Advanced-High-Temperature-Reactor Spent-Fuel Characteristics and Repository Impacts

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Abstract ─ *The Advanced High-Temperature Reactor (AHTR) is a large [>2400-MW(t)] liquid-salt-cooled high-temperature reactor that uses a graphite-matrix coated-particle fuel similar to that used in modular high-temperature gas-cooled reactors (MHTGRs). The high-temperature AHTR variant, the liquid-salt-cooled very high temperature reactor (LS-VHTR), is under development within the Department of Energy Generation IV program as an alterative-coolant VHTR. The characteristics of the spent nuclear fuel and the repository impacts are compared with those of light-water reactors and MHTGRs in terms of repository size, fuel cycle characteristics, waste volumes, and repository performance assessments. Comparisons are made per unit of electricity produced. While the AHTR has the same general fuel characteristics as an MHTGR, its higher burnup and potential fuel design options imply potentially significant differences and smaller repository impacts compared with those of MHTGRs.*

I. INTRODUCTION

The Advanced High-Temperature Reactor (AHTR) is a new reactor concept that has been under development for several years.[1, 2] One of the challenges in the development of any new reactor is the associated fuel cycle. This paper is an initial examination of AHTR spent nuclear fuel (SNF) characteristics and the potential repository impacts, assuming direct disposal of the SNF with a burnup of 150,000 MWd/t. The characteristics and repository impacts are compared with those of pressurized-water-reactor (PWR) SNF with a burnup of 50,000 MWd/ton and modular high-temperature gas-cooled reactor (MHTGR) SNF with a burnup of 100,000 MWd/ton. While the comparisons are made with PWR SNF, boiling water reactor (BWR) SNF [the other type of light water reactor (LWR)] has very similar characteristics; thus, the comparisons generally also apply to BWRs. All comparisons are made on the basis of a unit of electricity produced—the useful product of nuclear power generation. Section II describes the reactor concept. Sections III through VI discuss the four major impacts of the AHTR on a repository: repository size, fuel characteristics, SNF disposal volumes, and long-term repository performance.

As a new reactor concept, the AHTR is in an earlier state of development than the MHTGR where several reactors have already been built, two test reactors are operating, and a precommercial demonstration reactor is presently being built. While the differences in repository impacts between the AHTR and other reactors can be identified, the fact that the AHTR is in the early stages of development also implies significant uncertainties.

II. REACTOR DESCRIPTION

The AHTR (Fig. 1) is a large [2400- to 4000 MW(t)] liquid-salt-cooled high-temperature reactor that uses a graphite-matrix coated-particle fuel similar to that used in MHTGRs. The design goals are economic production of hydrogen or electricity with the same high-temperature capabilities and passive safety characteristics found in MHTGRs. The use of liquid salt cooling enables the construction of large reactors with potentially superior economics compared with traditional high-temperature reactors. Preliminary estimates[2] indicate capital costs per kilowatt (electrical) between 50 and 60% of those of an MHTGR. The lower costs reflect economics of scale [(2400 MW(t) vs 600 MW(t)] and the superior properties of a liquid coolant relative to those of a gas coolant, a factor that significantly reduces plant size for plants with equivalent power output.

The AHTR uses optically transparent liquid-salt coolants that are mixtures of fluoride salts with freezing points near 400°C and atmospheric boiling points of ~1400°C. The reactor operates at near-atmospheric pressure. At operating conditions, the heat-transfer properties of the salt 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, which uses a secondary liquid-salt coolant to move the heat to a thermochemical hydrogen production facility to produce hydrogen or to a turbine hall to produce electricity. The baseline AHTR facility layout that was developed is similar to that of the S-PRISM sodium-cooled fast reactor designed by General Electric. Several alternative 2400-MW(t) designs are being investigated that have peak coolant temperatures between 700 and 1000ºC and corresponding electrical outputs between 1151 and 1357 MW(e).

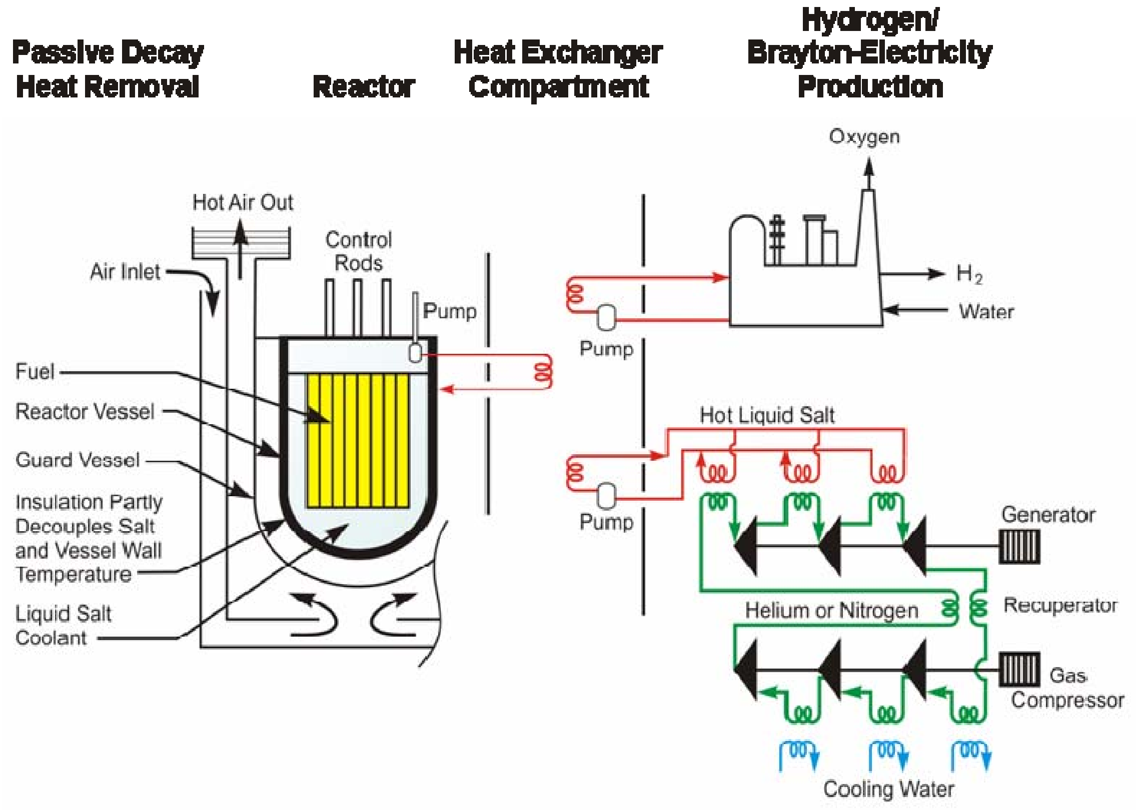

Fig. 1. Schematic of the Advanced High-Temperature Reactor.

III. REPOSITORY SIZE

The primary design constraint for a repository is temperature. Peak temperatures must be limited to avoid unacceptably rapid degradation of the SNF, the waste package (WP), and the geological structure. To limit temperatures, the quantity of decay heat per unit area of the repository must be limited. This consideration, in turn, limits the quantity of SNF than can be disposed of per unit area of the repository. For a repository with a fixed area, such as the planned Yucca Mountain (YM) repository, the capacity is limited by total SNF decay heat. Consequently, if one reactor type is more efficient than another reactor type at converting heat to electricity, more electricity can be produced for the same total decay heat in the SNF or equivalent area in a repository.

The electrical-to-thermal efficiency of a typical PWR is ~33%, whereas the electrical-to-thermal efficiency of the AHTR is ~50%. The higher efficiency reflects the higher reactor coolant temperatures (750 to 1000°C versus <300°C). For the analysis herein, it is assumed that the MHTGR has the same efficiency as the AHTR; however, it is expected that in optimized systems, the MHTGR will have slightly lower efficiencies for the same peak coolant temperatures.[3] Liquid-cooled reactors (such as the AHTR) deliver all of their heat over a relatively small temperature range (50 to 150°C), whereas gas-cooled reactors deliver their heat over a ~350°C range. This is a consequence of the high coolant pumping costs in gas-cooled reactors. To minimize this cost, gas-cooled reactors must be optimized to have larger temperature rises across the reactor core than liquid-cooled reactors. A gas-cooled reactor that has the same peak coolant temperatures as a liquid-cooled reactor will have a lower average temperature of delivered heat to the power conversion system and thus a somewhat lower plant efficiency.

The PWR generates 1 kWh of electricity for every 3 kWh of thermal heat, whereas a high-temperature reactor produces 1 kWh of electricity for every 2 kWh of heat. During reactor operations, the PWR SNF must produce 50% more heat per unit of electricity generated. As a result, its SNF also produces 50% more decay heat in the repository per unit of electricity produced compared with that from a high temperature reactor and thus requires a repository area that is 50% larger per unit of electricity. There are also smaller second-order effects due to the differences in SNF burnup, fuel design, and the neutron spectrum.

IV. SNF AND FUEL CYCLE CHARACTERISTICS

The AHTR and MHTGR use the same type of graphite-matrix coated-particle fuel; however, the AHTR fuel burnup is ~50% higher than that of an MHTGR.[4,5] The AHTR is a large reactor [2400 MW(t)] relative to the MHTGRs [600 MW(t)]. The baseline AHTR core is a large right cylinder made of columns of prismatic fuel blocks, whereas the MHTGR has a smaller annular reactor core to assist decay heat removal. Figure 2 shows the core layout of prismatic fuel blocks for both core types. The small annular core of the MHTGR implies high neutron leakage (3.5 to 6%), both inward toward a center graphite cylinder and outward toward the reactor vessel. In contrast, the small surface-to-volume ratio of the large AHTR core implies relatively small neutron leakage (1 to 2%). For nuclear criticality to be maintained, the average enrichment of the MHTGR core must be higher than in an AHTR. If the two reactors have similar initial fuel enrichments, the AHTR can have a lower end-of-life SNF enrichment and a corresponding higher SNF burnup. Table I shows relative SNF burnups for the two reactors with similar initial fuel enrichments.

The different reactor core designs are a consequence of the choice of coolants and the common requirement that these advanced reactors have passive decay-heat-removal systems—systems that do not depend upon human actions or active components to ensure removal of decay heat and thus ensure that fuel temperature limits are not exceeded. During an accident, the decay heat systems prevent excessive temperatures in the reactor core that could damage fuel. For both the gas-cooled and liquid-salt-cooled reactors, decay heat must be removed from the hottest fuel elements in the reactor core to the reactor vessel surface, where passive systems then dump the heat to the atmosphere. Different types of systems are used (Fig. 2).

- *Helium cooled*. Under accident conditions, decay heat is removed by conduction of heat from the fuel in the reactor core to the reactor vessel. In accidents involving depressurization of the reactor, natural circulation of helium does not transfer significant heat from the reactor core to the vessel. For a maximum allowable fuel temperature before fuel failure, heat can be conducted through a defined thickness of fuel blocks to the reactor vessel without failure of the hottest fuel because of excessive temperature. The thickness of the fuel zone is limited by decay-heat-removal requirements. To build larger reactors, an annual core is used with no fuel in the middle—the fuel thickness is limiting. While the annular zone can be made larger, the maximum size is limited by the size of practical pressure vessels. The requirement for passive decay heat removal systems restricts the power output to ~600 MW(t), with the core shown in Fig. 2.

- *Liquid salt cooled*. Natural circulation of liquid salts can efficiently move heat from anywhere in the reactor core to the reactor vessel. Reactor size is limited by the ability to remove heat from the vessel, not the ability to move heat from the fuel to the vessel wall. Reactors can be built with passive safety, large reactor cores, and more efficient burning of nuclear fuel.

This difference in core design, a consequence of the same passive safety requirements for both reactors, has several implications in terms of the fuel cycle and repository.

- *SNF volumes*. The AHTR SNF volumes are reduced by one-third relative to those of MHTGRs per unit of electricity produced due to the higher fuel burnup for the same fuel enrichments.

- *Nuclear criticality*. The AHTR SNF uranium enrichments are between those of the big PWRs and the MHTGRs. PWRs and the AHTR are both large reactors with large reactor cores and little neutron leakage from the core. In contrast, MHTGR SNF[6] has a relatively high end-of-life uranium enrichment of 5.6%. The high enrichment is necessary to maintain nuclear criticality during normal operation.

- *Uranium resources and depleted uranium (DU).* The AHTR and PWRs have somewhat similar natural uranium demands and generate similar quantities of DU per unit of thermal heat produced. The uranium consumption per kilowatt (electric) and DU production are lower for the AHTR than for the PWR because of the higher efficiency in converting heat to electricity. The natural uranium demands and the quantities of DU that are generated in the enrichment processes are ~15% higher per unit of electricity for the MHTGR[6] compared with those for an LWR because of the high residual uranium enrichments in the SNF sent to the repository.

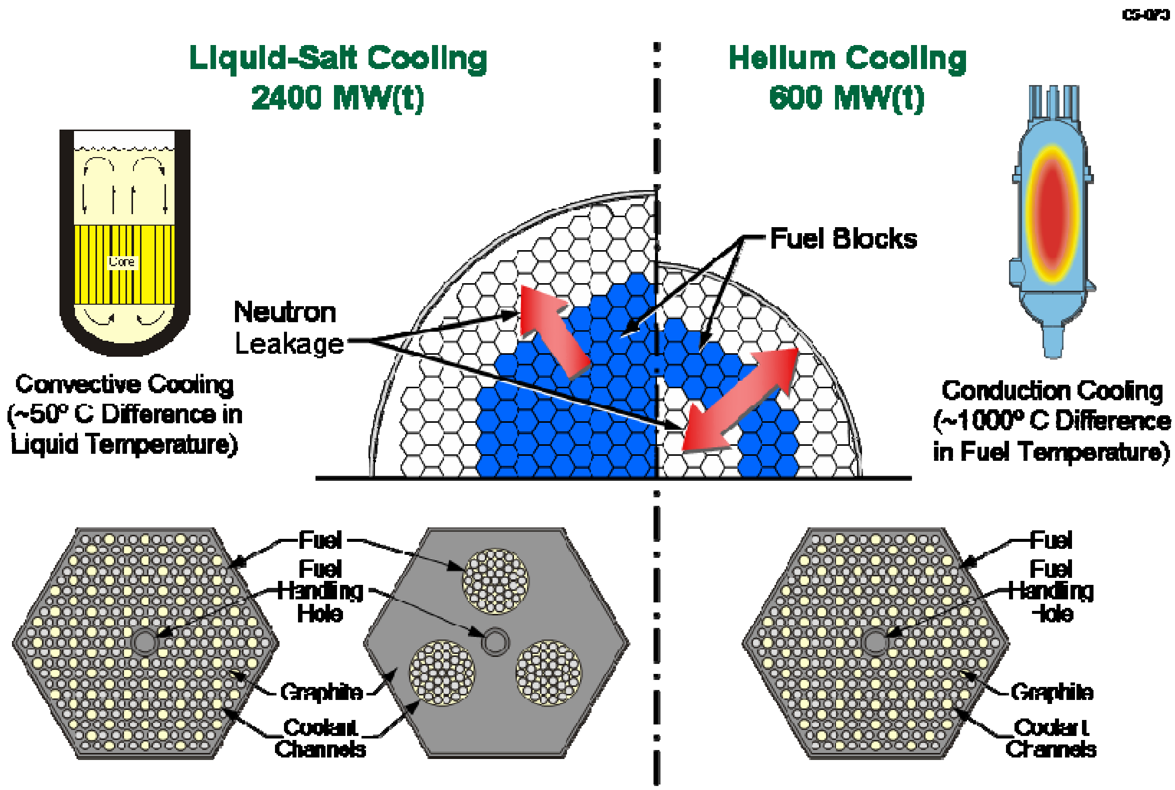

Fig. 2. Differences in liquid-salt-cooled AHTR and helium-cooled MHTG reactor cores.

TABLE I. Relative Core and Fuel Cycle Parameters
for the MHTGR and AHTR with Two Batch Refueling

Parameter	MHTGR	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
^{235}U Enrichment, %	14.0	15.3
Burnup, GWd/t	100	156

V. SNF VOLUME

While the SNF decay heat load determines repository area, the waste volume determines the size and number of WPs required. Because three alternative SNF strategies are currently being assessed, the waste volume per unit of electricity for the AHTR can not yet be determined.

- *Whole-block disposal of traditional prismatic block*. The entire fuel block may be disposed of.

- *Separation of fuel compact*. Traditional high-temperature reactor fuel is a prismatic graphite block where fuel holes and coolant channels are drilled into the block. Fuel compacts in the form of coated particles in a graphite matrix fill the fuel holes. The fuel compacts can be mechanically separated from the graphite SNF block. There have been limited experimental and theoretical investigations of this option for separation and direct disposal[7] of the SNF compacts and as a front-end step for reprocessing the SNF.[8]

- *Single and multipin fuel assemblies*. Alternative fuel designs use a prismatic graphite block where coolant holes are drilled into the graphite block with fuel pins in the coolant holes. The Japanese High-Temperature Test Reactor (HTTR) has small coolant holes with a single graphite fuel pin in each hole. For the AHTR, there is the option of having a multipin fuel assembly in large holes in the graphite blocks (Fig. 2); however, this option is in the very early stages of development. The concept of a multipin fuel assembly may not be viable for a gas-cooled reactor, because—should an accident occur—the decay heat must be conducted out of the reactor through the graphite blocks. With a fuel assembly concept in a gas-cooled reactor, there are large temperature drops between the middle of the multipin fuel assemblies and the graphite blocks. This results in higher fuel temperatures under accident conditions. This factor is not a concern for an AHTR because the fuel assembly is cooled by the natural circulation of the liquid salt in the reactor vessel. Studies have identified separate disposal options for the irradiated graphite.[7, 8, 9]

For traditional prismatic fuel blocks, Table II shows the electricity generation per unit volume of SNF measured in gigawatt-days (electrical) per cubic meter. PWRs generate the most electricity per unit volume of SNF, followed by the AHTR and the MHTGR. Alternative fuel designs or separation of the fuel compacts from the graphite matrix would likely result in the AHTR generating the largest quantity of electricity per unit volume of SNF.

In the development of PWR fuel, the electricity generation per unit volume of SNF has increased by about a factor of 3 since the 1960s and has now leveled off. The development of high-temperature fuel is at a much earlier stage of development, but offers the potential to increase fuel loadings per fuel assembly and to optimize the fuel design. The same economic incentives (longer fuel cycles and less SNF) exist to increase the capabilities of high-temperature reactor fuel. The expectation is that the electricity generation per unit volume of high-temperature reactor SNF will also significantly increase with experience.

In terms of repository design, the different characteristics of the graphite-matrix SNF compared with those LWR SNF allow for multiple strategies to compensate for the higher SNF volumes.

- *WP size*. To avoid SNF fuel degradation, the maximum PWR WP temperature limit is set at 350ºC. There is no incentive to use large WPs, because the peak PWR SNF temperature limits would then be exceeded. The equivalent temperature limits for graphite-matrix fuel have not been determined but are likely to be significantly above 500ºC. This allows the use of much larger WPs without the risk of overheating the SNF.

- *Placement*. To spread out the heat load and avoid exceeding near-field temperature limits, the traditional PWR WP placement is horizontal. For high-temperature reactor fuel with its lower decay heat per unit volume, the heat load per package does not drive package placement strategy. Alternative options, such as vertical placement, are available to place more WPs per unit length of disposal tunnel.

VI. REPOSITORY PERFORMANCE

Studies on the repository performance of high-temperature reactor fuel[7, 8, 10–13] suggest that this SNF may be a multimillion-year waste form, orders of magnitude better than traditional LWR SNF. *If fully confirmed*, this performance would radically reduce the requirements and costs for the WP and alter the licensing requirements for a repository with these graphite-based fuels. There are two major barriers to the release of radionuclides from these SNFs.

- *Graphite*. The fuel is incorporated into a graphite block. In the natural environment, graphite is an extremely inert material. Graphite is used in the chemical industry in heat exchangers and other applications for highly corrosive environments. Because of its inertness, it has been considered as a WP option for the planned YM repository.[11]

Projections of graphite performance under oxidizing conditions similar to those expected in the YM repository indicate potential lifetimes of tens of millions of years. Potential treatment options exist to improve performance, including methods to address uncertainties in repository performance that may be created by fuel irradiation. The coolant channels in the SNF could be filled with graphite and the fuel block treated with organic sealants, which are then pyrolyzed to create very low permeability monolithic graphite blocks.[8] There has been significant work on reducing the permeability of irradiated graphite moderator blocks from gas-cooled reactors[12] by this method. Many of the earlier gas-cooled reactors had metal-clad fuel and graphite moderator blocks. Fuel failures and the relatively-permeable moderator graphite that was used resulted in significant quantities of fission products in the blocks; thus, a need to seal the radionuclides into the graphite to make a waste with low leachability to groundwater. This experience and the investigations of graphite as a WP material indicate the need to consider waste management implications in the choice of graphite. Because of the potential of the graphite as a major barrier to radionuclide releases, it is unclear whether it is desirable to separate the fuel compacts from the graphite matrix.

- *Silicon carbide*. The uranium, fission products, and actinides are incorporated into microspheres with graphite and silicon carbide coatings that are all relatively inert to the repository environment.[10, 13, 14] This represents a second potential barrier, and initial analysis and experiments indicate orders-of-magnitude better performance than with LWR SNF with zircalloy cladding or high-level-waste glass.

TABLE II. SNF Characteristics

Property	PWR	MHTGR	AHTR
Fuel burnup, GWd(t)/ton uranium	50	100	150
Electrical efficiency, %	33	50	50
Electricity per unit volume SNF, GWd(e)/m^3	20	3.3	5

VII. GRAPHITE-MATRIX SNF REPOSITORY

Repository design and capacity depend upon the geology, waste characteristics, and requirements. The YM design is based on LWR SNF. If the quantities of graphite-matrix fuel are small, the repository design will remain unchanged. If large quantities of graphite-matrix SNF are present, a section of the repository will be optimized for this specific SNF. The inert high-temperature characteristics of this SNF, compared with those of LWR SNF, may allow repository designs with many times the capacity of an equivalent LWR repository measured as repository area per unit electricity produced. SNF characteristics drive the repository design.

VIII. CONCLUSIONS

Compared to PWR SNF, AHTR SNF will require less repository area per unit of electricity produced because of the higher efficiency in converting heat to electricity. Fewer radionuclides are produced per kilowatt-hour (electrical). The AHTR will have lower SNF volumes and a lower SNF fissile content than the MHTGR because of the higher SNF burnup for the same initial uranium enrichment levels. The AHTR will also require fewer uranium resources and generate less depleted uranium than a PWR per unit of electricity produced. The AHTR SNF volumes per kilowatt (electrical) are larger than those of high-burnup LWR SNF. Based on limited data, the potential performance of the graphite-matrix coated-particle SNF in a repository is several orders of magnitude better than that of PWR SNF.

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