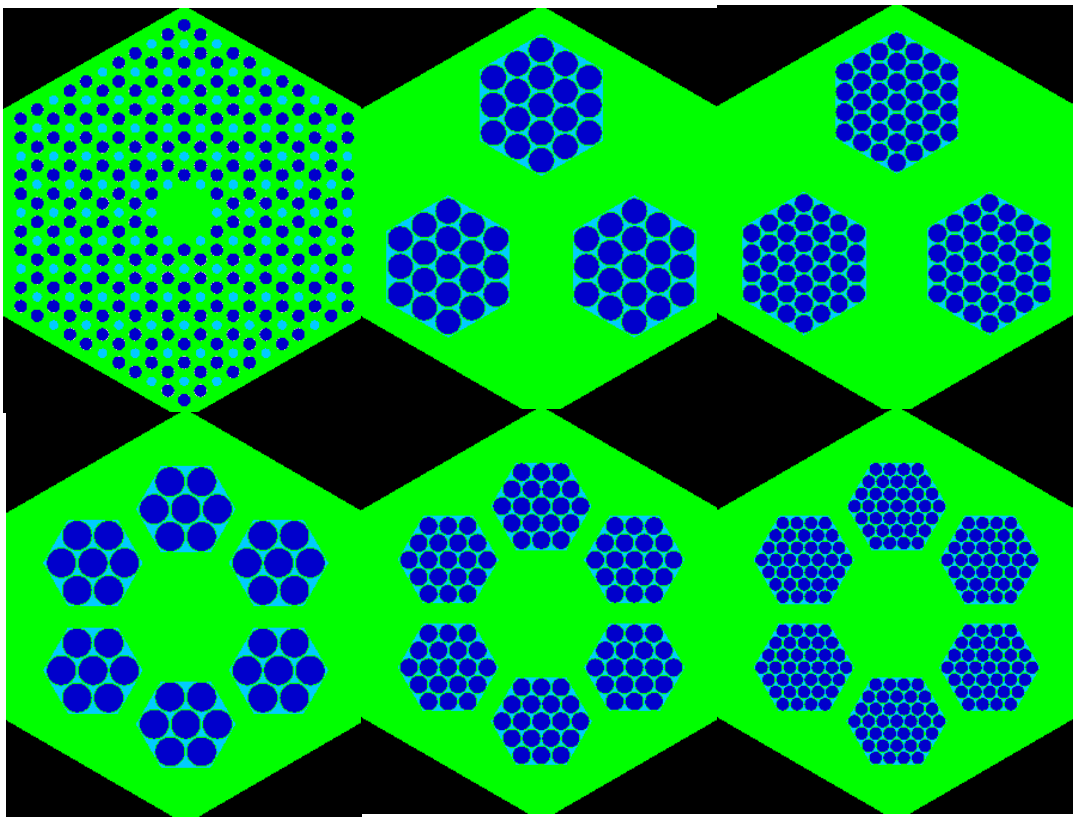
The AHTR uses the same basic fuel type as the Next-Generation Nuclear Plant (NGNP) and has fuel enrichments and power densities similar to those of high-temperature gas-cooled reactors. The basic design follows the NGNP point design [7] with a reduced coolant hole diameter. The components of the liquid salt are chosen to have very small neutron absorption cross sections. As a consequence, the reactor physics are very similar to those of gas-cooled reactors. In fact, the core of a fully voided AHTR reactor is nearly identical to that of an operating gas-cooled reactor.

The basic design of the AHTR consists of 2650 graphite blocks with 216 fuel channels in each (Figs. 1 and 2). The handling of this many fuel blocks for a routine refueling outage is possible but challenging. Therefore, clustering groups of rods into an assembly-like design would allow strings of fuel to be bundled together and removed from the core as a single unit—similar to fuel reloading in present light water reactors.

Simple modifications to this basic block design were considered, with a parametric study to determine the effect on the CVR and k_{inf} of changes in the pitch and diameter of the fuel and coolant channels. In addition, various "clustered-rod" configurations were considered (shown in Fig. 3), in which a group of graphite-clad fuel pins are placed within large holes in the graphite. This would simplify refueling; the basic configuration would need 3–4 times as many fuel blocks as the NGNP because of the much higher design power and larger core design. In addition, less graphite would be wasted because the pure reflector blocks would not have to be replaced at each refueling. This also would increase the passive safety limits because the reactor would be cooled by natural circulation of the liquid salt, not conduction through the graphite blocks. Each modified block configuration considered has comparable fuel/coolant/graphite ratios as the basic configuration. The channels for the clustered rods are hexagonal to maintain the coolant volume fraction at 7% of the block.

The general results of the parametric study, which focused on the cluster design in the top center of Fig. 3, indicate that there was not a significant change in the CVR and k_{inf} with the clustered configuration when the fuel enrichment was graded in the pins within the cluster [8]. Note that these results differ from those in a study performed by AREVA NP, which used pellet fuel rather than particle fuel [9]. The use of pellet fuel resulted in significant spatial shielding within the cluster. However, the addition of the graphite moderator in the fuel compacts combined with the enrichment grading reduces this impact of the self shielding effects.

Figure 3: Basic (top left) and clustered-rod fuel block configurations of the AHTR



5. Conclusions and Future Direction

A baseline design was developed for the AHTR, and studies were performed to evaluate the neutronics impact of the salt selection and fuel block design. The results provide a comparison of the reactivity coefficients for various salts and confirm that LiF-BeF_2 is the preferred salt. Fuel block designs were considered and shown to have minimal impact on the reactivity coefficients and block k_{inf} with appropriate design considerations.

The current design has further evolved to a loop design with an internal primary tank within a pool [10]. This has an advantage in that the volume of the more expensive salts used for the primary system is greatly reduced. The AHTR design that started with a configuration similar to that of gas-cooled reactors is being optimized and developed to take advantage of the unique characteristics of liquid salts with enhanced heat removal capabilities. The current neutronics focus is to develop a preliminary assembly/clustered fuel design that will enhance operational performance, minimize refueling effort, and reduce graphite waste. The design activities are focused on the refinement of the clustered assembly designs to obtain a more optimal design. As a starting point, the configuration used by the advanced gas-cooled reactor (AGR) in the United Kingdom is being considered, since a cluster arrangement of fuel rods within a graphite block is used for the AGR [11]. In addition, the established fuel design for high-temperature reactors based on HTTR [12] is being considered.

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