Overview and Status of the Advanced High-Temperature Reactor

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Abstract – *A new reactor concept, designated the Advanced High-Temperature Reactor (AHTR), is being developed that uses liquid fluoride salt as a coolant, graphite moderator and high-temperature coated-particle fuel. The concept is being supported by the U.S. Department of Energy as part of the Generation IV program as a "coolant variant" of the Very High-Temperature Reactor because it shares many of the same fuel, moderator and material technologies. The purpose of the AHTR is to provide an advanced design that is sufficiently robust to allow a growth path to higher power output and higher temperatures, and also offering the potential for highly competitive economics. Although it creates some unique technology challenges of its own, the AHTR has many strong advantages, such as: lower reactor fuel temperatures, low-pressure reactor vessel and piping, enhanced safety features, and improved economics. Several analyses have been performed during the past two years to demonstrate the physics viability of the concept and to support the development of a preconceptual design. The evolution of the concept is presented, along with a description of the present design and a summary of key performance analyses.*

I. INTRODUCTION

A major research program within the U.S. Department of Energy (DOE) is the Generation IV Program, which seeks to develop advanced reactor systems with higher performance levels than current nuclear power plants in terms of safety, economics, proliferation resistance and sustainability. A particular focus of the Generation IV Program is the economical production of hydrogen that will be needed in the near term for conversion of heavy oils to liquid fuels and in the longer term for hydrogen-based transportation fuels. The production of hydrogen 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 the 30–35% efficiency of current nuclear plants. For this reason, the emphasis of the Generation IV Program is the development of the Very High-Temperature Reactor (VHTR),[1] which is expected to have an outlet coolant temperature exceeding 900°C.

Historically, helium has been used as the coolant of choice for very high-temperature reactors to avoid two-phase phenomena introduced by boiling liquids. An alternative option is to use a mixture of molten fluoride salts, which can have a boiling temperature in excess of 1400°C. The superior heat transfer, heat capacity and heat transport characteristics of liquids compared with gases enable delivery of high-temperature heat at a near uniform temperature with lower reactor fuel temperature. A new reactor concept, designated the Advanced High-Temperature Reactor (AHTR), is being developed that uses liquid fluoride salt as a coolant in a graphite-moderated reactor using high-temperature coated particle fuel. The concept grew out of an internal study by Oak Ridge National Laboratory, Sandia National Laboratories, and the University of California at Berkeley.[2] The research consortium, now funded by the DOE, has grown to include Idaho National Laboratory, Argonne National Laboratory and Framatome-ANP, with related research being conducted at several universities. Because there is a large overlap of technologies between the AHTR and the flagship VHTR, the AHTR is frequently referred to within the Generation IV Program as the liquid-salt-cooled VHTR, or LS-VHTR.

The AHTR combines four established technologies in a new way: (1) coated-particle graphite-matrix fuels successfully used in helium-cooled reactors, (2) passive safety systems and plant designs previously developed for liquid-metal-cooled fast reactors, (3) low-pressure liquid-salt coolants studied extensively for use in liquid-fueled reactors, and (4) high-temperature Brayton power cycles. The new combination of technologies enables the design of a high-power [2400 to 4000 MW(t)], high-temperature (850 to 950ºC) reactor with fully passive safety capability and the economic production of electricity or hydrogen.

Although the primary novelty of the AHTR is the use of liquid-salt coolant, its technical basis is derived from billion-dollar programs in the 1950s and 1960s that developed technologies for the use of liquid salts in nuclear systems. Two experimental reactors were built and successfully operated. The Aircraft Reactor Experiment was a 2.5-MW(t) reactor that was operated in 1954 at a peak temperature of 860ºC. This was followed in 1965 by the Molten Salt Reactor Experiment (MSRE), an 8-MW(t) reactor that demonstrated most of the key technologies for a power reactor. In addition, test loops with liquid salts were operated for hundreds of thousands of hours, and materials of construction were code qualified to 750ºC.[3] Unlike these earlier molten salt reactors (MSR), which circulated the fuel within the coolant, the AHTR uses the same solid fuel as the gas-cooled VHTR and a clean liquid salt as a coolant. However, the earlier MSR programs demonstrated several critical technologies for the AHTR, and the results are documented in more than 1000 technical reports.

This paper provides an overview of the baseline AHTR concept and recent analyses that have been performed to develop the concept. Several more detailed papers by program participants are included in the same proceedings, including the assessment of candidate salt compositions, decay heat removal options, physics and thermal-hydraulics analyses, and a lower temperature variant that contains metallic reactor internals.

II. CURRENT AHTR CONCEPT

The AHTR uses coated-particle graphite-matrix fuel and a liquid-fluoride-salt coolant. The fuel is the same type that is being developed for the VHTR, which is similar to the fuel that was successfully used in high-temperature gas-cooled reactors such as Peach Bottom, Fort St. Vrain, the Arbeitsgemeinshaft Versuchsreaktor (AVR), and the Thorium High-Temperature Reactor (THTR). This type of fuel can be subjected to fuel-failure temperatures in excess of 1600°C without damage.

The optically transparent liquid-salt coolant is a mixture of fluoride salts with freezing points typically between 350 and 500°C and atmospheric boiling points as high as ~1400°C. Selection of optimal salt for the primary coolant or the secondary heat transport loop involves the assimilation of several considerations, including thermo-physical properties, material compatibilities, nuclear performance, toxicity, and cost. In the initial viability study,[4] it was observed that the neutronic performance of the core, especially the reactivity response of the core to voiding of the liquid coolant, was highly dependent on the isotopic and elemental composition of the salt. A subsequent study was performed to evaluate and compare thermo-physical properties and chemical behaviors of several candidate salts for this application.[5, 6] For the purposes of establishing a baseline concept, $^{7}Li_2BeF_4$, referred to as "Flibe" was chosen as the reference salt because it was used in previous nuclear applications and was also judged to be the most neutronically favorable of the candidate salts.

The reactor operates at near-atmospheric pressure and, at operating conditions, the liquid-salt heat-transfer properties are similar to those of water. Heat is transferred from the reactor core by the primary liquid-salt coolant to an intermediate heat-transfer loop. The intermediate heat-transfer loop uses a secondary liquid-salt coolant to move the heat to a thermochemical hydrogen (H_2) production facility or to a turbine hall to produce electricity. If electricity is produced, a multireheat nitrogen or helium Brayton power cycle (with or without bottoming steam cycle) is used. Figure 1 is a schematic of the plant system.

The reactor layout for the current AHTR concept is shown in Figure 2. The AHTR uses a 9-m-diameter reactor vessel and a passive reactor vessel auxiliary cooling system (RVACS) similar to that developed for decay heat removal in General Electric's sodium-cooled S-PRISM design.[7] The reactor decay heat is (1) transferred from the reactor core to the reactor vessel graphite reflector by natural circulation of the liquid salts, (2) conducted through the graphite reflector and reactor vessel wall, (3) transferred across an argon gap by radiation to a guard vessel, (4) conducted through the guard vessel, and then (5) removed from outside of the guard vessel by natural circulation of ambient air. There are no pumps, valves, or other active systems necessary for successful decay heat removal. Note also that the low pressure reactor vessel is enclosed by a guard vessel so that any coolant leaks from the reactor vessel will be

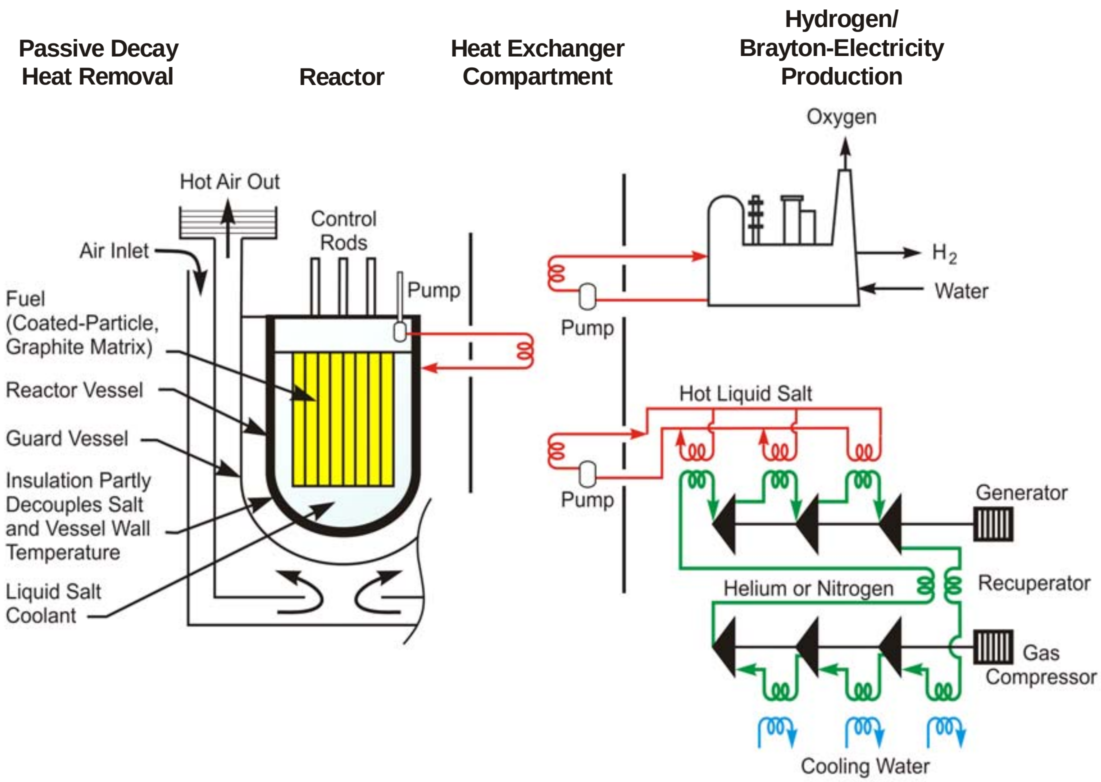

Fig. 1. Schematic of the AHTR for cogeneration of electricity and hydrogen.

contained by the guard vessel, making a loss-of-coolant accident nearly impossible. Therefore, the design basis accident for this reactor is a loss of flow accident.

Initially, the reactor core design was very similar to that of the gas-cooled VHTR reactor: the same coated particle fuel, cylindrical compacts, and hexagonal graphite moderator blocks; similar fuel enrichment and compact loading; similar power density; and a similar annular core shape. Subsequent physics analyses indicated that the annular VHTR-like core design, although optimized for gas coolant, was far from optimal for liquid coolant.[4] In particular, the inner graphite reflector is present in a gas-cooled reactor to improve heat transfer from the core, which is primarily by the process of conduction through the graphite, during a loss of forced circulation (LOFC) accident. A LOFC simulation for the AHTR showed that significant natural circulation of the liquid-salt coolant occurs during the transient and provides 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.

The change also eliminates the problem of severe power peaking near the inner reflector-core interface as observed in the gas-cooled VHTR. It is for the same reason that the AHTR can be designed with much higher thermal output than a gas-cooled system, i.e., it is not limited by conduction-driven decay heat removal during an accident. Figure 3 compares the original AHTR core and reflector layout with the current baseline design. The individual hexagonal moderator and fuel blocks continue to be identical to those used by the VHTR, although several alternative designs are being considered, 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. Table 1 lists the current design features and baseline assumptions used by all organizations participating in this study.

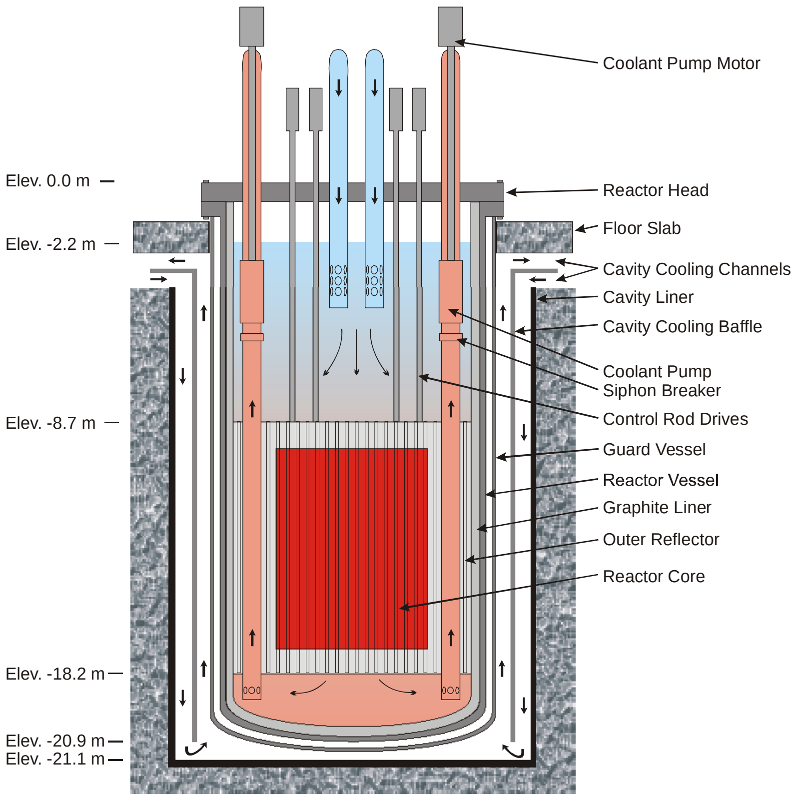

Fig. 2. Elevation view of baseline AHTR reactor.

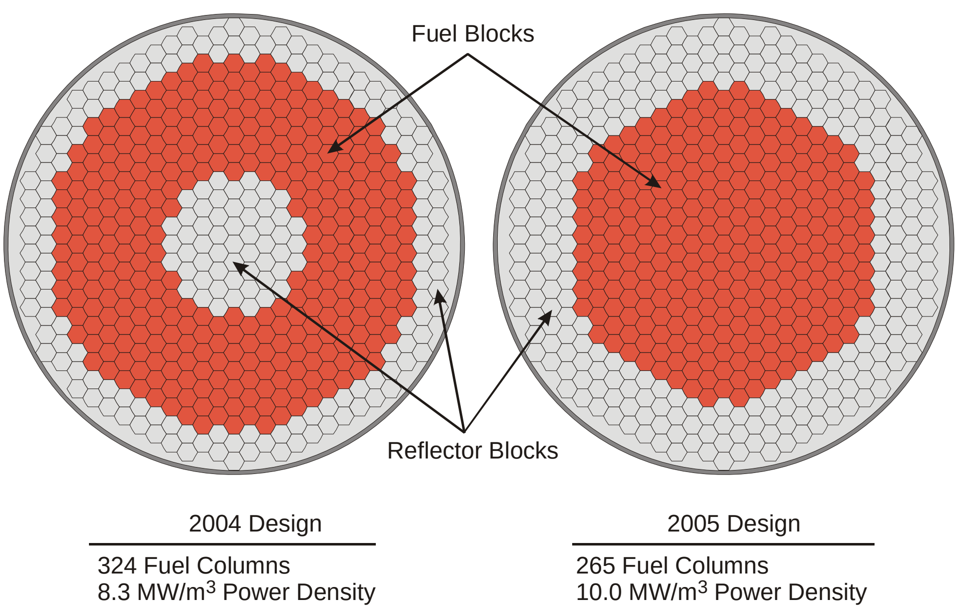

Fig. 3 AHTR core and reflector pattern from original (left) and current (right designs).

TABLE 1
Key Parameters for Baseline AHTR Concept

Parameter	Value
Coolant salt	$2LiF-BeF_2$
Li-7 isotopic concentration	99.995%
Outlet coolant temperature	950 °C
Inlet coolant temperature	850°C
Total thermal power	2400 MWt
Reactor vessel diameter	9.2 m
Fuel kernel composition	$U_{1.0}C_{0.5}O_{1.5}$
Fuel kernel diameter	425 μm
Particle diameter	845 μm
U-235 enrichment	15%
Particle packing fraction	25%
Fuel cycle length	18 months
Discharge burnup	156 GWd/t
Fuel element:	
-Graphite density	1.74 g/cm^3
-Diameter (across flats)	36.0 cm
-Height	79.3 cm
-Fuel channel diameter	1.27 cm
-No. of fuel channels	216
-Coolant channel diameter	1.4 cm
-No. of coolant channels	108
-Pitch between channels	1.88 cm
Power density	10.0 MW/m^3
No. of fuel columns	265
No. of fuel blocks per column	10

III. PHYSICS AND SAFETY PERFORMANCE

Parametric studies were performed for the AHTR in order to determine the fuel enrichment needed to achieve the target cycle length of 18 months and target discharge burnup of greater than 100 GWd/t. The overall fuel cycle behavior of the AHTR is similar to the helium-cooled VHTR: the cycle length increases with uranium enrichment and packing fraction, and the optimum packing fraction is approximately 25%. Additionally, the discharge burnup increases with an increase in the number of batches, but the cycle length decreases.

The results of the fuel cycle study are summarized in Table 2. The three-batch scheme appears to require an enrichment that is slightly greater than the 20% limit that is imposed by proliferation concerns; however, it is likely that further optimization of the core could result in acceptable enrichment for even the three-batch case. The required enrichment is smallest for the one-batch case, but its discharge burnup is smaller than the target value.

Therefore, the two-batch scheme was selected to satisfy simultaneously the target cycle length and discharge. For the purpose of comparison, the results for a two-batch helium-cooled VHTR core having a power level of 600 MW(t) are included in the table. It is important to notice in Table 2 that the baseline AHTR core yields a 50% higher burnup than the VHTR with a similar fuel enrichment. This is due to the better neutron economy in the large cylindrical AHTR core resulting from reduced neutron leakage. This translates to 30% lower fuel cycle costs and reduced waste volume relative to the gas-cooled VHTR.

TABLE 2
Comparison of fuel cycle parameters for VHTR and AHTR

Parameter		VHTR	AHTR
Power, MW(t)		600	2400
Total number of fuel columns		102	265
Power density, MW/m^3		6.6	10.2
Specific power density, MW/t		103	158
Single-batch	Enrichment, %		10.4
	Burnup, GWd/t		78
Two-batch	Enrichment, %	14.0	15.3
	Burnup, GWd/t	100	156
Three-batch	Enrichment, %		20.6
	Burnup, GWd/t		234

A concern raised, early in the development of the AHTR, was its response to voiding of the liquid-salt coolant. While there is virtually no impact of voiding the helium in a helium-cooled reactor, the concern was that the AHTR may be plagued by the same positive reactivity feedback effects observed in earlier liquid-sodium-cooled reactors. Considerable analyses have been performed to characterize the reactivity response of the AHTR to coolant voiding; including coolant isotopics, fuel temperature and enrichment, use of burnable poisons (BP), and fuel/moderator geometry.[8]

Flibe was found to produce a strong negative void coefficient of reactivity (CVR) for the uranium block loadings needed to achieve an 18-month power cycle length for the AHTR at 2400 MW(t) power. Like the VHTR, the AHTR was shown to have a highly negative fuel temperature reactivity feedback due to the Doppler effect, which is two orders of magnitude greater than traditional light water reactors (LWR). With a small addition of erbium BP into the baseline AHTR design, the strong Doppler effect results in a zero net change in reactivity due to voiding if accompanied by a modest 10^{o}C rise in fuel temperature. Therefore, no realistic voiding scenario could occur that would lead to an increase in reactivity of such magnitude that the increase in fuel temperature would not quickly lead to a subcritical core.

The use of BPs showed that a spectral shift in the neutron flux can be utilized to increase the absorption in the BP resonances when the coolant voids. Among the several candidate BPs, erbium makes the CVR more negative, especially at zero burnup. This is because of the proximity of the erbium absorption cross-section resonance peak to the neutron spectrum peak in the low energy range. Modification of the fuel element dimension was another approach that was found to reduce the CVR.

The homogeneous distribution of fuel compacts in a regular, hexagonal lattice within the fuel blocks can be modified to a 'clustered-rod' design. The clustered-rod design, a tight-pitch array of fuel rods within a single coolant channel surrounded by the graphite block, simplifies refueling by creating a set of assemblies that are removed like traditional LWRs and increases the heat exchanged between the coolant and fuel, which improves thermal performance and decreases the fuel temperature response time (Doppler effect) during an accident scenario. The clustered-rod designs, shown in Figure 4, did not appear to significantly reduce the CVR but could be configured to have a low CVR similar to the baseline design because of the spatial distribution of the spectral shift during a voided coolant scenario.

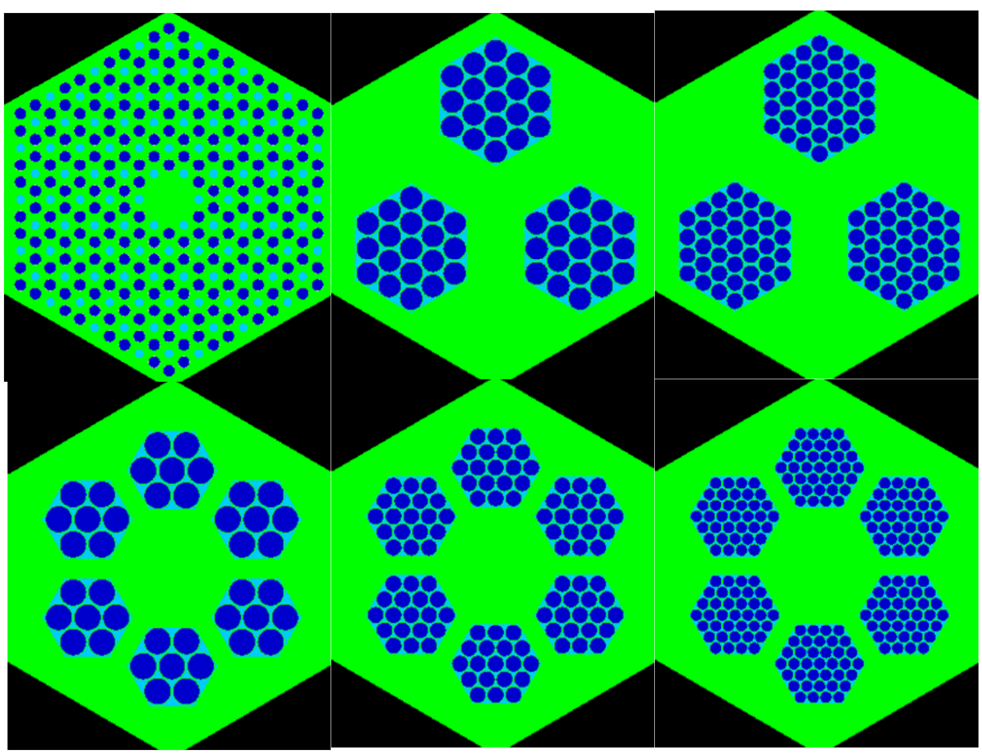

Fig. 4. Baseline (top left) and 'clustered-rod' fuel block configurations of the AHTR.

A RELAP5-3D model of the AHTR was used to simulate the reactor's performance during steady-state operation and during a transient initiated by a loss of forced circulation (LOFC). An RVACS model, based on the PRISM design, was able to adequately cool the AHTR, even with a conservatively high decay heat curve. The maximum calculated fuel temperature during the transient was about 1260°C and occurred about 60 h after the LOFC. The peak fuel temperature during the transient was considerably less than the transient temperature limit of 1430°C, which is set by the boiling temperature of the coolant. Natural circulation of the liquid salt was the dominant heat transport mechanism within the reactor vessel during the LOFC accident. Transient temperature limits were met even for a LOFC with a failure to scram. Figure 5 compares the maximum fuel temperatures achieved during a LOFC with and without a reactor scam. An enhanced RVACS model, based on the proposed S-PRISM design,[7] was also studied and found to further improve decay heat removal. The maximum fuel temperature during the transient was reduced by about 90°C with the enhanced RVACS.

IV. ECONOMICS

The big promise of the AHTR, compared to gas-cooled high-temperature reactors, is greatly improved economics. Because of the overlap of technologies and design features with the S-PRISM and GT-MHR, the initial cost of the AHTR was estimated by scaling relevant cost information from the S-PRISM[9] and the gas-turbine modular high-temperature reactor (GT–MHR)[10] for major components and cost accounts. The result is that the AHTR overnight capital cost (without contingency) is estimated to be approximately \$820/kW(e) (in 2002 dollars), which is 50–55% of the S-PRISM and GT–MHR costs for the same total output of 1300 MW(e).

From the perspective of an economist, the potential for improved economics is an expected consequence of the economy of scale. The AHTR electrical output is approximately four times that of the other two reactors but with similar physical size and complexity. The potential for improved economics compared with LWRs, for a plant of similar power output, is a consequence of higher

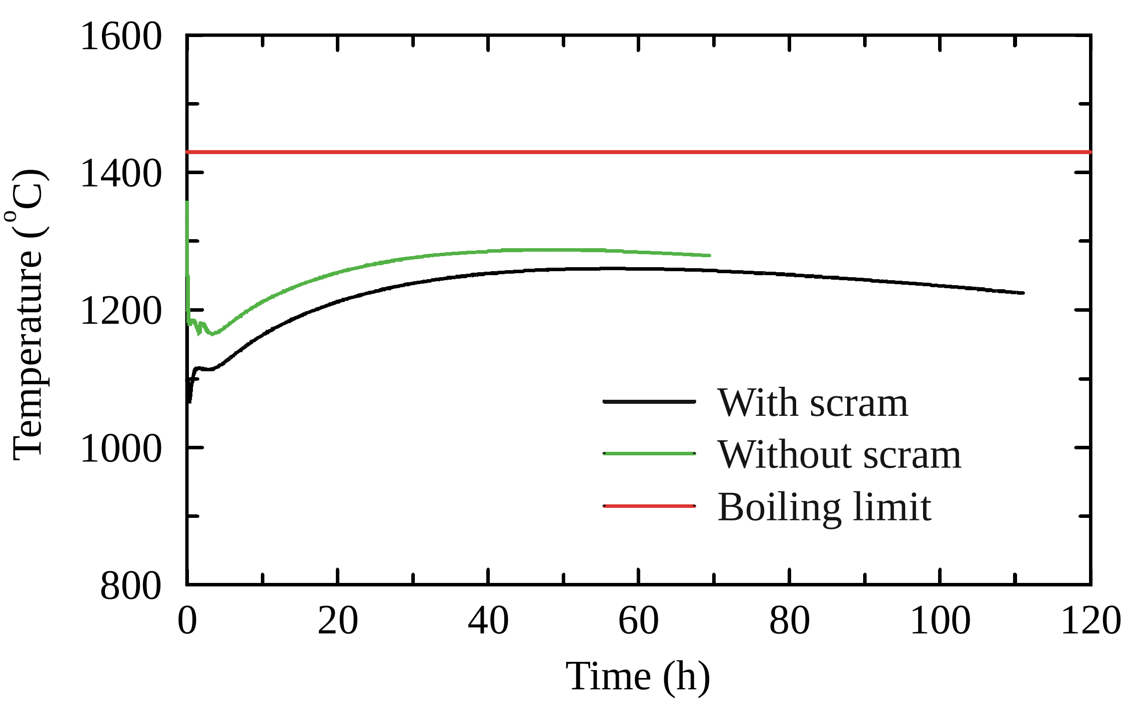

Fig.5. The effect of a failure to scram on the maximum fuel temperature following a LOFC (long-term).

power conversion efficiencies, higher fuel burnup, a low-pressure containment, and the elimination of active safety equipment.

V. SUMMARY AND FUTURE ACTIVITIES

A growing laboratory, industry and university team continues to advance the ATHR concept. Recent analyses indicate that the revised design satisfies the top-level operational and safety targets, i.e., it demonstrates strong passive safety performance, achieves a fuel cycle length of at least 18 months with a fuel discharge of greater than 100 GWd/t, has a negative reactivity response to coolant voiding, and has highly competitive economics. In particular, the current design appears to have excellent steady-state and transient performance, and the previous concern regarding the core's response to coolant voiding has been resolved convincingly for the case of Flibe coolant. Also, the AHTR appears capable of surviving a LOFC, even with failure to scram, because of the significant natural convection that occurs within the reactor vessel and the effective heat removal of the passive RVACS.

Considerable effort is needed to further develop, optimize, and validate the AHTR design. The most immediate needs include the following:

- The current design focused on only Flibe, which is clearly the most favorable from a neutronics perspective. Toxicity concerns for beryllium-containing materials and the potential expense of ^{7}Li enrichment motivate the interest in considering other candidate salts. These other salts need to be evaluated in terms of their impact on the physics and thermal-hydraulic performance of the AHTR and may require design changes to compensate for their less favorable neutronics properties.
- The commercial viability of plant operation and maintenance requirements must be evaluated related to the high freeze temperature of the coolant salt. For example, a high-temperature refueling protocol needs to be developed that ensures a high level of plant safety and availability.
- A more detailed design of the reactor core and vessel internals must be developed, including core supports, vessel insulation, control and shutdown systems, flow baffles, etc. The ultimate performance of the reactor will be substantially impacted by the details of the design. Specific materials will need to be selected consistent with the final coolant choice, temperature distributions, and radiation environments.
- Basic phenomenology associated with potential accidents, including "beyond design basis" accidents, must be studied and prioritized as a guide to the safety validation process.
- Fundamental salt properties and chemistry at higher temperatures must be studied. Previous studies for salts in nuclear environments were limited to 700–800°C and must be extended above 1000°C to cover the temperature range resulting from transients.

In the longer term, numerous coolant validation experiments will be needed to study coolant performance, such as mixing in the inlet/outlet plena, conductive and radiative heat transfer across small salt-filled gaps, etc. The optically transparent liquid salt is expected to have heat transport characteristics much different from other high-temperature liquids, such as liquid metals. Also, activities should include: a study of the licensing basis for the AHTR, component effects tests and scaled separate effects tests, integral effects tests, the development and operation of a small-scale liquid-salt-cooled test reactor, and ultimately the development of a demonstration-scale reactor prior to full commercialization.

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