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ADVANCED HIGH-TEMPERATURE REACTOR NEUTRONIC CORE DESIGN[1]

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ABSTRACT

The Advanced High-Temperature Reactor (AHTR) is a 3400 MW(t) fluoride-salt-cooled high-temperature class design concept intended to serve as a central generating station type power plant.

This paper focuses on preliminary neutronic design studies performed at Oak Ridge National Laboratory (ORNL) during fiscal year 2011. After a brief presentation of the AHTR design concept, the paper summarizes several neutronic studies. An optimization study for the AHTR core is first presented, and a design that reaches a single-batch fuel length of just over 2 years is chosen as reference. The temperature and void coefficients of reactivity are then analyzed for a few configurations of interest, and, in all cases analyzed, the isothermal temperature coefficient is proved to be negative. A discussion of the limiting factors due to the fast neutron fluence follows. The neutronic studies conclude with a discussion of the control and shutdown options. The studies presented confirm that sound neutronic alternatives exist for the design of the AHTR to maintain full passive safety features and reasonable operation conditions.

While significant technology development and demonstration remain, the basic design concept appears sound and tolerant of much of the remaining performance uncertainty. No fundamental impediments have been identified that would prevent widespread deployment of the concept.

Key Words: Fluoride-salt-cooled high-temperature reactor (FHR), Advanced High Temperature Reactor (AHTR), tristructural isotropic (TRISO) fuel.

1. INTRODUCTION

The Advanced High Temperature Reactor (AHTR) is a 3400 MWth fluoride-salt-cooled high-temperature reactor (FHR) design concept developed at Oak Ridge National Laboratory (ORNL). The overall goal of the AHTR development program is to demonstrate the technical feasibility of FHRs as low-cost, large-size power producers while maintaining full passive safety. The current AHTR core design [1] differs substantially from previous AHTR designs analyzed at ORNL [2]. Figure 1 shows an overall view of the AHTR core. From the neutronic standpoint, the main difference is the slab fuel element design used in the new AHTR concept. Table I contains the main characteristics of the AHTR core for the reference model.

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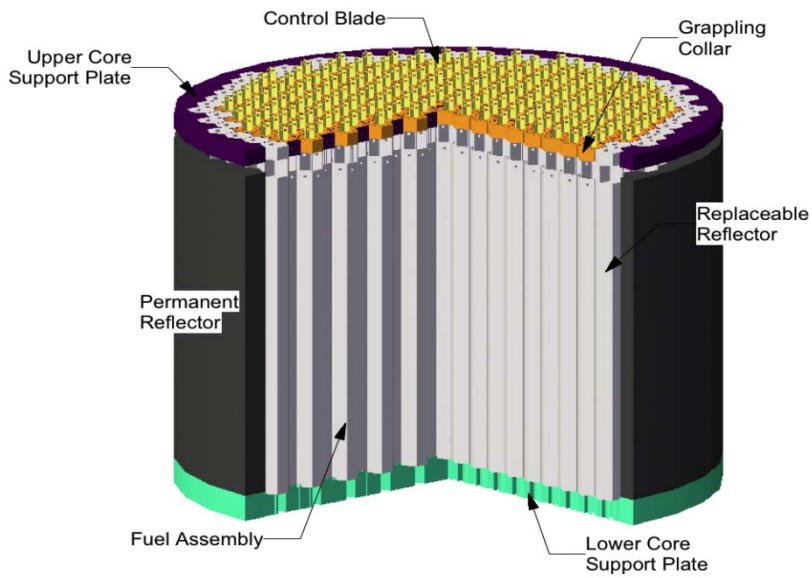

Figure 1. A 3-D view of the AHTR reference core.

Table I. Main core characteristics of the AHTR reference model

Parameter	Value	Units
Power (thermal)	3400	MW
Number of fuel assemblies	253[a]	-
Assembly lattice type	Hexagonal	-
Fuel type	Coated Particle	-
Moderator	Graphite	-
Reflector	Graphite	
Coolant	FLiBe	-
Core height (fueled region)	5.5	m
Core height (including axial reflector)	6.0	m
Equivalent core diameter (fueled region)	7.81	m
Core diameter (including radial reflector)	9.56	m
Average power per grain	41	mW/particle
Volumetric core power density	12.9	MW/m^3
Mass of heavy metal	32.91	MT
Fuel enrichment	19.75	^{235}U/U wt %
Mass of fissile material	6.5	MT
Fuel cycle length (once-through)	2.2	years
Inlet coolant temperature	650	°C
Outlet coolant temperature (average)	700	°C

[a]The number of fuel assemblies was decreased to 252 for multibatch refueling studies.

This paper describes several neutronic aspects of the AHTR core. The neutronic design includes consideration of thermal-hydraulic issues to achieve a more realistic design. Starting with the slab fuel design, the reference AHTR core design was obtained after a series of iterations involving thermal-hydraulic, mechanical, fuel, and structural limitations. Each of the design elements imposes its own constraints on the AHTR core. The baseline for the AHTR core design was to achieve a lifetime of 2 years for a core that uses 19.75 wt % or less enriched fuel. The 19.75 wt % enrichment is just under the uranium low-enrichment limit, and using it should provide the maximum expected fuel cycle length for the AHTR. The baseline core should not be interpreted as an optimized design, but as the model used to evaluate whether design changes represent improvement.

In Table I, the 32.9 metric tons of heavy metal (MTHM) correspond to the fresh initial core, as does the amount of fissile material. In the axial direction, the 5.5 m fueled region in the fuel assemblies is continued by 25 cm of carbonaceous material for additional moderation and reflection.

The FLiBe coolant flows from the bottom of the core to the top of the core. To allow criticality, the lithium component in FLiBe has to be highly enriched in ^{7}Li, as ^{6}Li has a high thermal neutron capture cross section. The reference enrichment that was used in these studies is 99.995 wt % ^{7}Li. During reactor operation, the isotopic fraction of ^{7}Li reaches an equilibrium value and, with an even higher starting enrichment, would be degraded back to the equilibrium value. The maximum fuel temperature calculated with a simple model is 854°C in the average fuel assembly (which would correspond to a radially flat power distribution) and a peak temperature of 1004°C in the hottest channel (calculated using a bare cylindrical reactor model), both well below the 1250°C limit that has been employed as the upper operational fuel temperature in gas-cooled reactors. The particular fuel configuration used in this design permits a raise in core power density of ~ 30%, compared to a prior AHTR design [2], to almost 13 MW/m^{3}.

2. DESCRIPTION OF FUEL ASSEMBLY

The fuel assembly is a 6 m tall hexagonal prismatic box with 1 cm thick walls made of carbon-carbon (C-C) composite. The outer apothem (half the side-to-side distance) of the hexagon is 22.5 cm. The interior ("channel") of the hexagonal box ("channel box") is divided into three symmetric regions by a 4 cm thick Y-shaped structure ("Y-shape"), also made of C-C composite. In each of the three regions, six equidistant fuel plates are supported at one end by the Y-shaped structure and at the other end by the channel box. The 0.7-cm distance between two fuel plates is filled by the FLiBe coolant ("intra-assembly coolant"). For the plates adjacent to the walls (channel box or Y-shape), the coolant thickness is half the thickness between plates: 0.35 cm. Table II summarizes the main geometric characteristics of the AHTR fuel assembly. Figure 2 shows a cross section through a fuel assembly inside the core.

The channel boxes of two adjacent assemblies are not in direct contact but are separated by 1.7 cm of coolant ("interassembly coolant") (Figure 2).

At the center of the Y-shape is the control blade slot, also Y-shaped. The slot is 1 cm thick, and each wing is 10 cm long.

Table II. Geometric characteristics of the fuel assembly for the reference AHTR design

Characteristic	Value	Units
Total height	600	cm
Fueled region height	550	cm
Fuel assembly pitch	46.75	cm
Outer apothem	22.5	cm
Channel box wall thickness	1	cm
Y-shape thickness	4	cm
Coolant thickness between plates	7	mm
Coolant thickness between plate and wall	3.5	mm
Control blade location thickness	1	cm
Control blade location wing length	10	cm
Fuel plate thickness	2.55	cm
Number of fuel plates	18	–

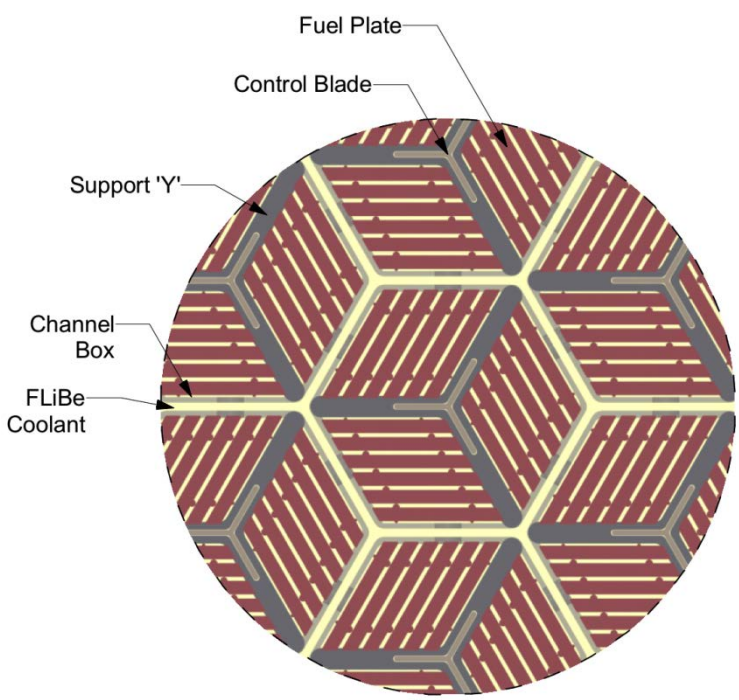

Figure 2. Transverse cross section of a group of fuel assemblies.

2.1. Fuel Plate

The plate design for the fuel elements represents a design evolution from the Small Modular Advanced High-Temperature Reactor (SmAHTR) design [3]. Within the slab fuel element, the tristructural isotropic (TRISO) fuel particles are distributed in two stripes separated by central carbonaceous material. This represents the main departure from the SmAHTR fuel, where the

fuel was distributed throughout the plate. The design takes into account both neutronic considerations, to allow better moderation, and thermal-hydraulic considerations, to allow better cooling. A thin (~1 mm thick) sleeve of carbonaceous material separates each fuel stripe from the FLiBe coolant, preventing individual fuel particles from eroding away.

Figure 3 shows a cross-sectional view of the fuel plank as modeled for the depletion calculations. The thickness of the fuel stripes is designed to allow maximum thermal-hydraulic performance (i.e., minimize the maximum temperature in the fuel) and to maximize the lifetime of the reactor core.

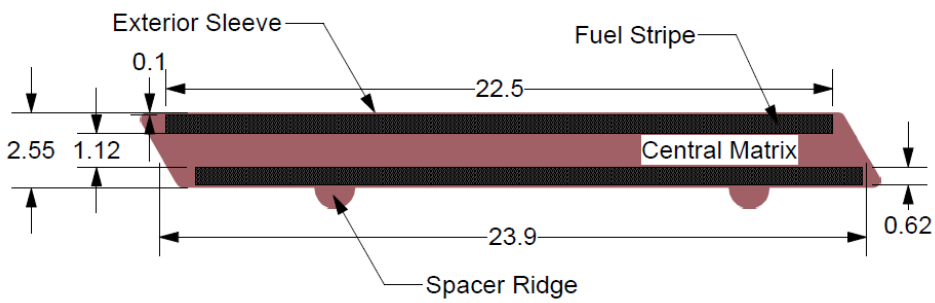

Figure 3. Transverse cross section of a fuel plate.

The prototype for the reference coated particle fuel grain used in this study was the AGR-2 fuel described in a recent paper [4]. The reference fuel differs from the prototype in that it uses a packing fraction of 40% and an enrichment of 19.75 wt % ^{235}U/U.

3. COMPUTATIONAL TOOLS

All calculations described in this paper used a prerelease version of the SCALE 6.1 software package developed at ORNL [5]. The neutron transport calculations were performed using the 3-D KENO code with both the continuous energy and the multigroup options. The continuous energy version was used mainly for evaluating the fresh AHTR core; the multigroup version was used as an engine in the depletion calculations of the TRITON module. The TRITON sequence executes a series of modules from the SCALE package, beginning with the cross-section processing modules (CRAWDAD, BONAMI, WORKER, CENTRM, and PMC) to create problem-dependent self-shielded multigroup cross-section libraries. The multigroup cross-section libraries thus created are then used in the KENO-VI module to generate a transport solution (fluxes and eigenvalue) for the system, which is post-processed by KMART. Finally, the fluxes from the transport calculations are used (after preparation by COUPLE) in the ORIGEN-S code to calculate the fuel depletion. These steps are repeated until the time evolution of the system is completed.

The depletion calculations used an approach based on the Dancoff factor to correctly account for the self-shielding effects [6]. Multibatch refueling schemes were developed and analyzed in a companion paper [7].

4. NEUTRONIC STUDIES RESULTS

4.1. Optimum Configuration Search

The purpose of this study was to determine the optimum fuel loading that maximizes the once-through fuel cycle length, with a target minimum of 2 years. For this purpose, the thickness of the central carbonaceous matrix of the fuel plate was varied while preserving the fuel-packing fraction and the geometry of the other elements of the fuel assembly. The fuel loading and the carbon-to-heavy metal (CHM) ratio thus were inherently varied to preserve the fuel plate external geometry.

Evaluations were performed with fuel plate central matrix thicknesses up to 20 mm (corresponding to CHM ratios between ~200 and 600) to assess once-through cycle lengths of the AHTR core. The results show a decrease in cycle length as the CHM ratio increases (Figure 4), combined with a sharp increase in the discharge burnup. The baseline reference AHTR core was chosen as the one that maximized the cycle length and corresponds to a core once-through cycle length of 2.17 years and a discharge burnup of ~80 GWd/MTHM.

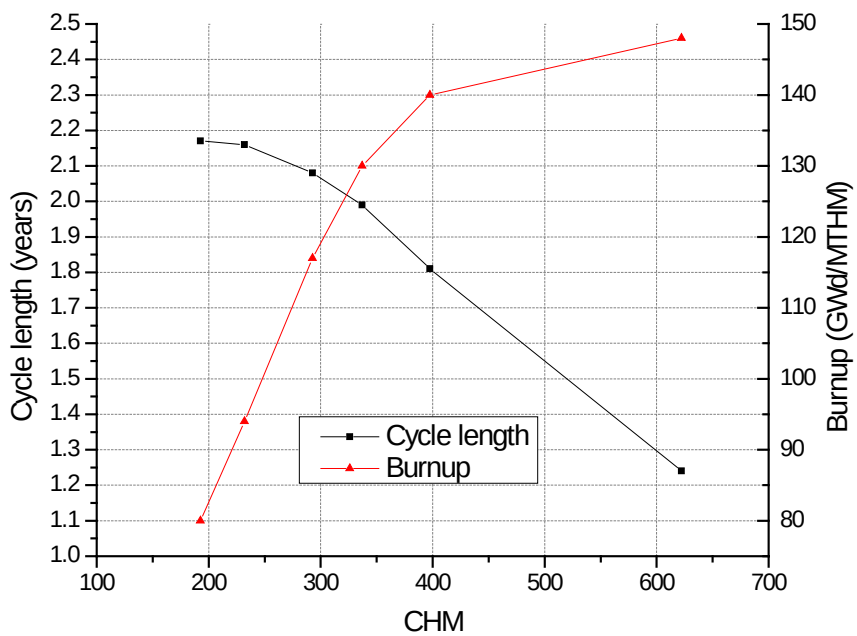

Figure 4. AHTR once-through cycle length and discharge burnup as a function of CHM.

4.2. Temperature Reactivity Coefficients

The isothermal temperature reactivity coefficient was evaluated for the reference configuration at the beginning of cycle (fresh fuel) over the temperature interval [800 K, 1850 K] (approximately [500°C, 1600°C]). The isothermal temperature coefficient of reactivity, $(\partial\rho/\partial T)_{iso}$, was calculated

by using the procedure described in Reference [8]. The following calculations do not account for the change in coolant density (or any other material) with temperature and therefore represent strictly the effect on reactivity of changes in cross sections due to changes in temperature. However, the coolant density variation with temperature can be combined with the calculated results to give the total effect (see Section 4.3 for the evaluation of the total effect, $(d\rho/dT)_{iso}$).

For the fresh reference core, the average isothermal temperature coefficient over this interval is $(\partial\rho/\partial T)_{iso} = -3.80$ pcm/K. The isothermal coefficient is not constant but varies over the above interval and can be better approximated[3] with a linear function:

$$\left(\frac{\partial\rho}{\partial T}\right)_{iso}(pcm/K) = -[5.78 \pm 0.20] + [(0.75 \pm 0.07) \times 10^{-3}]T(K)\ . \qquad (1)$$

Fuel burnup has a beneficial effect on the isothermal temperature coefficient. For the reference core at the end of cycle, the average isothermal temperature coefficient over the above temperature interval is $(\partial\rho/\partial T)_{iso} = -7.53$ pcm/K and as a linear function it can be approximated with

$$\left(\frac{\partial\rho}{\partial T}\right)_{iso}(pcm/K) = -[2.95 \pm 0.24] - [(1.74 \pm 0.09) \times 10^{-3}]T(K)\ . \qquad (2)$$

In this case, the coefficient becomes more negative with temperature increase, a desirable feature.

The reference core corresponds to a CHM ratio of just below 200. If the CHM increases, the average isothermal temperature coefficient becomes less negative. For a CHM of 337, the coefficient has an average value of –2.92 pcm/K over the same temperature interval with the linear approximation

$$\left(\frac{\partial\rho}{\partial T}\right)_{iso}(pcm/K) = -[4.84 \pm 0.14] + [(0.73 \pm 0.05) \times 10^{-3}]T(K)\ . \qquad (3)$$

Figure 5 shows the variation of the isothermal reactivity coefficient with temperature for the three cases discussed above.

All the calculations of the temperature coefficient of reactivity were performed for an uncontrolled configuration of the AHTR core (i.e., no control rods, no burnable poison).

[3] The uncertainties in the coefficients of the linear approximations are purely statistical uncertainties and do not account for any approximation due to the physics used in the codes.

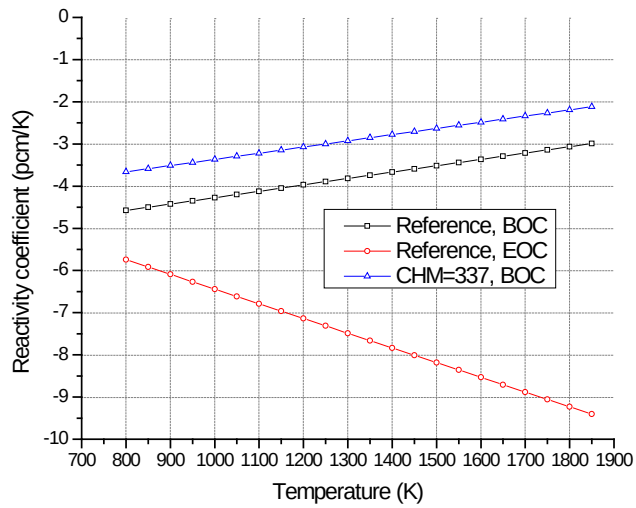

Figure 5. Variation of isothermal reactivity coefficient with temperature.

4.3. Void Reactivity Coefficients

The effect of voiding the AHTR core was studied for the reference fresh core by calculating the insertion of reactivity that would result from complete removal of the coolant (100% void fraction). The negative reactivity thus inserted is –2056.3 ± 26.3 pcm, or about –2% Δk/k. This results in an average void reactivity coefficient of

$$(\partial\rho/\partial\alpha) = -20.56 \pm 2.63 \text{ pcm/\% void} . \quad (4)$$

The coolant density can vary with its temperature as a linear function [2]:

$$\rho_c = 2.280 - 0.000488T(°\text{C}) = 2.413 - 0.000488T(K). \quad (5)$$

Combining Eq. (5) with the above result for the average void reactivity coefficient gives an average value of $(\partial\rho/\partial T)_c = -0.51$ pcm/K for the rate of reactivity insertion due to the voiding of coolant with temperature increase. The isothermal and void coefficients of reactivity can be combined, yielding the combined effect of $(d\rho/dT)_{iso} = -4.3$ pcm/K over the temperature interval [800 K, 1850 K].

The coolant void reactivity coefficient can vary drastically with the fuel design and can become positive for high CHM ratio. This is shown in Figure 6, where the calculated void coefficient values for several CHM ratios were fitted with a straight line. The calculations were performed by varying the mass of fuel (fuel stripe thickness) in the fuel plate. For CHM ratios beyond ~450, the void coefficient of the coolant becomes positive. The coolant void coefficient for the reference model corresponds to a CHM value just below 200.

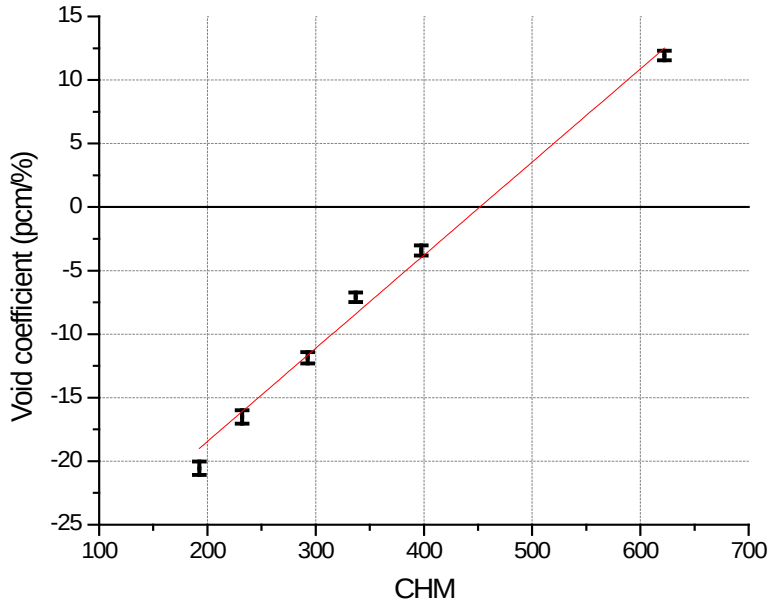

Figure 6. Coolant void coefficient as a function of carbon–to–heavy metal ratio.

The coolant void coefficient also varies substantially during fuel irradiation as a result of fission product buildup and change in actinide composition. Calculations were performed for the reference model for the beginning of cycle (BOC) and for the end of cycle (EOC) (once-through) compositions. The results and the associated statistical uncertainties are shown in Table III. The void coefficient becomes more negative and almost doubles in magnitude by the end of the fuel cycle.

Table III. Coolant void coefficients for the reference core

AHTR core	$(\partial\rho/\partial\alpha)$ (pcm/%)	σ (pcm/%)
BOC	–20.56	0.26
EOC	–38.70	1.22

4.4. Fast Fluence

The fast fluence distribution in the core determines the life of the carbonaceous materials that are part of the core and nearby structures, including the C-C composite core barrel. The average 60 year fast fluence at the core barrel is 5.7×10^{18} n/cm^2, and the total fluence is 4.8×10^{21} n/cm^2, mostly due to thermal neutrons.

The mechanical distortion of C-C composites and coated particle fuel compacts due to neutron irradiation is not well documented at present. Figure 7 shows the behavior of the graphite distortion at different temperatures and fast fluences (from Reference [9]). For typical AHTR operating temperatures, the distortion of graphite is below 2% for fast fluences up to 4×10^{22} n/cm^2.

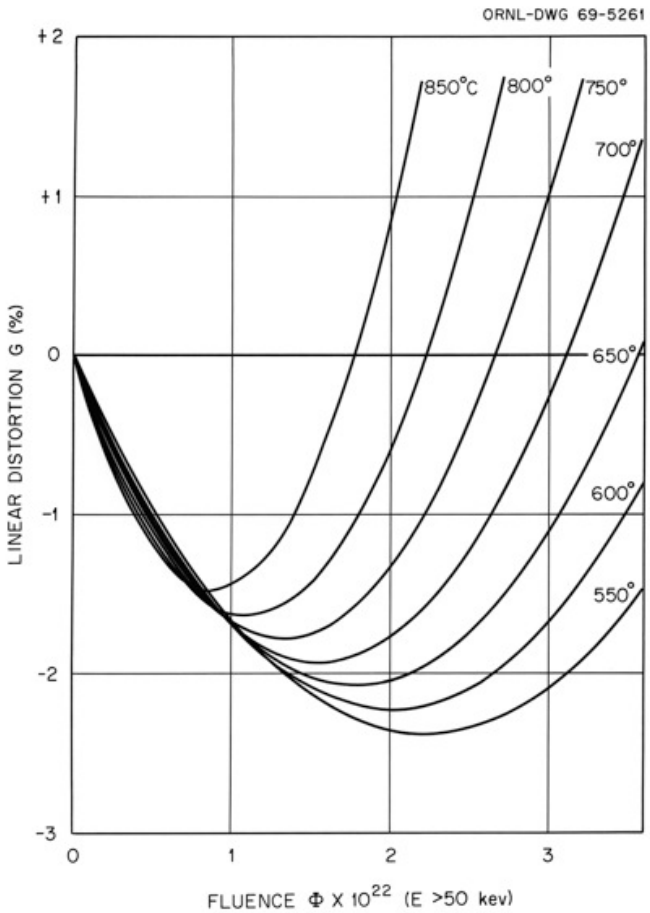

Figure 7. Graphite linear distortion as a function of fluence at various temperatures.

The distribution of fast neutron flux (E > 50 keV) in the reactor core is shown in Figure 8 for the reference AHTR core. At the maximum fast flux of 0.19×10^{22} n/cm^2/year for the central region, the graphite needs ~20 years of in-core irradiation before reaching a fast fluence of 4×10^{22} n/cm^2.

Although 20 years seems to provide adequate margin for the residence time of a fuel assembly in the core, the design margin can be substantially reduced for lower enrichment designs. The reason is that the neutron flux varies with the inverse of the enrichment fraction for the same core configuration to provide the same level of power. The design margin is further reduced with

fuel burnup, because the net effect of the burnup is a decrease in the fissile fraction (and hence the enrichment) of the fuel. Assuming a 10% initial enrichment for fresh fuel and an average of 7% fissions per initial metal atom for equilibrium fuel (in case a multibatch refueling scheme is adopted), this results in a threefold increase in the fast fluence rate (fast neutron flux) and a corresponding decrease to an approximately 7 year lifetime for core graphite components and correspondingly lower time (~ 3 years) until the graphite starts to swell.

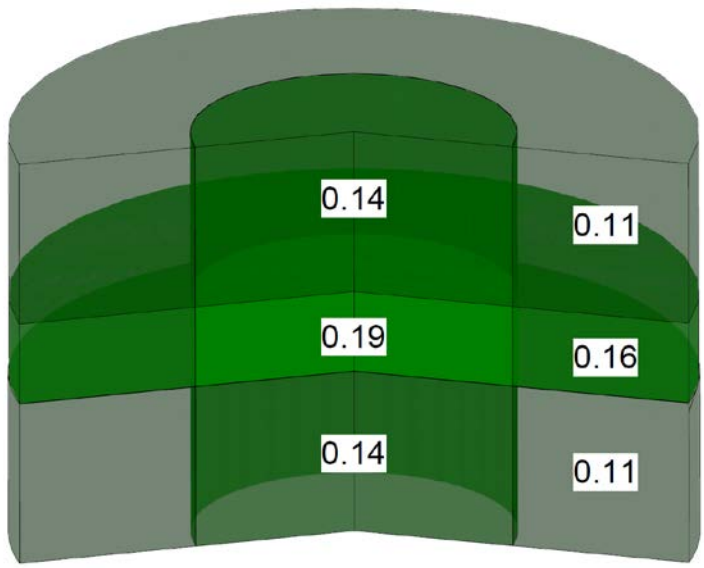

Figure 8. Regionwise fast flux (E > 50 keV) in the reference core at BOC ($\times 10^{-22}$ n/cm^2/year).

A recent study [10] performed for the Next Generation Nuclear Plant program includes a series of measurements on the deformation of C-C materials, but more radiation performance data are needed for the carbonaceous materials intended for use in the AHTR core.

The average total fluence rate over the height of the reactor vessel is calculated for the fresh reference core to be 2.0×10^{18} n/cm^2/year, which yields a total fluence of 1.2×10^{20} n/cm^2 over an anticipated lifespan of 60 years. At the midplane of the reactor, the total fluence can reach 4.8×10^{20} n/cm^2 over a 60 year period. In both cases, the main contributor to the fluence is the thermal component (E < 0.625 eV). High-nickel alloys, such as Alloy N, embrittle presumably due to helium accumulation at the grain boundaries when subjected to high neutron fluence (>~10^{20} n/cm^2) at high temperatures (> 500°C) [11]. The embrittlement has both thermal and energetic neutron pathways. The lifetime neutron fluence is sufficiently high that some shielding is needed to protect the vessel. Incorporation of a thin boron carbide layer just inside the core barrel is a possible shield configuration.

4.5. Control Blade Design

The AHTR employs one control blade per fuel assembly, with each control blade having relatively low reactivity worth. The control blade remains with the fuel assembly throughout the

fuel loading and unloading procedure to preclude criticality accidents during fuel handling. The current AHTR design employs a molybdenum–hafnium carbide alloy (MHC) as its neutron absorber and structural material. The control blade may be removed, and possibly reused, following fuel transfer into the poisoned, spent fuel storage pool after a decay period of ~18 months. Figure 9 shows the evolution of both the total activity and the total decay power in a control blade as a fraction of their value after blade's removal from the core as well as the contributions of ^{181}Hf to the quantities. The dependence is shown as a function of time after the blade's removal from the core. The blade activity calculation was performed by simulating the irradiation of a control blade for a long time (on the order of the fuel assembly's residence in the core), which is not typical for the intended normal operation of the control blades. During operation, most of the control blades would be fully withdrawn from the core, so the result is conservatively high.

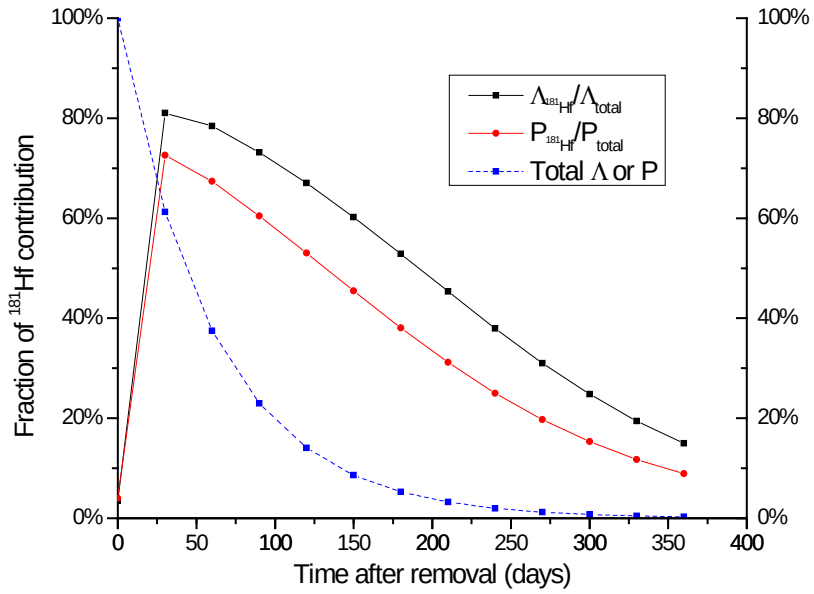

Figure 9. Evolution of the ^{181}Hf contributions to the total control blade activity and decay power after removal from core and the fraction of initial value for the total activity (Λ) or power (P) after removal.

MHC is a commercial, particle-strengthened molybdenum-based alloy with 1.2 wt % hafnium and 0.1 wt % carbon. MHC has a density of 10.28 g/cm^3. Neutron absorption occurs in both the hafnium and the molybdenum. The blade is 8 mm thick and configured in a Y-shape. A normally incident thermal neutron has a 15% absorption probability through the blade.

The configuration of the control blades allows for further flexibility in meeting reactivity requirements. The blades can be made either thicker or wider to provide increased reactivity control with little additional design impact.

The control blade is designed to be gripped by a control blade leader using the grappling holes located in its upper end. Figure 10 shows the control blade inserted into the top of a fuel assembly. The control blade is designed to sink into a full-length slot within the C-C composite

support Y upon release, displacing primary coolant. The control blade has a 1 mm clearance on all sides to its guide slot. Design rules remain to be established for the mechanical distortion of C-C composites under core level irradiation to confirm whether this is an adequate mechanical clearance to avoid control blade jamming.

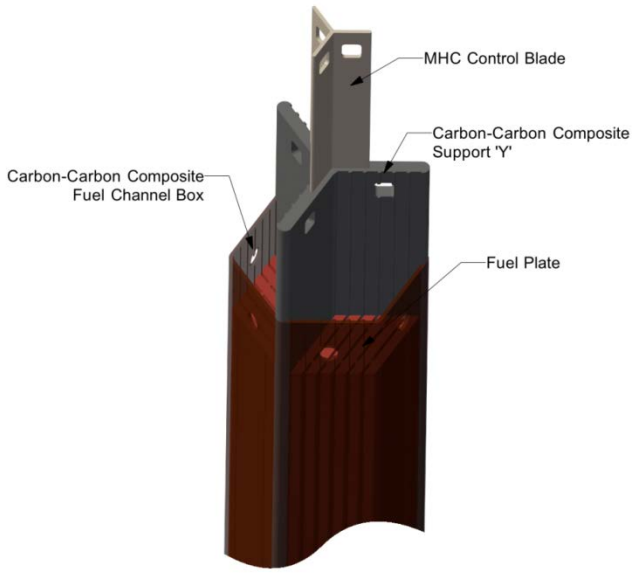

Figure 10. Upper end of the fuel assembly with MHC control blade inserted.

The control blades provide a total negative reactivity of ~10% Δk/k when all the blades are fully inserted. This reactivity combined with the required shutdown margin determines the maximum allowed excess reactivity for the AHTR core. The limited negative reactivity available from the control blades necessitates the use of burnable poisons in the AHTR core to reduce the amount of excess reactivity.

4.6. Secondary Shutdown System

There are several possible techniques for inserting an adequate amount of negative reactivity into the core to shut down the chain reaction. However, because deploying the secondary shutdown mechanism implies that the primary control rod insertion shutdown mechanism has failed, a technique that does not involve insertion of a mechanical component into the core would provide greater diversity.

The secondary shutdown mechanism selected for the AHTR is the injection of a rare earth fluoride neutron poison (EuF_3 and/or GdF_3) into the primary coolant. The poison salt injection mechanism design is currently immature. Both EuF_3 and GdF_3 have melting points well above the operating temperature of an FHR. However, their Gibbs energy lies between that of LiF and BeF_2, indicating that the poison salts will mix with the primary coolant fluorides and not plate out onto the container surfaces. Further, both EuF_3 and GdF_3 have low-temperature (<700°C) eutectics with LiF, indicating that both will dissolve readily into the primary salt.

One potential mechanism for injecting rare earth poison fluoride into the primary coolant would be to position a cartridge of poison salt within the primary coolant. A nitrogen accumulator would be configured to drive a piston forcing the poison salt powder into the primary coolant once the cartridge is triggered. An Au-Ni braze with a specific melting point (slightly above operating temperature) could be used to seal the cartridge. Heater wiring near the braze joint would enable operator triggering of the cartridge. Figure 11 shows a potential poison salt cartridge configuration.

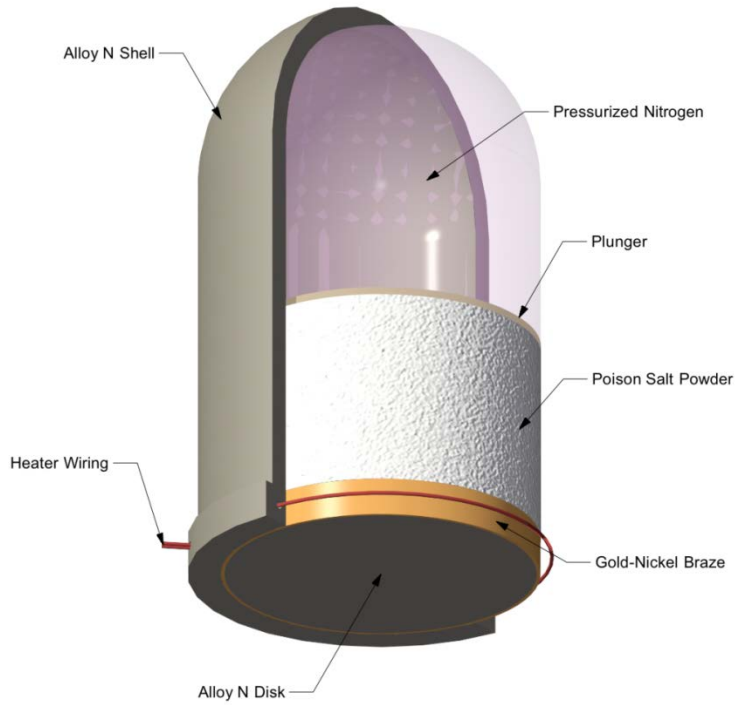

Figure 11. Potential secondary shutdown poison salt cartridge configuration.

The poison salt must be removed from the primary coolant following conclusion of an accident. The molten salt reactor program developed and demonstrated technology for removing rare earth fluorides (including EuF_3) from FLiBe [12]. In the rare earth fluoride separation process, the poisoned salt is countercurrently contacted with liquid bismuth containing a reductant such as lithium. Bismuth is essentially immiscible with the salt. The reductive extraction reaction between EuF_3 in salt and metal phases can be represented by the equation

$$\mathrm{EuF_3} + 3\mathrm{Li(Bi)} \leftrightarrows \mathrm{Eu(Bi)} + 3\mathrm{LiF}\ . \qquad (6)$$

The rare earth metal is subsequently removed from bismuth by contacting it with LiCl. This technique has been used in demonstrations where separation factors of 10^4 have been reached for the trivalent rare earths dissolved in FLiBe [13].

The negative reactivity insertion provided by the EuF_3 varies linearly with the europium concentration in the coolant salt, as shown in Figure 12. The concentration of europium is given as a weight fraction and can be read as grams of europium per tons of mixture (FLiBe+EuF_3). The plot shows, for example, that half a kilogram of europium in EuF_3 compound per metric ton of FLiBe provides a negative reactivity insertion similar to that provided by the insertion of all the control rods (~ –11% Δk/k).

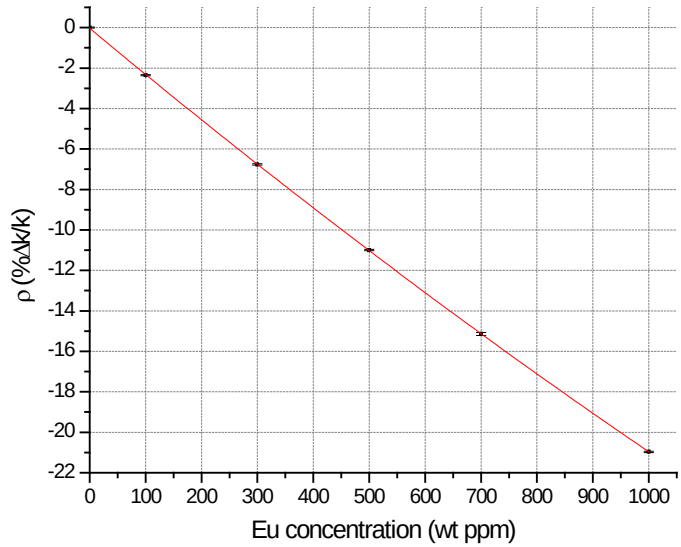

Figure 12. Reactivity insertion as a function of Eu concentration.

The computed points can be fitted with a quadratic function over the europium concentration interval [0, 1000] ppm:

$$\rho[\%\Delta k/k] = -[3.70 \pm 2.95] \times 10^{-2} - [2.30 \pm 0.02] \times 10^{-2} \times Eu + [2.10 \pm 0.17] \times 10^{-6} \times Eu^2 , \quad (7)$$

where *Eu* is the europium concentration in parts per million. The quadratic term has a small coefficient and can only have an impact for large europium concentrations.

The calculations assumed uniform dispersion of EuF_3 in the FLiBe salt. With the roughly 2310 tons of FLiBe in the primary circuit, ~1150 kg of EuF_3 must be added to the primary coolant by the poison salt injectors to roughly equal the reactivity worth of complete control blade insertion.

4.7. Burnable Poison

To compensate for the large excess reactivity of the initial bare core, burnable poison particles will be incorporated into the central matrix material of each fuel plate. Europium was selected as the burnable poison material, because it burns out at nearly the same rate as the fuel, minimizing

the requirement for additional control blades. Particle-based neutron poison was selected so that the poison would be retained in the fuel during fabrication, thus eliminating the poisoning of the primary coolant that could result in a positive void coefficient. The high-temperature impurity burnout phase of the coated particle compact manufacturing would tend to burn out bulk impurities in the carbon matrix material. In the present design, the poison kernels have a diameter of 500 μm. A uniform load of ~ 50 g of europium per fuel plate will lower the initial reactivity of the AHTR core below 10% Δk/k and maintain it below that limit for the life of the once-through cycle. (The calculations were performed for a core with a design slightly different from the reference design. We expect that the load will remain close for the reference core.) Control rods are used to control the remainder of excess reactivity over the fuel cycle. The impacts of burnable poison on the cycle length and the fuel temperature reactivity coefficient remain to be studied.

5. SUMMARY AND CONCLUSIONS

This paper focuses on the preliminary neutronic design studies of the AHTR performed at ORNL during fiscal year 2011. The AHTR is a 3400 MW(t) FHR design concept intended to serve as a central generating station type power plant.

A core design with a single-batch cycle length over 2 years and a discharge burnup of ~80 GWd/MTU was chosen as a reference. The reference design is shown to have negative temperature and void coefficients during the entire single-batch fuel cycle. Our studies show that core designs with more economically attractive CHM values up to 400 can still have negative reactivity coefficients. Control and shutdown safety systems are discussed, and preliminary quantitative results are given. For long in-core fuel residence times, the fast fluence can be a limiting factor in the design, and more accurate fuel performance data are needed for precise evaluation of designs that depart from the current reference.

These studies confirm that viable AHTR neutronic design alternatives exist that maintain full passive safety features and reasonable operation conditions. Although significant technology remains to be developed and demonstrated, the basic design concept appears sound and tolerant of much of the remaining performance uncertainty. No fundamental impediments have been identified that would prevent deployment of the concept.

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