FHR Development Roadmap and Test Reactor Requirements White Paper

Integrated Research Project Workshop 4

Fluoride-Salt-Cooled, High-Temperature Reactor (FHR) Development Roadmap and Test Reactor Performance Requirements White Paper

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Preamble

The University of California, Berkeley; Massachusetts Institute of Technology; and University of Wisconsin, Madison, hosted a series of four workshops during 2012 under a U.S. Department of Energy-sponsored Integrated Research Project (IRP) to review technical and licensing issues for fluoride-salt-cooled, high-temperature reactors (FHRs). This white paper reports results from the fourth and final workshop, discusses functional requirements and goals for an FHR test reactor (FHTR), and recommends a scaling approach to address these functional requirements and goals. This approach includes ways to maximize the information that an FHTR can produce to predict the operational reliability of commercial-scale plants, as well as their response to licensing basis events. This workshop also provided input for drafting an FHR development roadmap, which will be issued in the third year of this IRP. Finally, this workshop helped identify the institutional challenges that must be addressed in the roadmap—including the commercialization strategy and the fuel cycle. The four workshops provided an opportunity to receive critical feedback from technical experts in relevant fields.

The four workshops are a central element of developing an FHR preliminary conceptual design report to be completed in 2014. This fourth white paper focuses on material covered by the fourth workshop and is divided into five main chapters and a series of appendices. The first chapter provides an overview of the FHR workshop series, FHR development goals and commercialization challenges. The second chapter describes an FHR development and commercialization strategy. The third chapter reviews preparatory activities needed to develop an FHTR, while the fourth chapter discusses major design and licensing requirements for it. The fifth discusses the commercialization roadmap. Appendices elaborate on certain issues in the second through fifth chapters.

The comments of the experts attending the workshop, who are listed on the title page, were also integrated into this white paper. The IRP team sincerely appreciates the input of all of the experts who attended and contributed to the fourth FHR workshop, as well as the hard work of the graduate students and postdoctoral scholars who organized it, facilitated the sessions, and drafted the major sections of this white paper based on their research and the review and input of the workshop experts.

Executive Summary

Fluoride-salt-cooled, high-temperature reactor (FHR) technology uses a novel combination of high-temperature coated-particle fuel, fluoride-salt coolant, and a low-pressure primary system to deliver heat in the temperature range from 600°C to 700°C or higher. FHRs exhibit different thermal hydraulic, neutronic and structural mechanics phenomena compared to conventional – and more extensively studied – advanced nuclear reactor concepts. This white paper reviews the results from a 2-day expert workshop hosted by the FHR Integrated Research Project (IRP) and held in Cambridge, Massachusetts, in November 2012. This workshop reviewed and discussed the commercialization strategy for FHRs, the resultant reactor functional requirements, the requirements for the FHR test reactor (FHTR) and the preparatory activities required to support an FHTR.

The FHR technology that has been selected as the baseline for the IRP presents a strong economic case because it can tackle markets that are not usually targeted by the nuclear energy sector. The three main markets are baseload electricity, peaking power, and process heat production. Baseload electricity is a conventional market for nuclear reactors given their high capacity factors. The FHR coupled with a nuclear air-Brayton combined cycle (NACC) is projected to have a baseload efficiency between 40% and 47%, which is significantly greater than the efficiencies of current nuclear power plants. The NACC power cycle has air going through an air-Brayton cycle and then discharged to a heat recovery steam generator (HRSG), where added steam is produced for electricity production or sales. While steam cycles and closed gas cycles are also candidate technologies for FHR power conversion, the IRP has selected NACC power conversion as its baseline because of its unique ability to produce peaking power to meet future electrical grid needs and to enhance revenue generation by providing additional grid support services besides baseload electricity.

To produce peaking power with this approach, natural gas is injected after the last stage of nuclear air heating. The increased inlet temperature into the last turbine stage allows for higher power output. Peak electricity generation is a premium market (increased revenue) and will be available to whatever technology has low capital cost for peak power production. The initial assessment is that the FHR has a large competitive advantage in this market, and thus air-Brayton combined power cycles should include peaking capability. There is also the longer-term option of using biofuels or hydrogen for peaking power.

The output temperature of the FHR using available structural materials is 700˚C, allowing for process heat production that could be used for many applications such as shale oil recovery or hydrogen production. Because hot air is the heat transfer media to the HRSG, the steam plant is decoupled from the nuclear plant such that the steam plant can be optimized for electricity or process heat production independent of the reactor—ensuring no potential for radioactive contamination in the steam. NACC provides high-temperature steam.

These capabilities lead to an IRP commercialization strategy with two goals: (1) the enabling technology for a low-carbon nuclear-renewable electricity grid to meet national energy objectives and (2) over 50% greater revenue after subtraction of natural gas fuel costs in a deregulated electricity market than a traditional baseload nuclear power plant—a capability to ensure competitive economics.

FHR technology needs to be demonstrated in an FHTR before a commercial reactor can be built. The two goals of the FHTR are to (1) provide confidence that a full-scale demonstration prototype reactor is warranted, and (2) develop the necessary data to enable the design and licensing of a demonstration power reactor. The FHTR will provide thermal hydraulic, neutronic, structural mechanics and corrosion data for code validation. It will also provide operational experience with handling the fluoride-salt coolant and the fuel, as well as the system for tritium control and management. Some of the main functional requirements of the FHTR differ from the FHR because of scaling or added flexibility for testing. These requirements include the salt pumps, tritium management system, redox control system, and the reactivity control system. The FHTR also provides an important opportunity to test high-temperature instrumentation.

Another key function of the FHTR is to test structural materials such as 316 stainless steel, Alloy N, graphite, and silicon-carbide composites. The data from the FHTR would support qualification of in-core materials at the FHR irradiation conditions. Additionally, fuel will be tested in the FHTR using one of two approaches: (1) a prototypical fuel is used, or (2) a variety of fuel forms and types are tested, with the aid of a driver fuel. This decision will depend on the design of the FHTR, whether it is a prototype test reactor or a general-purpose test reactor. The FHTR will also test licensing basis events such as loss of heat sink, loss of forced circulation, overcooling transients, and reactivity insertion transients. Thus, the FHTR is a key tool in ensuring the viability of the FHR concept, as well as providing important information to validate neutronic, thermal hydraulic, and structural codes through integral effects tests.

A key for development of the FHR technology is the enrichment of ^{7}Li to 99.995% in the flibe coolant to reduce parasitic neutron capture in the coolant to a level that enables the reactor to have negative temperature and void reactivity feedback. Several options exist, and if the market for enriched lithium can be proven, the process will be facilitated. Promising methods in terms of economics and effectiveness include crown ether solvent extraction or atomic vapor laser isotope separation. The use of crown ethers has been demonstrated recently by the Chinese Academy of Sciences (CAS) and is now the baseline enrichment method for the CAS Thorium Molten Salt Reactor project.

The workshop concluded that a separate component test facility will not be necessary. Components can be tested in the FHTR or at other appropriate test loops located at universities or national laboratories.

Ownership of the FHTR has implications on its ultimate mission, funding, and licensing strategy. The ownership options include the government, the government with international partners, a university, industry, or a consortium or combination of owners. A consortium of owners – both international and domestic – is beneficial because it allows costs of construction, maintenance, and operation to be shared, while increasing applicability of the technology at a global scale. The baseline strategy is that the U.S. Department of Energy would be the owner, and the reactor would be licensed as a Class 104 (c) test reactor for research and development.

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Acronyms and Abbreviations

AEC – Atomic Energy Commission
AGR – Advanced Gas-Cooled Reactor
AHTR – Advanced High-Temperature Reactor
ALWR – Advanced Light-Water Reactor
ANL – Argonne National Laboratory
ANS – American Nuclear Society
ASME – American Society of Mechanical Engineers
ATR – Advanced Test Reactor
ATWS – Anticipated Transient Without Scram
AVLIS – Atomic Vapor Laser Isotope Separation
AVR – Arbeitsgemeinschaft VersuchsReaktor
BDBE – Beyond Design Basis Event
BPV – Boiler and Pressure Vessel
BWR – Boiling Water Reactor
CCS – Carbon Capture and Sequestration
CFRC – Carbon Fiber-Reinforced Carbon
C/HM – Carbon to Heavy Metal ratio
COLEX – Column-Exchange (process)
CTAH – Coiled-Tube Air Heater
CTF – Component Test Facility
CVD – Chemical Vapor Deposition
DOE – U.S. Department of Energy
DPA – Displacements Per Atom
DRACS – Direct Reactor Auxiliary Cooling System
EM – Evaluation Model
EPRI – Electric Power Research Institute
FHR – Fluoride-salt-cooled, High-temperature Reactor
FHTR – FHR Test Reactor
FSVR – Fort St. Vrain Reactor
GA – General Atomics
HFIR – High Flux Isotope Reactor
HRSG – Heat Recovery System Generator
HTGR – High-Temperature Gas-cooled Reactor
HTR – High-Temperature Reactor
HTTR – High-Temperature Test Reactor
IAEA – International Atomic Energy Agency
IET – Integral Effects Test
IGCC – Integrated Gasification Combined Cycle
INL – Idaho National Laboratory
IRP – Integrated Research Project
JAEA – Japan Atomic Energy Agency
LBE – Licensing Basis Event
LCOE – Levelized Cost Of Electricity

LOFC – Loss Of Forced Circulation
LOHS – Loss Of Heat Sink
LWR – Light-Water Reactor
MHA – Maximum Hypothetical Accident
MIT – Massachusetts Institute of Technology
MSRE – Molten Salt Reactor Experiment
NAA – Neutron Activation Analysis
NGCC – Natural Gas Combined Cycle
NGNP – Next Generation Nuclear Plant
NITE – Nano-powder Infiltration and Transient Eutectoid
NRC – U.S. Nuclear Regulatory Commission
ORNL – Oak Ridge National Laboratory
PB-AHTR – Pebble-Bed Advanced High-Temperature Reactor
PIE – Post Irradiation Examination
PIRT – Phenomenon Identification and Ranking Table
PR – Pressure Ratio
PWR – Pressurized-Water Reactor
RDC – Regulatory Design Criteria
SAR – Safety Analysis Report
SET – Separate Effects Test
SmAHTR – Small modular Advanced High-Temperature Reactor
SS – Stainless Steel
TFHR – Transportable FHR
THTR – Thorium High-Temperature Reactor
TRISO – Tristructural Isotropic
VHTR – Very High Temperature Reactor
UCB – University of California, Berkeley
UW – University of Wisconsin, Madison

1 Introduction

Fluoride salts have unique thermophysical properties compared to other reactor coolants, which make them potentially attractive to use as coolants for high-temperature, low-pressure reactors called fluoride-salt-cooled, high-temperature reactors (FHRs). The U.S. Department of Energy (DOE) has initiated an Integrated Research Project (IRP) with the Massachusetts Institute of Technology (MIT); University of California, Berkeley (UCB); and University of Wisconsin, Madison (UW), to develop the technical basis to design, develop, and license commercially attractive FHRs. This is one of four white papers developed during the first year of the IRP to aid in identifying the technical basis and unique issues for the design and licensing of FHRs.

This white paper reviews key issues for design and licensing of an FHR test reactor (FHTR) and the commercialization strategy for subsequent commercial prototype reactors. The following sections provide an overview of the FHR workshop series, FHR development goals and commercialization challenges. The second chapter reviews preparatory activities needed to develop an FHTR, while the third chapter discusses major design and licensing elements for it. The fourth chapter discusses FHR commercialization and provides input for the commercialization roadmap in the fifth chapter.

1.1 Overview of the FHR Workshop Series

To initiate the IRP, UCB, UW, and MIT organized a series of four workshops in 2012 to engage reactor technology experts in identifying and reviewing key FHR development issues (Figure 1-1). The first FHR workshop discussed the major technical characteristics that differentiate FHRs from other power reactor technologies, the major systems and subsystems expected to be used in FHRs, high-level functional requirements for these systems and subsystems, and licensing basis events (LBEs) that should be considered in FHR design and licensing. The second workshop studied key thermal hydraulic, neutronic, and structural response phenomena and identified system response codes to predict the response of FHRs under steady-state operation and design basis events, along with experimental data needs to validate these models. The third workshop reviewed key fuel and material needs unique to FHRs, as well as methods for tritium and beryllium control.

The experts who attended the fourth FHR workshop brought extensive experience in reactor design and development, including licensing and commercialization. Their specific areas of expertise include the following:

- High-temperature component test facilities
- Test reactor design and functional requirements
- Test reactor and commercial prototype reactor licensing
- Test reactor deployment and operations
- National policy goals/requirements for advanced reactor development
- Utility goals/requirements for advanced reactor development.

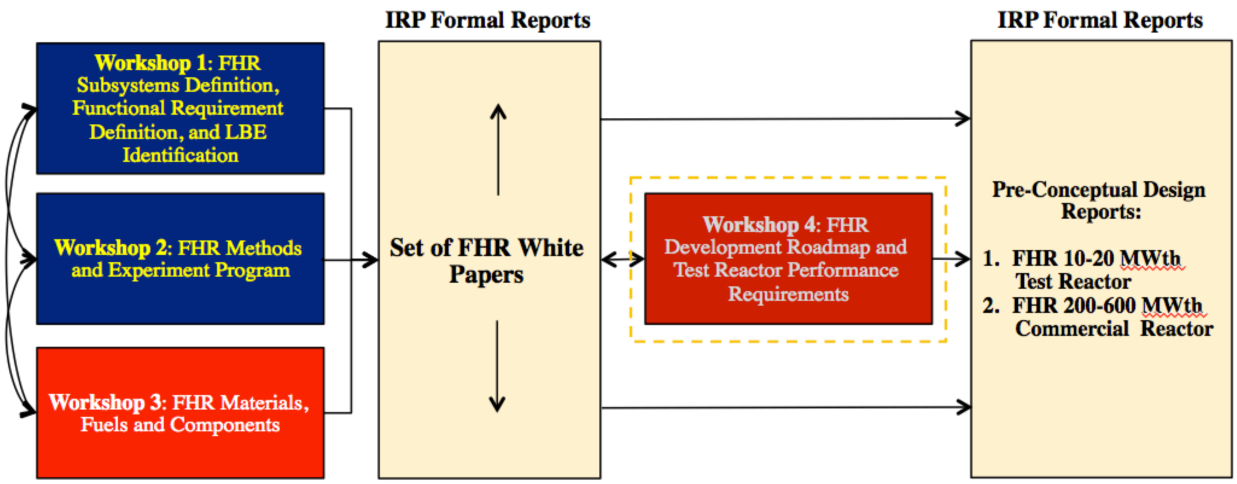

Figure 1-1. IRP Structure Illustrating Workshop Rationale and Key IRP Objectives (this white paper focuses on the fourth workshop)

1.2 Stakeholder Perspectives on FHR Design and Development Goals

The IRP includes an advisory panel primarily consisting of industry members and led by Regis Matzie (retired Chief Technical Officer of Westinghouse). This panel provides a stakeholder perspective where the ultimate customers are the vendors and utilities. The primary recommendation of the panel is that, if the FHR is to be successful, there must be a compelling economic case. Most advanced reactor development programs have failed because of economics, not technology. As a consequence, the IRP has placed a major emphasis on developing a commercialization strategy, an emphasis that was highlighted at this workshop. This approach has led to the development of a top-down strategy: market defines top-level reactor goals, reactor goals guide reactor design decisions, and commercial reactor technical challenges define test reactor goals.

The other major goal is safety—particularly in the context of the post-Fukushima world where safety is important in terms of both licensing and social acceptance. This subject has been primarily addressed in earlier workshops but becomes important in the context of an FHTR where one of its primary purposes is to help provide the data for the licensing of a commercial prototype FHR.

1.3 Overview of FHTR and Commercialization Challenges

The earliest commercial deployment of the FHR is ~2030; thus, the FHR should be designed to meet the requirements of the electricity grid in 2030. By then, the world will be rapidly progressing to a low-carbon nuclear-renewables electricity system with different requirements that include large variations in electricity demand on an hourly to seasonal basis. This change is partly the consequence in the growing use of non-dispatchable wind and solar electricity production. The commercialization challenge is thus to define nuclear power plant requirements based on the expected market.

The market assessment has led to the selection of an FHR baseline design with a nuclear air-Brayton combined cycle (NACC) that can produce baseload electricity, process heat, and peak power using auxiliary gas co-firing. This system can significantly increase plant revenue relative to a nuclear plant designed for baseload electricity production and thus improve the economics. The use of such an approach, in turn, defines reactor requirements such as peak temperatures and the need for advanced tritium control strategies.

The needs of the commercial prototype FHR define one set of requirements for the FHTR. However, an FHTR can address multiple irradiation needs—many that are not directly associated with development of an FHR commercial prototype. As a consequence, the market and design requirements for the FHTR do not all follow from the development of a commercial prototype FHR. There are a separate set of challenges associated with the FHTR, as discussed more fully in the following chapters.

2 FHR Development and Commercialization Strategy

Over the last decade, several reactors have undergone development and started the commercialization process. The different characteristics of FHRs (Table 2-1) will require a different development and commercialization strategy. FHRs combine high-temperature graphite-matrix coated-particle fuel developed for high-temperature, gas-cooled reactors with a high-temperature, low-pressure liquid coolant that is a mixture of ^{7}Li fluoride and beryllium fluoride (flibe) originally used in molten salt reactors. However, in molten salt reactors, the fuel was dissolved in the coolant, whereas the FHR will use a solid fuel. There are several other candidate coolant salts but they all have somewhat similar properties. Like other liquid-cooled reactors, the heat is delivered at high temperatures (700°C) over a relatively small temperature range. The baseline FHR is described in Appendix A.

Table 2-1. Reactors by Peak Temperatures and System Pressures

Coolant Temperature	System Pressure	
	Low	High
Low	—	Light-water reactor (LWR)
Medium	Sodium fast reactor	—
High	FHR	High-temperature gas-cooled reactor (HTGR)

The combination of characteristics may enable the FHR to enter energy markets not available to other reactor types. This potential implies a different commercialization strategy as well as different requirements for the reactor that feed back to the test reactor requirements that in turn feed back to the research and development needs.

This chapter defines (1) commercial markets for the FHR, (2) alternative markets, and (3) the path forward. It provides a framework to address the issues of test reactor and research and development needs.

2.1 FHR Commercial Markets

Under the IRP's base case assumption of an FHR coupled to a NACC, three primary markets can be considered: baseload electricity, peak electricity, and process heat. These markets hold implications for others as well. Baseload electricity is the usual target market for nuclear generating plants because of their high capacity factors. Providing peak electricity is a new market accessible because of the open air power conversion cycle. The process heat market for an FHR with NACC should be more accessible because the reactor has the ability to provide a wider range of temperatures, and no reboiler is needed to control tritium from diffusing into the process steam.

High operating temperatures allow coupling of the FHR with a variety of power cycles using different fluids – steam, supercritical carbon dioxide, helium, and air. For the base case, the IRP is coupling to a NACC with a steam bottoming cycle (Figure 2-1). The projected thermal efficiency is 43% to 47% (Summerson 2012). The power cycle is similar to that used in combined-cycle natural gas plants except the operating temperatures are lower (McDaniel et al. 2012).

In the power cycle, air would be compressed using a compressor identical to those used in natural-gas combined-cycle plants. Heat would be added to the compressed air via high-temperature salt-to-air heat exchangers similar to those developed originally for the Aircraft Nuclear Propulsion Program. The hot compressed gas would go through multiple turbines, producing electricity. After the Brayton cycle, the hot air would be sent to a steam recovery boiler to produce steam that would generate additional electricity or process heat. The nuclear-heated gas turbine would be coupled to the steam recovery boiler by a low-pressure hot-air duct with large hot-air flows.

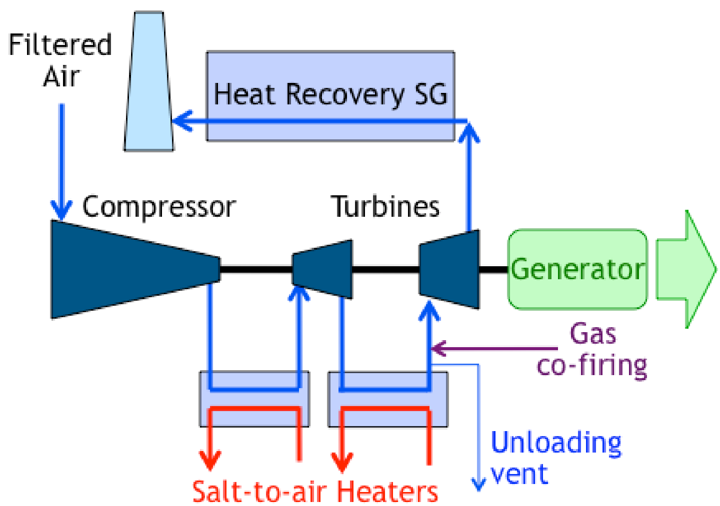

Figure 2-1. FHR Combined-Cycle Power System

Large, rail-transportable gas turbines associated with natural-gas combined-cycle plants at FHR conditions imply ~250 MWth of nuclear heat input per turbine. The typical natural-gas combined-cycle plant has two or three gas turbines feeding one steam plant (Chellini 2007). The existing commercial Brayton power-cycle technology implies FHR thermal output in multiples of 250, 500, or 1,000 MWth.

This power cycle can only be coupled efficiently to high-temperature, liquid-cooled nuclear power plants such as FHRs, molten salt reactors, and high-temperature lead-cooled fast reactors. Coupling to HTGRs causes efficiency penalties. In a gas-cooled reactor, the pumping power for the gas through the core is significant. To reduce that pumping power, there is a large temperature drop across the core to maximize heat removal per unit of gas. Typical gas-cooled reactor inlet temperatures are 200°C to 300°C. However, the exit temperature of a modern Brayton power-cycle compressor is nearly 400°C. The inlet temperature of the gas-cooled reactor has to be raised to be in excess of the 400°C temperature from the compressor to add heat to the cycle after air compression, with major penalties in the reactor design.

2.1.1 Baseload Electricity Production

The traditional market for a nuclear power plant is providing baseload electricity. In this regard, the FHR will be no different than other nuclear power plants. In 2011, 40% of all U.S. energy consumption was for electrical power. Of that, 21% was nuclear generated (U.S. Energy Information Administration 2012). Increasing this percentage will enable the U.S. to meet its goals of reducing greenhouse gas emissions by replacing aging coal-fired plants. In the near term, replacing natural gas plants does not seem feasible because their capital and operating costs are so low. Currently, the major advantage in baseload generation that nuclear power enjoys is its cheap variable productions costs. Figure 2-2 shows the consumer price ("Retail Electricity") versus provider price. Note that nuclear has nearly an order of magnitude reduction in variable costs compared to coal and more than an order for natural gas. While nuclear power currently displays this variable cost dominance, an increase in the fuel costs associated with the FHR's pebble-type fuel will need to be considered in a more detailed analysis.

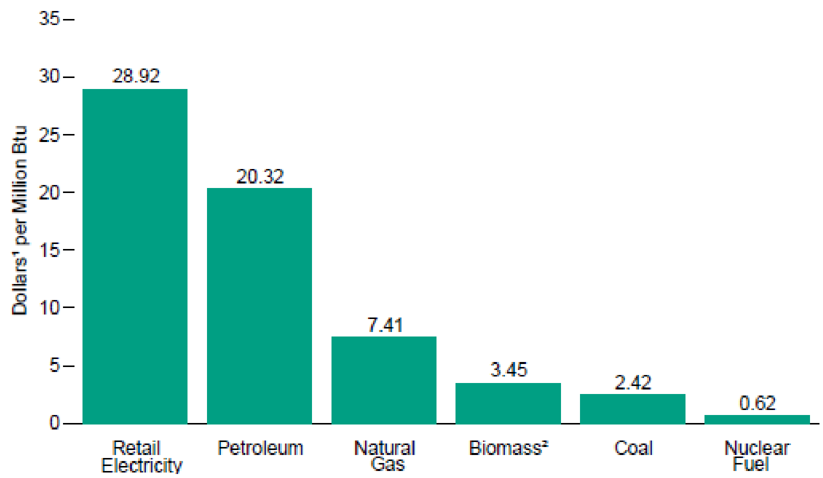

Figure 2-2. Consumer Price Estimates for Energy by Source in 2010 (U.S. Energy Information Administration 2012)

A more realistic view of the baseload market must take into account the initial construction costs and their effects on the levelized cost of electricity (LCOE). Current estimates place nuclear as one of the most capital-intensive choices for new electricity generation in terms of construction. In dollars per kilowatt, nuclear power plants are expected to carry a premium of nearly three to five times the initial capital for the same capacity. The IRP's current FHR base case design of 100 MWe does lessen the total project cost by having an order of magnitude less nominal capacity. Table 2-2 summarizes overnight costs and total project costs for various electricity sources. Note that the dollars per kilowatt figure used in both of the nuclear cases was calculated for large LWRs/advanced LWRs (ALWRs). There are likely distortions (both positive and negative) from using the same number for the much smaller FHR. A more in-depth analysis could discern these distortions, but this white paper assumes that the positives and negatives will roughly balance out. While the total project cost for an FHR is much less compared to a large ALWR, FHRs are still very costly compared to natural gas plants.

Table 2-2. Comparison of Construction and Overall Project Costs

Plant Type		Overnight Capital Cost		Nominal Capacity		Total Project Cost	
		$/kW	Rank	MWe	Rank	$, millions	Rank
Nuclear	***FHR***	***$5,339***	***14***	***200***	***11***	***$1,068***	***13***
	ALWR	*$5,339*	*14*	*2236*	*1*	*$11,938*	*19*
Coal	Advanced pulverized coal without carbon capture and sequestration (CCS)	$2,844	*7*	1300	*2*	$3,697	*17*
	Integrated gasification combined cycle (IGCC) without CCS	$3,221	*9*	1200	*3*	$3,865	*18*
	IGCC with CCS	$5,348	*16*	600	*4*	$3,209	*16*
Natural Gas	Conventional natural gas combined cycle	$978	*3*	540	*5*	$528	*10*
	Advanced natural gas combined cycle	$1,003	*4*	400	*7*	$401	*7*
	Advanced natural gas combined cycle with CCS	$2,060	*5*	340	*9*	$700	*11*
	Conventional Combustion Turbine	$974	*2*	85	*15*	$83	*2*
	Advanced Combustion Turbine	$665	*1*	210	*10*	$140	*3*
	Fuel cells	$6,835	*18*	10	*19*	$68	*1*
Renewables	Biomass	$3,860	*10*	50	*16*	$193	*4*
	Geothermal	$4,141	*11*	50	*16*	$207	*5*
	Metropolitan solid waste landfill gas	$8,232	*19*	50	*16*	$412	*8*
	Conventional hydropower	$3,078	*8*	500	*6*	$1,539	*14*
	Wind	$2,438	*6*	100	*13*	$244	*6*
	Wind offshore	$5,975	*17*	400	*7*	$2,390	*15*
	Solar thermal	$4,692	*12*	100	*13*	$469	*9*
	Photovoltaic	$4,755	*13*	150	*12*	$713	*12*

The LCOE should also be considered when combining both the initial construction costs and operating costs (fixed and variable). LCOE allocates the upfront costs over the expected life of the plant and adds that to operating costs – maintenance, overhead, fuel, interest, etc. This metric is often used as a one stop method for comparing projects with very different financial structures and output size. This metric still places nuclear behind coal and natural gas but not by as much just looking at the capital cost alone. Table 2-3 summarizes the LCOE for various energy sources.

2.1.2 Peak Power Production

The coupling of the FHR with a NACC allows the injection of natural gas after air compression and nuclear heating to enable the production of peak electricity. Natural gas injection raises the gas temperature, enabling more power output from the turbine and the steam recovery boiler. Because this is a nuclear-heated Brayton power cycle, the FHR will have several unique technical capabilities (U.S. Energy Information Administration 2011):

- *Variable peak electricity output*. Fossil-fueled gas turbines have narrow operating ranges to (1) avoid compressor surges and other instabilities and (2) ensure the required air-to-fuel ratio for combustion. In a nuclear Brayton power cycle, the hot compressed air is hotter than the auto-ignition temperature of natural gas or jet fuel. These fuels will burn at any ratio of fuel to air. The natural gas input is fully variable so the peak electricity output is fully variable. This variability allows the system to be used for fine tuning of electricity production to help control grid frequencies.
- *Fast response*. The response time for peak power production is under 50 msec—faster than any other power-generating technology. A conventional gas turbine experiences a time lag when natural gas is added before the power level increases, because the added heat input is used to increase the compressor speed. This factor is why a jet aircraft on takeoff has a lag between when fuel is injected into the engine and increased power. In the nuclear turbine, the compressor operates at a constant speed on nuclear heat and is unaware of the addition of natural gas. The time for the start of increase in power output depends on the flight time between fuel injectors and the first set of turbine blades. In utility systems, only expensive energy storage devices such as batteries and flywheels have faster response times.

The capital cost ($/kWe) for peaking capability is expected to be less than for other power-generating systems. Adding peaking capability does not change the design or cost of the expensive air compressor. Peaking capability does require incrementally larger turbines and steam recovery boilers. In both of those systems, the incremental capital costs of added capacity are low relative to stand-alone power peaking systems—including natural-gas-fired peaking power plants. While not part of this analysis, using hydrogen as the fuel for peaking operations in a low-carbon world is a long-term option.

Electricity demand varies by the hour, day, week (weekday and weekend), and season. In deregulated electricity markets, this variable demand results in variable prices. Figure 2-3 shows the number of hours per year electricity is sold at different prices in Southern California. At times of high demand, electricity prices are several times the average price of electricity. At times of low demand, electricity prices are negative; that is, the electric generator pays the grid to

Table 2-3. LCOE for Plants Entering Service in 2016 (Forsberg and Conklin 2007)

Plant Type	Capacity Factor, %	U.S. Average Levelized Costs, 2009 $/MWh				Rank by Total LCOE
		Levelized Capital Cost	Fixed O&M*	Variable O&M*	LCOE	
Advanced nuclear	90	90.1	11.1	11.7	113.9	11
Conventional coal	85	65.3	3.9	24.3	94.8	5
Advanced coal	85	74.6	7.9	25.7	109.4	9
Advanced coal with CCS	85	92.7	9.2	33.1	136.2	13
Natural gas with conventional combined cycle	87	17.5	1.9	45.6	66.1	2
Natural gas with advanced combined cycle	87	17.9	1.9	42.1	63.1	1
Natural gas with advanced combined cycle and CCS	87	34.6	3.9	49.6	89.3	4
Natural gas with conventional combustion turbine	30	45.8	3.7	71.5	124.5	12
Natural gas with advanced combustion turbine	30	31.6	5.5	62.9	103.5	8
Biomass	83	55.3	13.7	42.3	112.5	10
Geothermal	92	79.3	11.9	9.5	101.7	7
Hydro	52	74.5	3.8	6.3	86.4	3
Wind	34	83.9	9.6	0.0	97.0	6
Wind offshore	34	209.3	28.1	0.0	243.2	15
Solar thermal	18	259.4	46.6	0.0	311.8	16
Photovoltaic	25	194.6	12.1	0.0	210.7	14

* O&M=operations and maintenance

take electricity. This difference occurs because many power plants can't shut down for short periods of time. It is more economical to pay to get rid of electricity in the middle of the night so the plant can produce full power in the middle of the day with the high prices for electricity.

This variation in market prices for electricity creates the economic incentive to modify the FHR NACC to produce peak electricity using natural gas.[1] Initial studies (Beauvais 2012) examined such an FHR in California using that state's electricity and natural gas prices. These studies indicated about a 10% increase in profitability for an FHR with peaking capability versus an FHR producing baseload electricity.

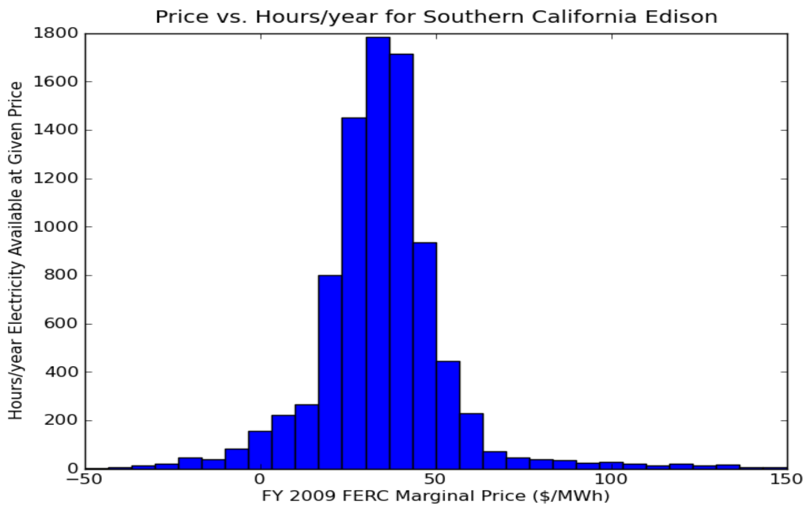

Figure 2-3. Electricity Prices Over 1 Year in Southern California

The incentives for peak power capability depend on the incremental capital cost to the FHR for that capacity, the cost of natural gas, the efficiency in converting natural gas into peak electricity, and future markets. Initial analysis indicates that the capital cost of the peaking capability will be less than any other power-generating technology. This potential creates large economic incentives to spend the added capital to have peaking capability. Because such peaking capability only operates a limited number of hours per year, the price of natural gas and the efficiency in converting to peak electricity are not the primary economic drivers.

An alternative understanding of peak power demands and markets can be obtained by looking at capacity duration curves. Figure 2-4 shows the number of hours different peaking units operate in the Midwest electrical grid.[2] This grid has about 100 GWe of capacity. Over 10 GWe (~10% of the total generating capacity) operates less than 100 hours per year. This peak electricity is extraordinarily expensive because the capital cost of the gas turbines is the primary cost. The fuel costs are almost irrelevant because of the short duty cycle. While the capacity factors for this peaking are very low, the capacity is necessary to prevent major electrical blackouts.

[1] The peak fuel option in the near term is natural gas. In the long term, it could be low-carbon biofuels or hydrogen.
[2] Information from the Midwest Independent System Operator website, Market Reports, at https://www.midwestiso.org/Library/MarketReports/Pages/MarketReports.aspx. Accessed May 24, 2012.

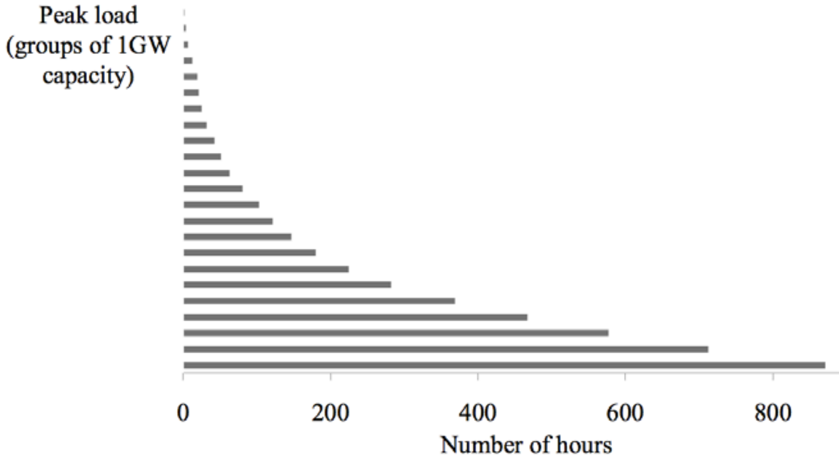

Figure 2-4. 2009 Operating Hours Per Year for Electricity Peaking Plants in the Midwest Independent Transmission System[1]

Recent years have seen a major effort to commercialize renewables; thus, it is important looking forward to see potential impacts on the electricity system and peak power demand and how it might affect commercialization of the FHR. For renewables, output depends on the sun or wind and is not necessarily connected to electricity demand. Recent studies (Denholm and Margolis 2007) examined the Texas electrical grid with different levels of photovoltaic electricity generation and determined the load curve. The results are shown in Figure 2-5. By the time photovoltaic provides 22% of the total electricity demand, over a quarter of the electrical generating capacity is peak capacity

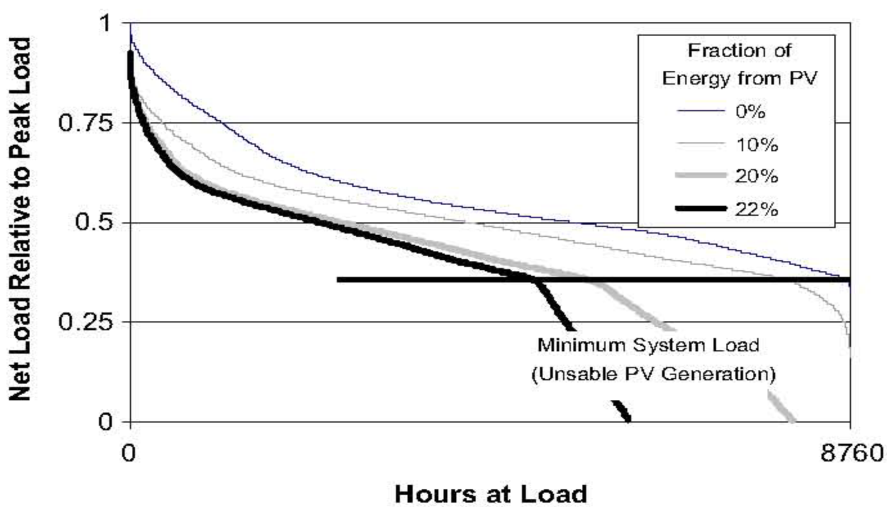

Figure 2-5. Electricity Load Curve Versus Percentage of Yearly Electricity from Photovoltaic

[1] From the Midwest Independent System Operator website, Market Reports.

that will operate less than a few hundred hours per year. The peak power market is large and – under most scenarios that include expansion of renewables – is expected to continue to grow as a fraction of all electrical generating capacity.

Peak electricity generation is a premium market that will be available to whatever technology has low capital cost for peak power production. The IRP's initial assessment is that the FHR has a large competitive advantage in this market, and thus the FHR's NACC will include peaking capability. While LCOE is a good benchmark for the economic competitiveness of a nuclear power plant's baseload generating capabilities, peaking competitiveness must be considered in a different light. As a provider of variable electricity, the plant's peaking efficiency (ratio of added peak electric output to added peak heat input) must be equal to or better than other providers. Simulations suggest that co-firing with a NACC can achieve 52% to 60% peaking efficiency. This efficiency bests not only peak-only plants but the steady-state efficiency (50% to 55%) for most natural gas combined-cycle plants as well. Thus, economically it would make sense for a plant owner to always operate with co-firing, because those extra megawatts would always be cheaper than natural gas could produce. This fact may make the "peaking" configuration of an FHR coupled to an open-air cycle the "baseload" case in a perfect deregulated market.

The capability to rapidly vary electricity output may enable the FHR to enter the power regulation markets not previously open to most electric generating technologies. Two electricity regulation markets (U.S. Department of Energy 2007) (Table 2-4) feature the value and price of electricity far above the price of baseload electricity and are often met by storage devices. Historically, generating technologies have not been considered to meet such demands because of slow response times. Depending on the response time of the open-air Brayton cycle, some of these markets may be accessible to the FHR.

Table 2-4. Premium Power-Grid Electrical Markets

Storage Technology Parameter	Regulation	Reserve Power for Grid Stability and Reliability
Capital cost, \$/kW*	700	300–1000
Total U.S. market potential, GW	30–40	70–100
Storage system power level	Up to 200 MW	10 MW to 1 GW
Discharge time at rated power	Seconds	0.2 to 2 hours
Capacity, storage time	Seconds	~ 2 weeks
Lifetime, years	20	40

*This is the capital cost for a stand-alone plant addressing these needs. If an FHR can meet these needs and is located where the needs exist, the numbers measure the added value of the FHR over a traditional baseload power plant.

Electricity regulation refers to the need to maintain constant frequency and voltage across the grid. Electric consumers turn equipment on and off with switches that operate in a fraction of a second. Electric generators can vary the power output over a period of minutes. If the demand and generation do not match, the system frequency and voltage change. If the changes are too great, both the customer's and the utility's equipment are damaged. Ultimately, the system fails, with a resultant blackout. The electrical system works because the electric grid averages demand over many customers so that the generators do not experience rapid changes in power demand. As a consequence, the larger electrical grids are more stable, have higher-quality electricity (proper voltage and frequency), and are more reliable than smaller electrical grids.

Because the stability of many electrical grids is decreasing, better methods are needed to ensure grid regulation and delivery of high-quality electricity. This situation is partly a consequence of the growing electricity demand associated with electronics. With traditional electrical loads, such as incandescent light bulbs, if the line voltage drops (insufficient power generation), both the electrical current and the power demand drop. This change provides time for the electrical generators to speed up or slow down as required to match power production with power demand. With many electronic devices, as the voltage drops, the device demands more current and the power demand goes up. The system provides less time for electrical generators to respond to the demand. The system becomes more prone to failure, and the quality of electricity decreases.

Figure 2-6 shows the variations in demand for 1 day on the Texas utility grid and the rapid variations in power demand over 1 hour.[1] This figure gives some perspective to the rapid change in electrical demand, partly caused by changing electrical load and partly caused by the changing frequency and voltages on the grid. There are strong incentives to reduce these short-term variations with better grid regulation.

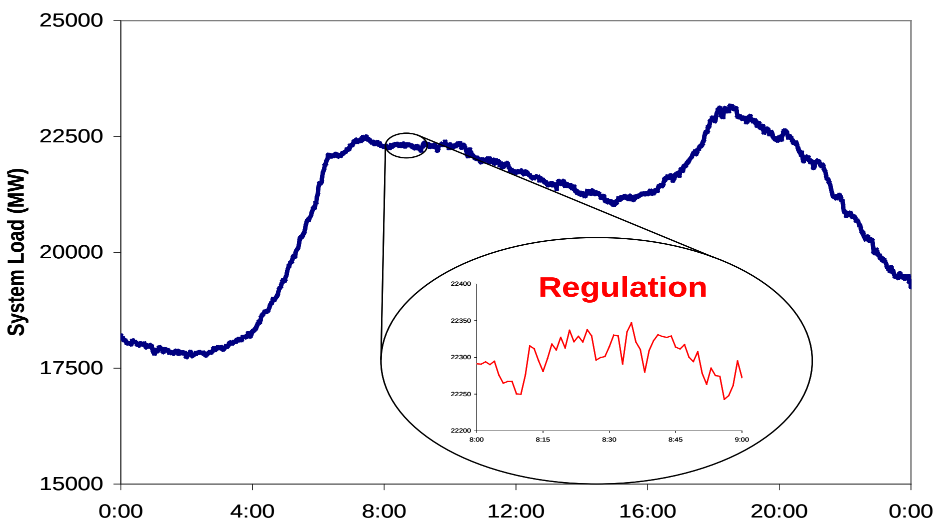

Figure 2-6. A Typical Electric Power Demand Load on the Electric Reliability Council of Texas Electric Grid Over 24 Hours on a Winter Day

Reserve power fills the need to provide generating capacity in the event that an electrical generator goes off-line for unexpected reasons. This need is currently met by putting on-line

[1] Electric Reliability Council of Texas, 1999 submission of data, available online at http://www.ferc.gov/docs-filing/forms/form-714/view-soft.asp. Accessed May 24, 2012.

additional power plants that run at partial load with the capability to produce more power if another unit goes off-line. This process represents expensive backup power. For this application, high-power output must be provided for periods of time, from tens of minutes to a few hours, the time required to startup another power generation system.

2.1.3 Process Heat Production

In many countries (Russia, Switzerland, etc.) reactors produce power and process heat in the form of steam for commercial or residential use. The FHR with the NACC has three unique characteristics that set it apart from other nuclear process heat systems (Forsberg 2012a):

- *No impact on design*. The design of the heat recovery boiler has no impact on the FHR design or the reactor's safety case, in contrast to all other systems that use nuclear heat for process heat applications and where changes to the reactor are required. Cold air enters the Brayton cycle and exits via a hot-air duct to the heat recovery boiler. The heat recovery boiler cannot impact reactor design because fluid flow (such as condenser water) does not return to the reactor. The heat recovery system can be designed to (1) produce regular steam or pressurized hot water at 30 MPa or (2) heat some chemical.

- *Reliable steam*. The steam recovery boiler is isolated from the reactor. There is the option of a separate natural-gas fired boiler to provide hot air to ensure steam supplies if the reactor is down for any reason. Therefore, steam reliability can be independent of reactor reliability. Steam reliability is a major concern of any user of process heat and has been a major barrier in the use of nuclear plants to provide process heat. The traditional strategy requires construction of multiple reactors—an option that often implies buying more steam capacity than required. With the FHR, this would not be required. It implies that even one FHR can sell steam with reliability determined only by the heat recovery steam generator.

- *Isolation of radioactivity from the reactor*. About 80 reactors have sold steam for district heating, water desalination, and various industrial applications (IAEA 2007) (Figure 2-7). Concerns about tritium and other radionuclides have resulted in using reactor steam as the heat source for a second steam generator to produce clean steam for sale. For an LWR, heating requires process steam temperatures of ~240°C—significantly below what goes to the steam turbine. The political concerns about using steam from nuclear power plants have been a major barrier for sale of steam from nuclear plants in the United States. The NACC with a large flow rate of air as the heat transfer media from the nuclear plant to the steam boiler prevents tritium transfer to the process steam—it's extreme isolation without the penalties of intermediate heat exchangers with concerns about heat exchanger leaks.

The 500+°C steam temperatures can meet most U.S. steam demands (Konefal and Rackiewicz 2008). Advanced versions of the FHR could meet other process heat markets but will require development of higher-temperature alloys. Process heat could also be used for shale oil recovery and hydrogen production.

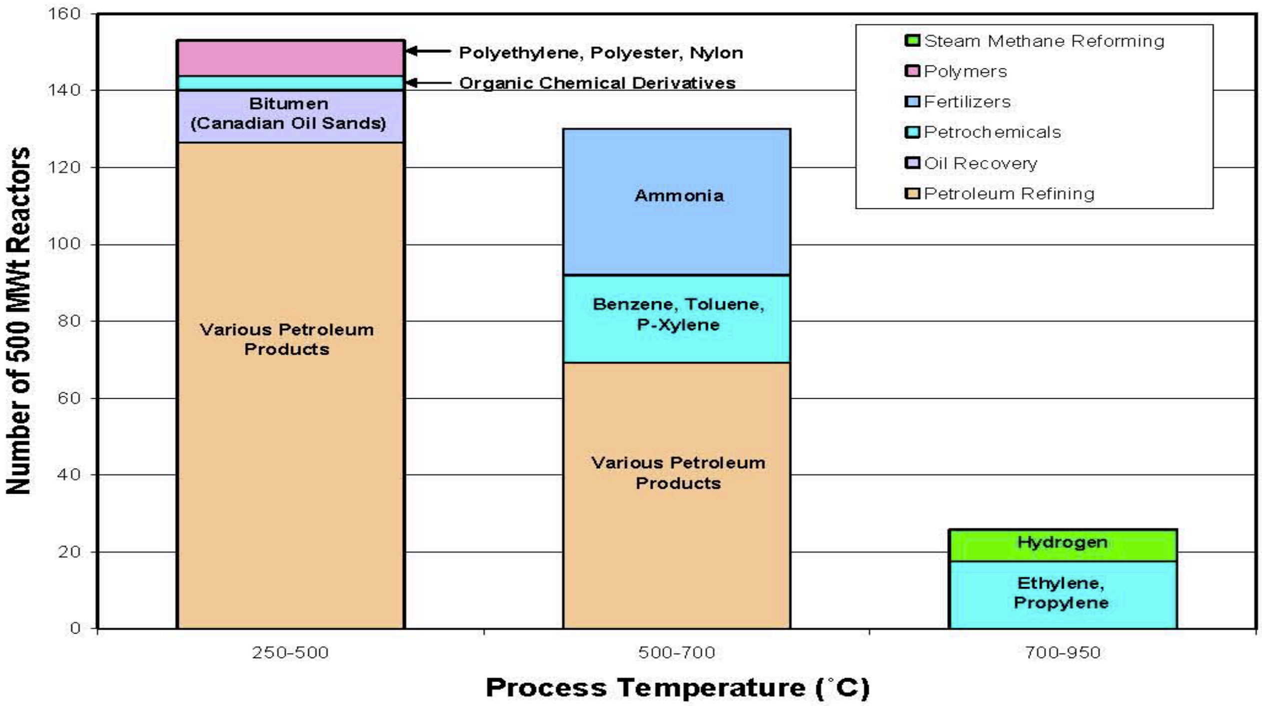

Figure 2-7. Number of 500-MWth Reactors at 85% Capacity Required to Meet U.S. Process Heat Demand Versus Process Temperatures (Forsberg 2012a)

2.1.3.1 Process Heat for Shale Oil Recovery

Most of the process heat market consists of multiple small markets with one exception—the potential shale oil market where the heat demand could ultimately be several hundred gigawatt-thermal.

Oil shales contain the largest fraction of the world's fossil fuel resources (Table 2-5). The United States has the largest reserves of oil shale in the world—resources that exceed total global oil production to date. These oil shales have the highest concentrations of fossil fuels in the world with deposits that can produce over a million barrels of oil per acre. The development of these resources would free the world from dependence on Mideast oil.

Oil shale contains no oil, rather it contains kerogen that on slow heating is converted to a high-quality light oil and various light gases. The oil shale must be heated to ~370°C for this conversion. Kerogen is the fossilized plant residue that is the precursor to oil and coal. Shell and other companies are developing in situ processes that use electricity in the form of resistance heaters or microwave heating to convert kerogen to oil. The slow heating process without oxygen creates high-quality light oils with high yields of gasoline, diesel, and jet fuel. In effect, the process is a variant of the refinery processes of thermal cracking and distillation with one important difference. Thermal cracking involves the heating of a hydrocarbon until it decomposes into oils, gases, and a residual carbon byproduct. Underground thermal cracking leaves the carbon residue sequestered underground—unlike in refineries where the resultant petrocoke from thermal cracking of heavy oils is burnt to provide energy. The underground heating moves the oil by vaporizing it with condensation on cooler rock (Figure 2-8).

Table 2-5. World Fossil Fuel Resources (Dusseault 2002; Marano and Ciferno 2001)

Feedstock for Liquid Fuel	% World Fossil Hydrocarbons	Heat Input as Fraction of Heating Value of Liquid Fuel
Oil	2%-3%	6%-10%
Heavy oil	5%-7%	25%-40%
Natural gas	4%-6%	~50%
Gas hydrates	10%-30%	
Oil shales	30%-50%	>30%[a]
Coal/lignite	20%-30%	>100%
Biomass	Annual*	To 40%

*Fraction of liquid fuel heating value that could be provided by biomass—a renewable resource versus the other fuels in this table.

One option for nuclear heat design is to build a machine that produces peak electricity and can be used for shale oil recovery (Robertson, McKeller, and Nelson 2011; Robertson 2011a; Robertson 2011b) in the form of high-temperature steam in pipes to heat the kerogen in situ. While the oil shale must be heated to ~370°C, the steam temperatures need to be near 500°C to provide the temperature drop to drive heat into the rock—achievable with the FHR steam system. Alternatively nitrate salts can be used as the heating fluid (Sabharwall et al. 2010), with the option of replacing the steam plant with a nitrate heating system. The heating requirements are about a quarter of the heating value of the products (shale oil and some gas), thus there are major incentives to use nuclear heat to avoid burning much of the product (usually the light gases), reduce greenhouse gas emissions, and stabilize the grid. The IRP has not completed the much more complex analysis required for this commercialization strategy.

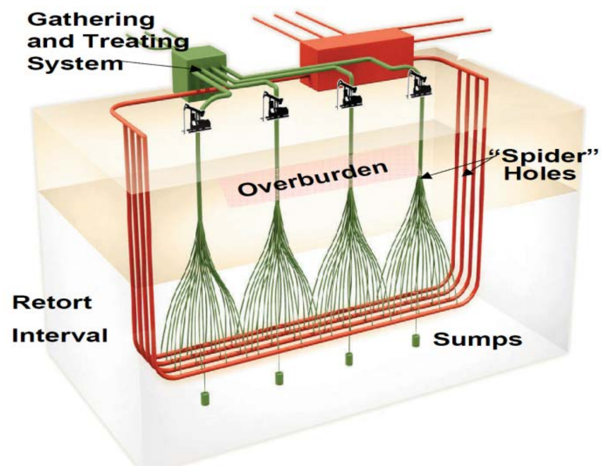

Figure 2-8. Heating System for Shale Oil Recovery

A single 600-MWth high-temperature reactor could operate for its 60-year lifetime, and the longest distance to a well head for heat injection would be less than 2 miles. The U.S. imports about 10 million barrels (42 gallons per oil barrel) per day, which equates to a total thermal output of 200 GW, which could produce sufficient shale oil to replace total oil imports.

Unlike other industrial processes, heating rock is a slow process requiring months to years because of the low thermal conductivity of the rock. Because of this slow heat transfer, it is not required to heat the rock continuously—one can choose to heat the rock primarily at night and use most of the process heat during the day for production of electricity (Forsberg 2012b). This characteristic enables the use of baseload nuclear power for simultaneous oil recovery and variable electricity production. In the U.S., the shale oil deposits are in the west where there are large wind and solar resources. A hybrid system could provide the variable electricity to backup renewables and the market for low-cost electricity would create a minimum price for electricity.

The capital cost penalty for variable electricity production is small. The steam turbine and generator are not fully utilized at times of low electricity demand, but this is a relatively small fraction of the total capital costs. The capital investment in underground steam heating piping is only about 20% of the total capital investment. In a shale oil production system, different blocks of oil shale will be heated sequentially, with different blocks in different stages of production. If the average heat load to one block is reduced because peak electricity is also being generated, the production of oil in that block will be stretched over time. The total kilowatt of heat per meter of steam pipe that must be delivered over its lifetime is the same in both cases, but the average kilowatt power level is lower. The particular steam pipe operates for a longer period of time. For any set oil production rate with variable electricity production, this difference implies added blocks of oil shale with steam heating pipes must be developed earlier—implying earlier investment in underground steam heating piping than in a system that only produces shale oil. Some of the investments in underground development must be moved forward in time—a relatively small effect because the underground piping is a small fraction of the total costs.

A limited assessment indicates that this system has the lowest environmental impact of any method using fossil fuels to produce gasoline, diesel, and jet fuel. This fact makes it a preferred fossil-fuel transition option to a low-carbon world. It enables replacement of fossil fuels for variable electricity production. It has the lowest greenhouse impact of any fossil fuel. The conversion of a feedstock into gasoline and diesel requires heat and hydrogen, which in traditional processes results in carbon dioxide releases to the air. Nuclear shale oil is underground refining where the carbon residue with impurities remains underground. In effect the process results in carbon sequestration of the byproducts of the production and refining process, whereas all the other options release large quantities of carbon dioxide during production or refining. Furthermore, the sequestered carbon is in the form of char where there is a much higher assurance of long-term carbon sequestration than with carbon dioxide sequestration. Lastly, U.S. oil shale deposits per unit area are the most concentrated fossil fuel resources on earth, implying fewer kilometers of drill pipe, pipeline, and disturbed land per unit of production.

2.1.3.2 Process Heat for Hydrogen Production

The U.S. hydrogen demand is about 9 million tons per year for upgrading of crude oil to gasoline and diesel, production of fertilizer, and many smaller chemical applications. Hydrogen can be produced from water via high-temperature electrolysis (heat and electricity) and several thermochemical processes (heat). About 150 to 200 reactors each with a capacity of 500 MWth would be required to make this quantity of hydrogen. This long-term market is not being considered at this time for the first-generation FHR for several reasons:

- *Characteristics of natural gas*. Most of this hydrogen is made from natural gas. Natural gas prices are low. More importantly, natural gas is primarily methane (CH_4). The chemical energy to break the hydrogen away from the methane is less than the chemical energy to break hydrogen away from water. Economics will favor the FHR replacing natural gas for process heat applications before replacing natural gas for hydrogen production.
- *Process availability*. All of the processes to produce hydrogen using high-temperature heat are either in the research or pilot plant stage of development. The market depends on the success of the development programs. In contrast, all of the previously described heat markets are near term.
- *Reactor requirements*. Many of the processes require very high-temperature heat in the range of 800°C to 900°C. This need implies a special FHR design and development of advanced materials for heat exchangers.

Analyzing the market for process heat hydrogen production depends on the specific electrical grid. In an electrical grid where a large fraction of the generating capacity is supplied by hydro (e.g., Quebec, Sweden), the value of these other capabilities is limited. For a grid with limited hydro or large quantities of wind or solar electricity, the FHR economics would be expected to be favorable because of the reactor's capability to address short periods of demand for large quantities of peak power and replace grid stabilization devices from capacitors to batteries.

The goal is to economically provide variable electricity or steam to the customer based on his demands. That future in a low-carbon world may require rethinking how electricity is produced.

2.1.4 Other Implications

The FHR's NACC can provide high-reliability process steam—including steam during refueling outages. This capacity cannot be achieved with other designs of nuclear reactors providing process heat without multiple reactors per site. In the electrical market, depending on the specific grid, the FHR peaking and grid regulation capabilities could result in a credit exceeding a $1,000/kWe of installed capacity. Equally important, the characteristics of the system match what is needed for a future low-carbon nuclear-renewable grid.

The market feeds back into the design by defining FHR power levels, required peak temperatures to match the gas turbine, intermediate heat exchanger design, the need to control tritium because it will not be isolated in an open-air power cycle, and a variety of other features of the reactor. The market also defines the customer—it is no longer just baseload electricity.

2.2 Alternative Markets

A variety of specialized markets are driven by different requirements with different economic criteria. One such market is described herein. In the context of a path forward, these markets are important in two different contexts: (1) alternative customer base and (2) requirements for the FHTR. If the FHR is to be used for different missions, the FHTR may be required to test different fuels and coolants over its lifetime. Therefore, a more general-purpose FHTR may be needed rather than one with a very narrow testing mission for a specific design.

There is a potential small reactor market for Antarctic bases, remote mining sites, container ships, military bases, and other applications. The reactor requirements are different. In many cases, the reactor must be flown into the site and ultimately flown out of the site. The cost of electricity and heat from conventional sources is often very high, making nuclear energy potentially attractive. However, there are several constraints:

- *Security*. Security requirements for power reactors imply large and expensive security forces. If large security forces are required for reactors with these missions, the economics are likely to be unfavorable. The remote locations imply very high costs for security personnel and slow response by outside security forces.
- *Accidents*. Accidents, including malevolent events, in remote locations imply no rapid response. Furthermore, in many cases the local residents may not able to evacuate.
- *Major structures*. For many of these applications, it is not feasible to use massive concrete silos or other building structures as part of the safety or security systems. Construction of such civil works at remote locations is extremely expensive. The small size of an FHR separates these reactors from what are generally defined as small modular reactors such as those proposed by Westinghouse, Babcock and Wilcox, and several other companies.
- *Power cycle*. Off-grid applications implies that the reactor must meet the variable load—there is no grid to average demands. The FHR described above with peak power and grid stabilization capability has the potential to meet these goals.

A possible design may meet these requirements: a transportable fluoride-salt-cooled, high-temperature reactor (TFHR)—reflecting the potential need to transport such a reactor to a remote site by air and its potential use in the merchant marine. The design is based on (1) a new fuel from the DOE accident-tolerant fuel program, (2) a different coolant, and (3) the NACC—the design basis system for the FHR.

The safety goal for a TFHR is that the consequences of any accident or malevolent event (such as attack with explosives) will be limited to within a few tens of meters of the reactor site. If this distance can be achieved, it can address many of the above constraints. If these goals are defined as TFHR requirements, the fuel and coolant must be the primary mechanisms to ensure safety under extreme events—not the entire system including reactor vessel, support systems, and containment.

To meet such extreme requirements, fuel integrity under accident conditions must meet two challenges: decay heat and explosive loadings. The leading fuel candidate to address these requirements is the silicon-carbide (SiC)-matrix coated-particle fuel, where SiC replaces the graphite in the baseline FHR. SiC has several relevant characteristics: (1) it is highly radiation resistant and a leading candidate for the inside walls of fusion machines; (2) the carbon density in SiC is similar to graphite, and SiC has a very low neutron cross section that makes it a reasonable neutronic replacement for graphite; (3) its peak temperature capabilities match graphite; and (4) it is used in many types of armor—it's a very tough material.

This advanced fuel is being developed at Oak Ridge National Laboratory (ORNL) as part of the DOE accident-tolerant fuel program, but limited work has been done. This fuel was originally proposed as a new matrix fuel for accident-tolerant LWR fuels but more recently has been proposed for the FHR (Forsberg et al. 2012).

Based on the evidence to date, the fuel may meet the above requirements in the following ways:

- *Decay heat*. All reactors produce decay heat. If the fuel is not cooled, its temperature will increase. The coated-particle fuel has failure temperatures near 1,600°C. If the integrity or operability of decay heat removal systems can't be ensured, conduction is the only method to remove decay heat in an accident to prevent fuel failure by overheating the fuel. No detailed calculations have been done, but based on analysis of various high-temperature reactors, the upper size of the reactor will likely be limited to several hundred megawatts.
- *Explosive loadings*. The reactor must withstand extreme events such as explosive loads from external assault (worst case). Limited information suggests that the FHR can meet this requirement. Explosives break objects and accelerate the debris to high speeds. The fuel is a high-integrity coated-particle fuel inside the SiC matrix. For such extreme events, the goal is for the matrix material to absorb the energy, the fuel particles to survive, and dispersal limited to a few tens of meters from the reactor. The goal is a fuel sufficiently robust that the safety case changes and thus changes the requirements for expensive security, response plans, and other factors that make small reactors for remote sites so expensive.

To meet the above goals also requires a relatively non-toxic coolant. The leading candidate is a lithium zirconium fluoride salt that is isotopically separated ^{6}Li and isotopically separated ^{90}Zr. Considerable work has been done on isotopically separated zirconium for use in water-cooled reactors. Urenco (uranium and stable isotope enrichment) and others believe this is achievable.

It is too early to determine the viability of such advanced concepts. What is important is that the FHR is a class of reactors with a set of common features which open new options—options that could fundamentally change the role of nuclear energy. However, such options require modified fuels and coolants and should be considered if an FHTR is to be built and designed to enable testing of such advanced concepts in addition to the baseline concept.

2.3 Path Forward

The potential for different FHR markets relative to other types of nuclear power plants has major implications on the path forward. The basis for judging economics is not LCOE—a metric designed for baseload plants but not applicable to a plant that also has peak electricity and grid stabilization capabilities. The different markets also impact research, design, and the FHTR. Decisions on reactor output, peak temperatures, intermediate loop design, tritium control systems, and other components are directly driven by the power cycle decisions that are in turn driven by market demands.

The potential for different types of FHRs has major impacts on the design of the FHTR, resulting in two strategies. The first is an FHTR designed for a specific reactor concept. It has the advantages of lower costs and significantly shorter schedules. The second option is an FHTR that can test a variety of fuels and coolants over a period of decades. Because of these specific possibilities, the IRP is investigating alternative designs for the FHTR.

Figure 2-9 provides a top-level schedule. The distinguishing feature is the broad set of market drivers, which leads to an FHTR that in turn leads to a commercial prototype reactor. However, because of the potential for multiple markets with different requirements, the schedule also implies an FHTR designed to serve multiple missions over its lifetime.

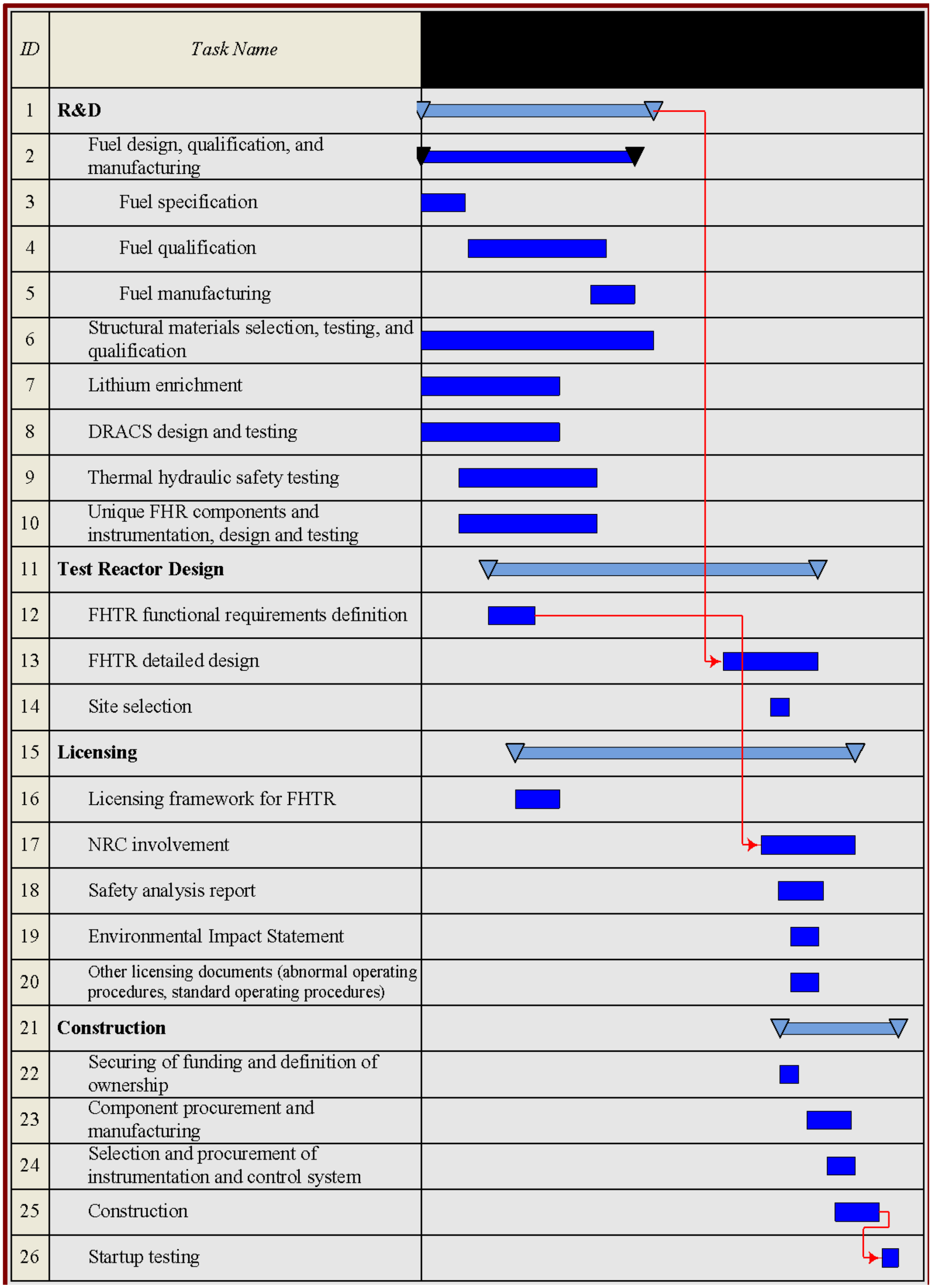

Figure 2-9. FHR Developmental Schedule

3 Preparatory Activities for FHTR Design and Construction

This chapter reviews activities needed to support FHTR design and construction, including the potential critical path issue of a new capability to enrich lithium. This chapter also reviews structural materials, fuel, and unique components as well as component and system testing. In addition, the chapter lays out activities associated with safety analysis and licensing.

3.1 FHTR Lithium Enrichment

The baseline primary coolant for the FHR is a 2:1 mixture of lithium fluoride (LiF) and beryllium fluoride (BeF_2). For nuclear applications, this coolant, called flibe ($2LiF\text{-}BeF_2$), is enriched to 99.995% in ^{7}Li. In this specification, flibe has a lower neutron absorption cross section than any other candidate salt coolant while maintaining other desirable properties. Natural lithium is 92.5 wt% ^{7}Li and 7.5 wt% ^{6}Li. It is critical that nearly all ^{6}Li be removed because the isotope has a large thermal neutron absorption cross section (about 1,000 barn at 0.1 eV). Neutron absorption in ^{6}Li is bad for neutron economy, and at higher concentrations (above approximately 0.01%) the effect of coolant neutron absorption exceeds its moderating capacity, and the coolant temperature reactivity feedback becomes positive. Additionally, nearly all of this 1,000 barn cross section in ^{6}Li is a result of a reaction (Eq. 3-1), where ^{6}Li absorbs a neutron to produce tritium (^{3}H) and an alpha particle (^{4}He). As a result of the high operating temperature of the FHR (between 600°C and 700°C), this radioactive tritium will readily diffuse through metals such as heat exchangers. By enriching the flibe in ^{7}Li, tritium production from ^{6}Li is reduced. However, small quantities of new ^{6}Li are produced by neutron interactions with beryllium. Thus, the ^{6}Li is never fully burnt out, and an equilibrium is established between ^{6}Li production from beryllium and destruction via neutron absorption and subsequent tritium production.

$$ {}_{0}^{1}n + {}_{3}^{6}Li \rightarrow {}_{1}^{3}H + {}_{2}^{4}He \qquad (3\text{-}1) $$

The following subsections discuss existing supplies of enriched ^{7}Li, the current market, and a potential enrichment method.

3.1.1 Existing Enriched ^{7}Li Supplies

At the Y-12 National Security Complex in Oak Ridge, Tennessee, between 1954 and 1963, the United States produced 442.4 metric tons of enriched ^{6}Li, mostly for tritium production for thermonuclear weapons.[1] In the process of producing enriched ^{6}Li, vast amounts of tails enriched in ^{7}Li (depleted in ^{6}Li) were generated. As summarized in Table 3-1, the stores of these tails are located at the Portsmouth Gaseous Diffusion Plant in Portsmouth, Ohio, and the Y-12 complex. Natural lithium is also stored at the K-25 Plant in Oak Ridge. The tails and the unused lithium are stored as lithium hydroxide monohydrate ($LiOH\text{-}H_2O$).

[1] From "Declassification of the Quantity of Enriched Lithium Produced at the Y-12 Plant in Oak Ridge, Tenessee," available online at https://www.osti.gov/opennet/forms.jsp?formurl=document/press/pc23.html. Accessed May 24, 2012.

Table 3-1. Summary of Tails (depleted in ^{6}Li) Enriched in ^{7}Li and Ununsed Lithium in the United States[1]

Location	Mass of Tails, metric tons	Approximate Tails Composition, wt% ^{7}Li	Mass of Unused Natural Lithium, metric tons
Portsmouth, Ohio	30,909	96%-99%	-
K-25 Plant	-	-	10,455
Y-12 Complex	8	96%-99%	12

Work is ongoing at ORNL to better characterize the lithium isotopic concentrations in these inventories (Grogan and Mihalczo 2012). DOE recently sold much of the unused and tails lithium, and the material is being transferred to the buyers.[1] While the tails compositions may not be 99.995% ^{7}Li, some of them are at least 99 wt% ^{7}Li, compared to natural lithium being 92.5 wt% ^{7}Li. Additional separations could achieve the desired ^{7}Li enrichment. Because the tails are already enriched in ^{7}Li, the work required for enrichment would be less than if starting with natural lithium. However, increasing the enrichment from 99 wt% ^{7}Li to 99.9% or from 99.99% to 99.999 % may still be a significant hurdle (Ingersoll et al. 2005).

The DOE retains quantities of ^{7}Li-enriched flibe remaining from the Molten Salt Reactor Experiment (MSRE) conducted at ORNL in the 1960s. Three batches of ^{7}Li-enriched flibe exist in this inventory and are summarized in Table 3-2: 2,250 kg of coolant salt, 4,650 kg of fuel salt, and 4,290 kg of flush salt. Both the fuel salt and the flush salt contain ZrF_4, actinides, and fission products. While these impurities could be removed from the salt, removal may prove to be prohibitively expensive. The 2,250 kg of coolant salt does not contain these impurities.

Table 3-2. ^{7}Li-Enriched Flibe in the United States (Massie et al. 2012)

Type of ^{7}Li-Enriched Flibe	Quantity, kg	Significant Impurities
Coolant salt	2,250	None
Fuel salt	4,650	ZrF_4, actinides, fission products
Flush salt	4,290	ZrF_4, actinides, fission products

3.1.2 Suppliers and Market for Enriched Lithium

Greater than 400 kg of enriched ^{7}Li is used per year worldwide as LiOH additions to pressurized-water reactors (PWRs) for water chemistry control, and there is no U.S.-based source (Massie et al. 2012). The demand for ^{7}Li has increased as new PWRs are built around the world. Additionally, with the construction of several fusion machines (such as the International

[1] From "DOE Environmental Management (EM) - Materials in Inventory Lithium Report - Chapter 2: Inventory," available online at http://www.em.doe.gov/bemr/BEMRPages/appxbc.aspx. Accessed May 24, 2013.

Thermonuclear Experimental Reactor), the need for enriched ^{6}Li (which generates enriched ^{7}Li tails) has also increased. Isotopically separated ^{6}Li has a variety of other applications, including light-weight components for space applications and advanced batteries. Consequently, a number of different groups and industries are potential markets for isotopically-separated lithium isotopes. This large market creates incentives to develop low-cost methods to separate lithium isotopes. Currently, several U.S.-based providers (such as Ceradyne, Inc.) bring ^{7}Li from overseas, namely China. The Chinese lithium enrichment procedure is based on column extraction using LiOH and mercury, a process that has been banned in the United States.

3.1.3 Possible Lithium Enrichment Methods

ORNL developed three methods for lithium enrichment by chemical separations in the 1950s. These methods include the column-exchange (COLEX) process, the electro-exchange process, and the organic-exchange process. Each of these methods was based on concentrating ^{6}Li in a mercury phase. Nearly all of the enriched lithium produced in the U.S. was via the COLEX process. The toxic nature of mercury and the fact that several hundred tons of mercury were released to the environment resulted in a ban on mercury-based extractions in the United States.

ORNL also evaluated a variety of aqueous extraction methods in the 1960s (Manning 2010; Lee and Begun 1959; Lee 1960; Lee 1961a; Lee 1961b). The majority of these methods were based on COLEX chromatography utilizing different stationary-phases and mobile-phase eluents. Generally, the ^{6}Li was concentrated on the resin in the column (stationary phase), and the ^{7}Li was concentrated in the aqueous eluate (mobile phase).

One factor that can be used to describe the efficiency of the separation after a single stage is the separation factor, α. The separation factor is the ratio of two distribution ratios as follows:

$$\alpha = \frac{\left({}^6Li / {}^7Li \right)_{resin}}{\left({}^6Li / {}^7Li \right)_{aqueous}} \qquad (3\text{-}2)$$

A higher separation factor indicates a better separation. The separation factor is not the only measure of performance, as the reaction kinetics (the rate of ion exchange) should be reasonably fast, so that a long elution time is not required to achieve the separation factor. As an example of the 1960s experiments, using Dowex 50 resin (a cross-linked copolymer of sulfonated polystyrene-divinylbenzene), Lee and Begun measured separation factors for two different mobile phases and varying resin cross-linking (Manning 2010). These data are summarized in Table 3-3.

Table 3-3. Summary of Dowex 50 Experiments Determining the Effect of Cross-Linking on the Separation Factor, α (Manning 2010)

Degree of Resin Cross-Linking	Eluent and Eluent Concentration, N	α
4	0.1 HCl	1.0010
8	0.3 HCl	1.0016
12	0.3 HCl	1.0027
16	0.3 HCl	1.0037
24	0.3 HCl	1.0038
2	0.25 NH_4Cl	1.0006
4	0.25 NH_4Cl	1.0010
8	0.25 NH_4Cl	1.0018
10	0.25 NH_4Cl	1.0023

In addition, experiments at ORNL investigated the separation afforded by ion exchange resins having different functional groups (e.g., carboxylate resins vs. aluminosilicate resins vs. zirconium phosphate, etc.). Some of these data are summarized in Table 3-4. In a later paper published in 1969, Lee concludes that the separation factors in ion exchange systems are "too small to be practical" (Lee 1969). In the same paper, Lee analyzes an extraction chromatography method, but this approach only achieved a single-stage separation factor of 1.003. Lee concluded that the prospects for separating lithium isotopes on a large scale by extraction chromatography were "not very promising" (Lee 1969).

More recently, work in the 1980s showed that ion-exchange chromatography and liquid-liquid extraction using crown ethers could achieve separation factors ranging from 1.05 to 1.06 (Heumann 1985; Lee 1969). Thus, this method has separation coefficients at least one order of magnitude higher than those reported for the methods studied by Lee. A summary of the separation factors and distribution coefficients $\left(D_{organic/aq.} = {Li_{organic}}/{Li_{aq.}}\right)$ for five different crown ethers used in aqueous-solvent extraction is provided in Table 3-5. A recent design project at the UCB showed that lithium enrichment based on liquid-liquid extraction using crown ethers would have reasonable economics and be able to produce large quantities of enriched lithium (Ault et al. 2012).

Table 3-4. Summary of Lithium Isotope Separation on Different Ion Exchange Resins (Lee 1961a)

Resin	Eluent and Eluent Concentration (N)	α
IRC-50	0.1 NH_4Cl	1.0020
CS-100	0.1 NH_4Cl	1.0026
Zeo-Karb	0.1 NH_4Cl	1.0023
Decalso	0.25 NH_4Cl	1.0047
Zirconium phosphate	0.25 HCl	1.0016
Bio-Rex 62	0.25 NH_4Cl	1.0005
Bio-Rex 63	0.25 NH_4Cl	1.0003
Dowex 50 (16 x cross-linking)	0.25 NH_4Cl	1.0023

Other alternatives to mercury-based and ion-exchange methods are distillation, thermal diffusion, and electromagnetic separation (Ault et al. 2012). ORNL continued to operate Calutrons for electromagnetic isotope separation until 1998 when DOE placed the reactor in standby.[1] Some work has been carried out to investigate the use of atomic vapor laser isotope separation (AVLIS) to separate isotopes of a number of different elements including lithium (Newman 1982; Scheibner 2004). Unlike uranium metal, which melts at 1,132°C and boils at ~3,900°C, lithium metal melts at 180°C and boils at 1,342°C, and thus is readily vaporized to produce an atomic vapor for laser isotope separation. The design project at UCB also considered AVLIS as a possible route to highly enriched ^{7}Li and concluded that the approach is also a commercially viable option (Ault et al. 2012).

[1] Information from the National Isotope Development Center, "Isotope Production Methods", available online at http://www.isotopes.gov/sites/production.html, accessed May 24, 2013, and Oak Ridge National Laboratory's "Isotope Development Group", available online at http://www.ornl.gov/sci/nsed/group_fcid_isotope.shtml, accessed May 24, 2013.

Table 3-5. Separation Factors and Distribution Coefficients at Equilibrium for Liquid-Liquid Extraction (Nishizawa et al. 1988)

Crown Ether	Separation Factor (α) at 0 °C	Distribution Coefficient (D) at 0 °C
12-crown-4	1.057	2.0E-5
Benzo-15-crown-5	1.042	7.1E-3
Lauryloxymethyl-15-crown-5	1.041	8.1E-3
Tolyloxymethyl-15-crown-5	1.043	5.0E-3
Dicyclohexano-18-crown-6	1.024	2.8E-2

3.2 FHTR Structural Materials

This section provides an overview of the options for FHTR structural materials (e.g., 316 stainless steel or SS, Alloy N, and SiC) and discusses the critical path issues for their application in a FHTR [e.g., demonstration of appropriate capability for coolant chemistry and corrosion control and American Society of Mechanical Engineers (ASME) Section III code qualification, respectively].

3.2.1 Metallic Structural Materials

FHRs operate at higher temperatures than liquid metal reactors (LMRs), as shown in

Figure 3-1. A key issue for the design of FHRs is that key metallic components must operate at temperatures where creep occurs and where time-dependent behavior must therefore be considered. These considerations greatly increase the complexity of the component design and require extensive test data. Currently, information on time-dependent creep deformation is available for only a small number of materials according to information from the third FHR workshop. Under joint work sponsored by the DOE and the U.S. Nuclear Regulatory Commission (NRC), a new Division 5 for Section III of the ASME Boiler and Pressure Vessel (BPV) Code is being developed. It covers rules for the design, fabrication, inspection, and testing of components for use in high-temperature nuclear reactors. This new division makes several significant improvements that are relevant to FHRs.

Table 3-6 shows the candidate materials that have extensive property databases and are already included in ASME Section III (except Alloy N, which currently has only ASME Section VIII qualification). It is desired that the primary pressure boundary use materials with existing ASME code qualification to enable more rapid development of an FHTR and commercial prototype reactor, given that FHR fuel can also be developed and qualified in an accelerated time frame (see the third FHR workshop white paper for more information).

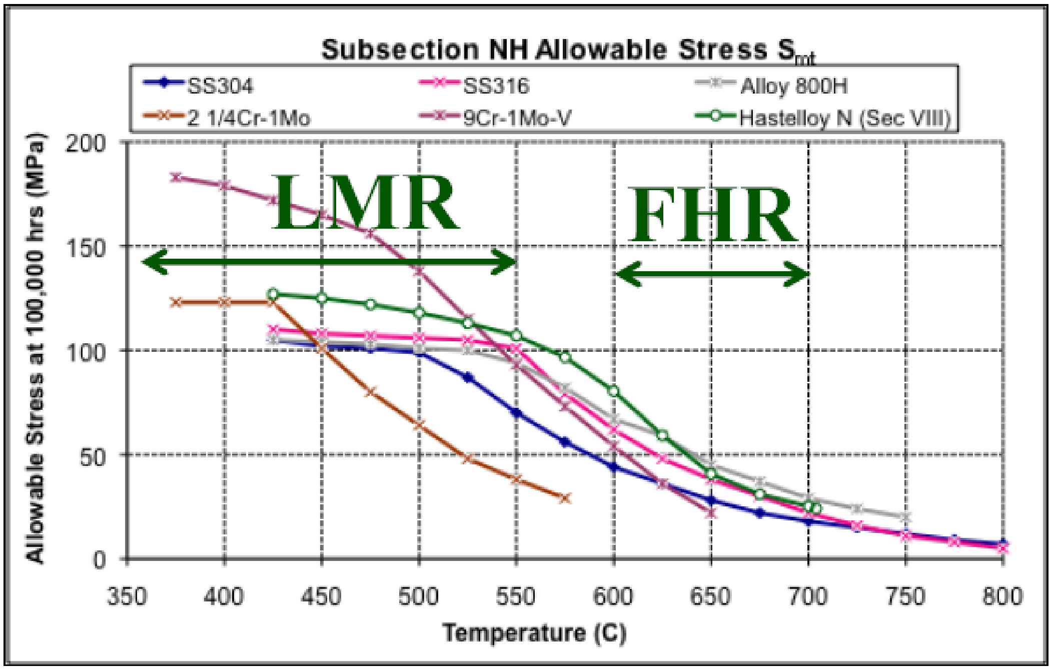

Figure 3-1. ASME Code-Allowable Stresses for Several Structural Materials

Table 3-6. FHR Constituents – Structural Materials

Component	**Candidate Materials for IRP**
Metallic Components	
Pressure vessels and piping	316 SS, Alloy N, Alloy 800H (clad), Alloy 617 (clad)
Heat exchangers	Alloy N, 316 SS, Alloy 800H (clad)
Core internal structures	316 SS, Alloy N, Alloy 800H (clad)
Ceramics	
Reflectors	Graphite
Core internal structures	Graphite, baked carbon, carbon fiber-reinforced composites (CFRC), SiC/SiC composites
Building Structures	
All	Steel-concrete composites

In the fabrication of reactor vessel and heat exchangers, 316 SS and Alloy N are two primary candidate materials. Both alloys have advantages and limitations. 316 SS has extensive experience for nuclear applications and good tolerance for neutron irradiation, and it is ASME Section III code-qualified for use at temperatures up to 816°C (1,600°F) in Subsection NH and Code Case N-201-5, which comprise extensions to Subsections NB and NG, respectively. These parts of the ASME code cover Type 316H in terms of high-temperature strength, creep, and creep-fatigue effects up to a design life of 300,000 hours.

However, the corrosion resistance in flibe for FHRs is unknown and needs further study, though 316 SS is reported to have good corrosion resistance in clean flibe and when beryllium is used as a redox. There is no experience using 316 SS in an FHR reactor vessel. Alloy N was successfully used in the MSRE, and it has outstanding corrosion resistance. But Alloy N has relatively poor performance under neutron irradiation. Based on MSRE experience, Alloy N can be used at temperatures less than or equal to 704°C in low neutron flux regions, generally less than 1 DPA. And Alloy N is not codified into BPV Code Section III - Rules for Construction of Nuclear Power Plant Components, particularly not into Subsection NH - Class 1 Components in Elevated Temperature Service (Ault et al. 2012). To qualify Alloy N for Subsection NH, a lot of time-dependent creep-rupture, creep fatigue, and other properties will be required. The process of ASME code qualification is costly and time consuming. It was suggested at the third FHR workshop that in addition to DOE, any parties interested should work together to create a code case for Alloy N, especially if China is involved.

In the selection of alloy for the reactor vessel and heat exchanger, it was agreed during the third FHR workshop that the reactor vessel and heat exchanger should be constructed using the same alloy to avoid galvanic corrosion. And it is also important to perform the creep tests in the relevant environment, because creep can be influenced strongly by the environment of the sample.

To combine the advantage of 316 SS and Alloy N, using 316 SS or 800H with Alloy N or pure nickel cladding is also possible. Many existing LWR pressure vessels use this strategy to reduce corrosion. This approach may be an intermediate solution to avoid the time-consuming ASME code-qualification process. However, the adhesion/compatibility at high temperature between cladding and substrate must be investigated.

3.2.2 Graphite and Ceramic Composites

Like HTGRs, FHRs make extensive use of graphite as a structural material and neutron moderator. The baseline FHR design assumes limited use of CFRC and SiC/SiC composite structures. The baseline design assumes the use of CFRC for the core barrel assembly, that together with the reactor vessel creates the downcomer which guides flow to the core inlet; and the use of SiC/SiC composites for structures in high neutron dose rate regions of the core, particularly for shutdown rod channels (see the third FHR workshop whitepaper for more information).

3.2.2.1 Graphite

Under neutron irradiation during FHR operation, there will be dimensional changes in the graphite structures. The dimensional change depends on the temperature and dose rate. When the neutron dose rate is lower than the critical point or turnaround, there is densification or dimensional shrinkage of graphite. After the turnaround, graphite will grow in the crystallographic c-direction. If the graphite is isotropic, as in candidate nuclear graphite IG-110, it will grow uniformly in all directions. The strength of graphite also depends on the neutron irradiation. Before the turnaround, the strength of graphite increases. But once turnaround is reached, graphite loses strength dramatically. According to researchers from ORNL, nuclear graphite can be used up to 25 DPA.

FHRs use a lot of graphite, including in inner and outer reflectors and fuel pebbles in a graphite matrix. From UW's experience, graphite will accelerate the corrosion of 316 SS and Alloy N in flinak salt (Sellers et al. 2012). An understanding of the compatibility of graphite with flibe and other structural alloys (316 SS and Alloy N) in FHRs is needed.

3.2.2.2 Ceramic Composites

It was agreed during the third FHR workshop that it is very desirable to use C/C and SiC/SiC composites to fabricate the core barrel and control rod housing, respectively. Because CFRC are far more susceptible to irradiation-induced degradation than SiC/SiC, C/C composites are the primary candidates for the core barrel with low neutron doses. The fiber in C/C composites is very important in controlling dimensional change. While the balanced weave composite is orthogonally isotropic, a significant anisotropic dimensional change occurrs under irradiation. Dimensional change is dominated by fiber dimensional change. Similar to graphite, and understanding of the compatibility of C/C composites with flibe and other structural alloys is needed. SiC/SiC composites have clear advantages for high-neutron-dose applications such as control rod housing. The cubic SiC crystal undergoes an isotropic dimensional change under irradiation, with the dimension change saturating at a modest level for the temperature range of interest for high-temperature reactors. Moreover, properties such as hardness, elastic modulus, and strength all undergo only modest changes and saturate at the same rate as the swelling. Based on experience of ORNL researchers, if the compatibility or good corrosion resistance of SiC/SiC composites with flibe is verified, SiC/SiC composites can be confidently used for control rod housing with high neutron dose (up to 70 DPA). Currently, the synergistic effects of neutron irradiation and chemical environment in FHR are unknown and have to be investigated.

3.2.2.3 ASME Code Qualifications

Although there are a lot of reports on the good performance of C/C and SiC/SiC composites for nuclear applications (Snead et al. 2008; Noda, Kohyama, and Katoh 2001; Bonal et al. 2009), there is no precedence for using ceramic composites within a nuclear reactor, and ASTM standard test procedures need to be established from the mechanical and environmental tests. Currently, a program for composite qualification is under way to enable the use of these materials in the proposed U.S. Next Generation Nuclear Plant (NGNP) gas-cooled reactor in 2021 (Ren et al. 2011). U.S. NGNP Composite Program conducts research and development for qualification and testing of the SiC/SiC composites, as the primary option, and C/C composites (Bonal et al. 2009). Some specific tasks of the program are 1) confirmative feasibility issues for SiC/SiC including irradiation effect and fabricability, 2) key technical issues governing the lifetime envelope such as irradiation creep and time-dependent fracture, and 3) support to test standards and design code development in the framework of ASTM International and ASME (Katoh, Wilson, and Forsberg 2007).

3.3 FHTR Fuel

This section provides an overview of the options for FHTR fuel development and production, including issues for performing irradiation testing and procurement of startup fuel. In this context, there is a distinction between the fuel for the commercial FHR and for the FHTR.

3.3.1 FHTR Fuel Development

There are two options for an FHTR:

- *Prototypical FHTR*. Under this option, the FHTR would be designed to test a specific concept with a specific fuel type. The reactor would be fueled with the expected fuel for the commercial prototype reactor. This option has the shortest schedule and lowest costs. A large-scale example is the Shippingport LWR, where changes in fuel design implied a complete change of the reactor internals.
- *General-purpose FHTR*. Under this option the FHTR would be designed to test different FHR fuels and concepts. Such a reactor may have a driver fuel and a large center test section. The Advanced Test Reactor (ATR) is one example of many.

While there are several options to consider for FHR fuel, this subsection focuses on the base case scenario of an annular graphite matrix loaded with coated-particle fuel, diagrammed in Figure 3-2. Appendix A and the third FHR workshop white paper contain more detailed information about the baseline fuel choice as well as other potential fuel types.

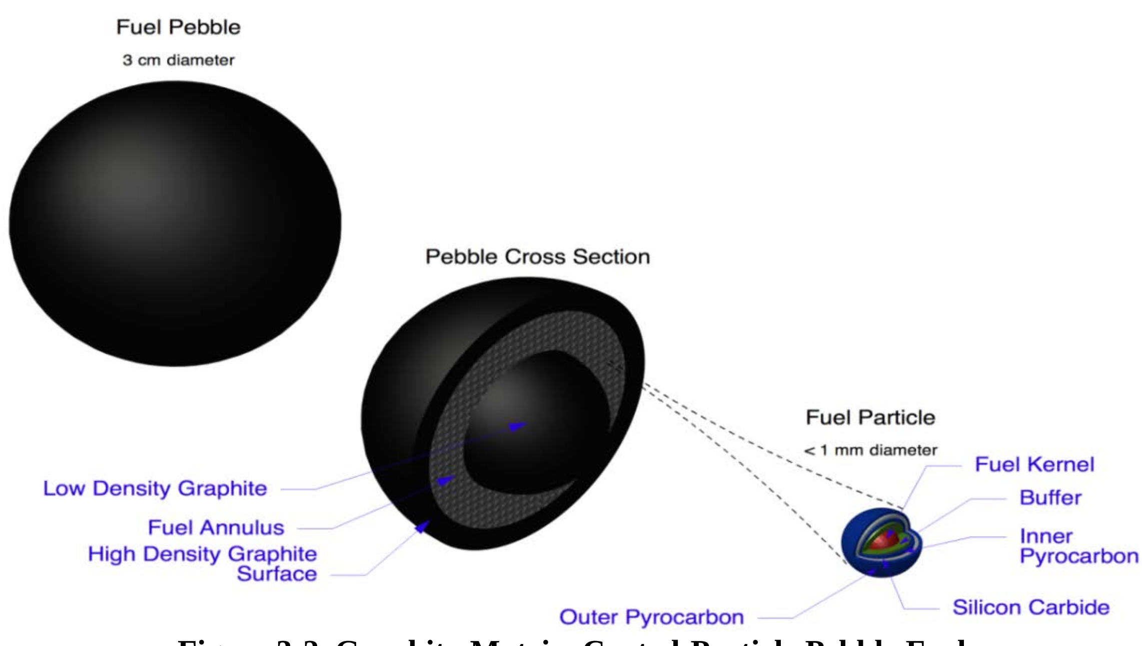

Figure 3-2. Graphite-Matrix, Coated-Particle Pebble Fuel

For the IRP, the FHR baseline fuel is the pebble-bed graphite-matrix coated-particle fuel because it appears today to meet all the requirements for a large power reactor and demonstrates the following characteristics that make it suitable for an FHR environment:

- *Chemical compatibility*: Graphite is chemically compatible in a radiation environment with high-temperate salts. This quality was demonstrated by the MSRE at ORNL in the 1960s.

- *High-temperature capabilities*: The coated-particle fuel is today the only demonstrated high-temperature fuel, with temperature limits for fuel kernel damage of approximately 1,600°C. This level is well below anticipated transient and accident temperatures the fuel will see in an FHR.
- *Simplified refueling*: Graphite floats in liquid salts. In a pebble-bed FHR, the pebbles are planned to cycle through the reactor once a month. Refueling in an FHR is simplified because the pebbles are engineered to float up through the core into the refueling machine (Forsberg 2008a; Katoh, Wilson, and Forsberg 2007).

While graphite-matrix pebble fuel has been used in the past, much research and development is still needed before starting an FHTR. While reactors such as the Thorium High-Temperature Reactor (THTR)-300, ArbeitsgemeinschaftVersuchsreaktor (AVR), and High-Temperature Reactor (HTR)-10 demonstrated that the tristructural isotropic (TRISO) particles can successfully be used as a high-temperature fuel, past applications and the current FHR design have several differences:

- *Smaller size.* The 3.0-cm diameter of the FHR pebble provides for a higher surface-area-to-volume ratio and thus enhances heat transfer in the reactor.
- *Thin annular fuel layer*. The annular design of the FHR pebble is engineered to reduce the temperature gradients throughout the fuel. This design also enables operation at higher volumetric power densities.
- *Inert graphite center.* The central region is intended to comprise a lower-density inert graphite matrix. This design helps to modulate the overall density of the fuel pebble and ensure the desired buoyant forces.
- *High heavy metal loading.* Because the flibe coolant in FHRs is an effective neutron moderator, FHR fuel optimizes to substantially higher heavy metal loading that HTGR fuel, requiring higher particle packing density.
- *Fuel environment.* Previous pebble bed reactors have been gas cooled. The introduction of a liquid salt changes the relationship between fuel and coolant via chemical interaction, coolant absorption, lubrication, etc.

The smaller size of the fuel pebbles does not introduce any greatly challenging complexities. However, the annular design and higher heavy metal loading introduce complexities that merit additional studies. Table 3-7 summarizes the design choices and some of the complexities they present.

Pebble fuels exhibit several potential advantages over other particle fuel geometries, including lower fabrication costs as a result of no complex geometry internal to the fuel, the ability to perform on-line refueling, and less complex – and potentially lower cost – refueling systems. While pebble fuel may be less expensive than other TRISO-based fuels, preliminary economic analyses indicate that particle fuels will have higher fabrication costs than traditional

UO_2 fuel pellets. This factor may be offset by increased fuel utilization as higher burnups and improved neutron economies are realized. Continuous refueling allows the reactor to operate with less excess reactivity.

Another factor driving up the cost of TRISO-based fuel is the defective particle fraction requirement. Current standards hold the fuel to a defective particle rate of 10^{-5}. This fraction requires extreme precision in the manufacturing process and large amounts of destructive testing. The extensive quality control testing often drives average fuel quality significantly beyond specification as a result of the statistical sampling (Forsberg 2006). To reduce the defective particle rate, a source term (and thus the allowable failure fraction) would need to be back-calculated from a given accidental release (Hunn 2012). Another method for reducing the cost would be to discover a non-destructive sampling method.

If a general-purpose FHTR is to be built, it would test the same fuels, but the driver fuel may be different. The short height of a test reactor allows more options in reactor design. Specifically, it would allow the use of the same coated-particle fuel in graphite-matrix prismatic fuel blocks as was developed for the Fort St. Vrain reactor. The use of fixed fuel allows the fuel enrichment and fuel loading to create the flux environment required for the test section. There are manufacturing limits to the height of the blocks. In a gas-cooled reactor, these fuel blocks are stacked on top of each other. This stacking would be more complicated in a large FHR where the blocks float. However, this is not an issue in an FHTR where the core is one block high.

Table 3-7. FHR Fuel Design Challenges

Design Choice	Important Questions
Annular fuel layer	How thin can the layer be made? How sensitive is the fuel performance to distribution of kernels? How precisely can the distribution of kernels be controlled? Can the annular geometry be made in an economic, scalable process? Can the fuel layer be reliably bonded to the inner and outer inert layers? Can high particle packing density be achieved to obtain optimal heavy metal loading?
Inert graphite center	Will differential radiation-induced shrinkage and swelling between the low-density central region and higher-density fuel region cause unacceptable damage or total failure? Will the lower density lead to enhanced shrinking/swelling? Will the potential void space lead to density changes via either coolant absorption or volumetric changes?
Liquid salt coolant	Will the coolant be absorbed by the outer graphite, leading to increased density and perhaps loss of buoyancy? Will lubrication via the coolant and reduced contract forces from near-neutral buoyancy decrease the generation of graphite dust as compared to gas-cooled pebble bed reactors?

3.3.2 Irradiation Testing

Once a final fuel form design has been chosen, manufactured, and characterized out of pile, it must be subjected to an extensive in-pile campaign that will include multiple irradiations and accompanying post-irradiation examination (PIE). Likely reactors for such a campaign include the ATR at Idaho National Laboratory (INL) and the High-Flux Isotope Reactor (HFIR) at ORNL. The scope of this campaign will likely be similar to the Advanced Gas-Cooled Reactor (AGR) Fuel Development and Qualification Program, but the irradiations can likely be accomplished more quickly because of the higher power density of FHRs as compared to HTGRs. Figure 3-3 details the purpose, goals, and feedback mechanisms of the eight planned AGR irradiation tests. Similar irradiation experiments would determine how FHR fuel responds to radiation-induced effects under prototypical and accident scenario FHR conditions. The insight provided about radionuclide transport would help validate the source term while other PIEs would help to validate fuel performance models.

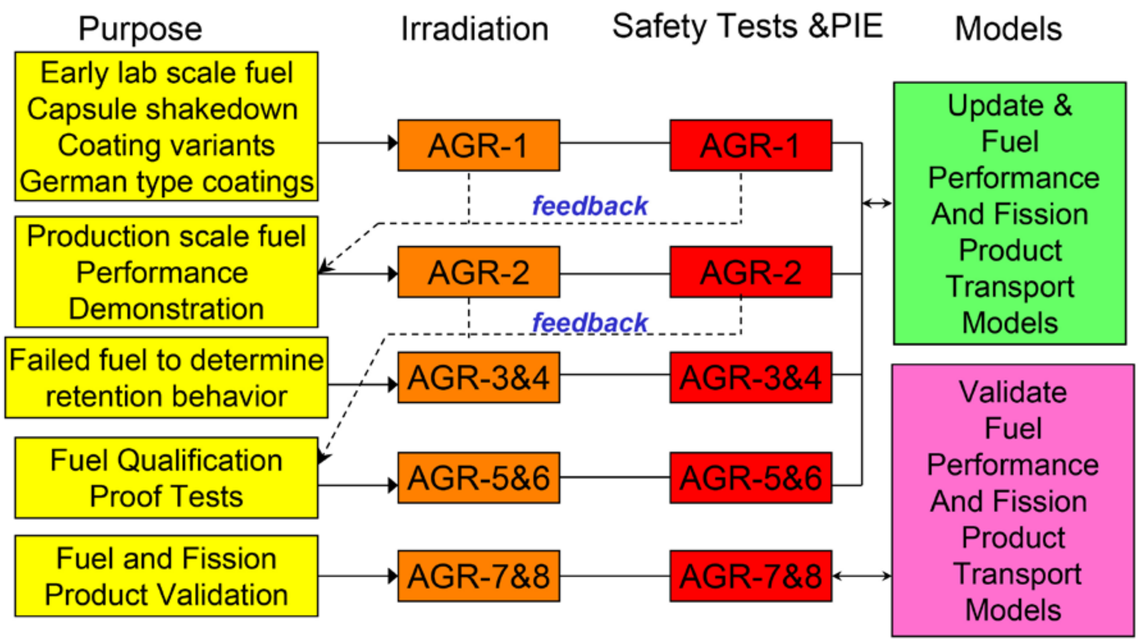

Figure 3-3. Overview of AGR Program Activities (Petti 2006)

Fuel irradiation campaigns are both time consuming and resource intensive. A 2004 General Atomics (GA) report estimated that the Very High-Temperature Reactor (VHTR) demonstration module's (now the NGNP's) fuel irradiation campaign would take approximately 12 years and cost \$80.4M (Hanson and Saurwein 2004). These GA figures were provided under the assumption that the testing program would be incremental, with the base science being provided by the AGR program. It could easily be argued that the FHR fuel qualification program would fall under the same assumption. If not and the FHR program will be similar in scope to that of the AGR, a 2005 INL report estimates the time to completion at 15 years and the total cost of fuel development and qualification at \$223M (INL, ORNL, and ANL 2005). However, FHR fuel operates at much higher power densities than VHTR and HTGR fuel and utilizes a higher heavy metal loading as well. These key differences enable FHR fuel to reach full depletion in significantly less time. Therefore, it should be possible to reduce the in-pile time required by the FHR fuel development and qualification program. This accelerated testing time should reduce completion time and budget for the FHR program compared to the AGR and VHTR programs.

Preliminary studies performed at UCB investigated both the ATR and HFIR as the location for the FHR fuel irradiation campaign. A 2010 report proposed that both the HFIR and ATR could be used to provide prototypical FHR fuel temperature and neutron environments by using various test rig designs. More importantly, the test fuel was predicted to reach full burnup in approximately 0.9 years, which is very close to the expected time for fuel in a commercial-sized reactor to reach full burnup (0.91 full-power-equivalent years). The higher fluxes required by the FHR's high power density enable this accelerated irradiation time. As stated previously, these "quick" irradiations could potentially reduce the required fuel qualification time and cost for the FHR when compared to those proposed by the AGR and NGNP (Gomez et al. 2010).

3.4 FHTR Unique Components

An FHTR will require control and monitoring equipment consistent with those found in other modern research and test reactors, as well as equipment and systems unique to the salt-cooled reactor design. This section identifies this critical equipment and discusses to what extent additional development is required for viability in the fluoride-salt reactor environment. Many of these advances will also be directly relevant to a commercial prototype FHR, although the choice of coolants and fuel may introduce additional demands.

3.4.1 Instrumentation

Reactor instrumentation provides on-line monitoring of plant conditions during all phases of reactor operation and should be usable over multiple operational cycles before maintenance is required. The key operating parameters for the reactor primary system are as follows:

- Temperature
- Neutron flux
- Coolant flow rate
- Pressure
- Coolant level/inventory
- Structural vibration.

If the reactor utilizes pebble fuel, the pebble bed geometry and inventory must also be monitored. Additional important coolant parameters include electrochemical potential, dissolved oxygen concentration, radioactivity, tritium concentration, and optical clarity.

Two major constraints on equipment for the FHTR (differentiating it from existing test reactors) are chemical compatibility and operating temperature. Material contacting the salt, or exposed to the inert cover gas above the salt, must resist corrosion from the salt itself as well as any products of normal activation, corrosion, radiolysis, or thermal decomposition such as hydrogen fluoride and BeF_2 fall-out in gas spaces (for flibe) and hydrogen diffusion. In addition, to keep the coolant molten, the reactor internals will be operating continuously at temperatures greater than 450°C even during maintenance outages; during power operation the temperature may be as high as 700°C.

Some instruments, such as thermocouples and flux monitors, can be isolated from the chemical environment with protective sleeves or by imbedding them in other structural components; therefore, their ability to perform at temperature is the primary concern during qualification. LWRs use fission chambers in and around the core to monitor flux; for an FHR the chambers and cabling require qualification of high-temperature variants. Some standard high-temperature thermocouples are available, such as type N, R, and S, though they may require more evaluation of their stability under irradiation.

The coolant level/inventory, static pressure, flow (dynamic pressure), and optical properties may be measured most conveniently from above the salt surface using standpipes; the main issue for these devices will be choosing optical windows and transducer membranes that are resistant to the salt environment and designing the reactor core to have relatively low pressure loss (for standpipe access to the reactor downcomer and core inlet). As with other equipment and seals located above the salt pool, there is the option of providing local cooling to reduce the operating temperature requirements; however in that case, it will need to be demonstrated that the cooled surfaces will not be fouled by deposits released from the salt.

Chemical concentrations and radioactivity can be measured outside of the primary vessel using a small loop with temperatures closer to the salt melting point; however, with the exception of gamma activity measurements, these vessels will still require chemical compatibility with the salt.

Similar instrumentation for temperature, inventory, pressure, flow, and chemistry will also be needed for the intermediate salt loop, as well as for salt volumes dedicated for support systems such as the direct reactor auxiliary cooling system (DRACS), fuel handling system, inventory control/holding tanks, and spent fuel storage.

3.4.2 Components

Several categories of active components are needed for the operation of the FHTR:

- Heat exchangers [primary, intermediate (if used), and auxiliaries including the DRACS heat exchangers]
- Pumps
- Valves
- Reactivity control mechanisms
- Fuel handling system
- Trace heating.

These items face the same thermal and chemical compatibility challenges as the instrumentation equipment. Among these, the heat exchangers are likely to represent the most significant developmental challenges. A heat exchanger is a cold spot in the system and supports a significant heat flux, both factors that can drive corrosion and corrosion product deposition. In addition, the heat exchangers may be required to provide a barrier for tritium diffusion to control tritium releases to the balance-of-plant and the environment. The designs must also allow inspection and be serviceable in the event of plugging or leaking. These challenges are compounded by a desire to maximize the thermal efficiency of the heat exchanger by minimizing its wall thickness and maximizing its surface area. The intermediate and auxiliary heat exchangers may also have to support different media depending on the plant design (e.g., salt/air, salt/water).

Salt pump technology primarily depends on the availability of bearings and shaft seals compatible with the salt at high temperatures. Qualification of such items will benefit other mechanical devices that need to penetrate into the salt environment such as control rod drive mechanisms, the fuel handling system, and valve seats (other than freeze valves, which may be too slow or too small in diameter for some applications).

The reactor will require significant amounts of trace heating, most of which will be located outside of the salt and significant radiation environments, but which will still be difficult or impossible to replace and thus will require high reliability. However, some in-vessel trace heating or other electrical heating systems may be required to prevent salt freezing in confined spaces (e.g., around the pump impeller and heat exchangers) when the reactor is not operating or defueled, as well as during startup and recovery from overcooling transients.

3.4.3 Inspection Equipment

The transparency of liquid fluoride salt provides the opportunity to make visual inspections during both power operation and shutdowns. Such inspections are important for monitoring the physical structures inside the reactor pool, as well as tracking any debris generated during operation and maintenance. In addition to viewing windows in the reactor lid gas space, standpipes can be used to penetrate the salt with mirrored surfaces attached to the bottom so that obscured areas can be reached. Additionally, access to different levels of the pool may allow infrared temperature and optical chemistry measurements in critical locations. However, the feasibility of using such devices in a large, flowing fluoride-salt pool with thermal gradients and significant radiation fields has yet to be demonstrated.

As part of the inspection equipment, but relevant to many other parts of a fluoride-salt reactor plant, salt removal systems will be also critical. Because of the risks of exposing fluoride salts to oxygen and other salt-specific concerns (such as beryllium contamination from flibe), comprehensive equipment is needed for salt and salt corrosion product removal from structures that move in and out of the vessel (such as the refueling machine, fuel elements, maintenance equipment, and components removed for service). This work needs to be performed remotely in an inert atmosphere, with provisions for collecting and storing the waste salt solutions.

3.5 FHTR Component and System Testing

One of the primary aims of the FHTR is to test the components and systems that make up the commercial prototype reactor in a scaled-down, integrated environment. This section provides an overview of a strategy for testing and qualification of FHTR components, based on discussion at the third FHR workshop. This strategy involves using existing facilities to the maximum extent possible and thus avoiding the time and expense needed to build a new component test facility (CTF).

The components should be tested to assess their reliability. Recognizing that it is not practical to test components for the full service life they will experience in the FHTR, the use of component designs that have operating experience from other applications with similar environmental conditions, and approaches to accelerate testing, should be emphasized. In general, components should be tested for fatigue, creep, and yield in high-temperature environments. Even though the power level of the reactor will be at most 20 MW, some tests may still need to be scaled geometrically. It may be more prudent to perform tests with actual liquid salt, rather than stimulant fluids, to preserve redox and solubility (of the cover gas or tritium), especially for primary coolant tests. The quality assurance procedures should also be tested at this stage, ensuring that all data logged and tests taken are to a satisfactory standard. These component and systems tests could be used to validate code or to demonstrate that a certain code can be used accurately for predictions. For example, in the South African HTGR CTF, one of the intentions of the Pebble Bed Micro Model was to show that Flownex could be used for accurate dynamic behavior of the system verification and validation (Van Ravenswaay et al. 2006). Verification and validation of codes is also an important step in the testing phase. Visual tests of manufactured or procured components are also crucial but simple, such as ensuring all components are of the correct and expected dimensions and are fully functioning.

Possible facilities that could be used in place of a CTF exist at universities and national laboratories, with some modifications. Some facilities are better suited to certain types of tests. For example, tests involving salt chemistry could be performed at UW. These facilities should be large enough to accommodate testing of FHTR components, because most of the components and subsystems will be much smaller than in the commercial prototype. Note also that the data from the earlier ORNL MSRE provide information on liquid salt components testing.

The following subsections provide additional detail on component testing and systems testing.

3.5.1 Components Testing

The critical components that should be qualified and tested before use in the FHTR include the following:

- *Heat exchangers.* For heat transfer testing using oil, the heat exchangers should be scaled geometrically and the number of tubes reduced, ensuring that the Reynolds number of the primary and secondary fluids in the heat exchangers match those of the commercial prototype. For salt tests, the numbers of tubes should be reduced to perform component testing at reduced power and flow. The FHTR will have the flexibility to assess different designs of heat exchangers. Tests and qualifications must be performed on all the designs. Other heat exchanger testing includes flow-induced vibration, which can cause fretting wear.
- *Reactor vessel and primary piping.* The reactor vessel for the FHTR will be much smaller than that for the commercial prototype and therefore can be tested in an existing facility.
- *Fuel elements*. Section 3.3 discusses fuel development for the FHTR. The fuel pebbles for the FHTR are not readily available and will need to be manufactured. NUREG-1537 (NRC 1996) outlines a list of criteria to which the fuel must adhere. Some of the salient points include fuel compatibility with the environment and integrity of the fuel design, taking into account “melting, softening, corrosion and erosion caused by coolant, physical stresses by mechanical or hydraulic forces.”
- *Liquid salt pumps*
- *Fluidic diode*
- *Valve technologies.* Valve testing includes high-temperature testing.
- *Power conversion components.* These components include turbines, compressors, shafts, regenerators, economizer, heat recovery steam generator (HRSG), superheater,

evaporator, and deaerator. Most of these components are available “off-the-shelf,” and the developer would have performed tests before sale. However, these components must still be qualified for FHR use.

3.5.2 Systems and Subsystems Testing

The critical components that should be qualified and tested before use in the FHTR are listed below. As much as possible these should be tested in existing facilities.

- *DRACS.* It may be possible to perform tests with simulant fluids and scaled-down geometry, because the DRACS loop is solely concerned with heat removal. Redox will be of lower concern in the DRACS loop because there is no graphite, fuel elements, or tritium production.
- *Control and shutdown rod technologies*
- *Salt processing and inventory control system*
- *On-line chemistry monitoring*
- *Cover gas chemistry and pressure control system*
- *Control and instrumentation.* This area includes testing sensor technologies, such as temperature monitoring, flux monitoring, and pressure monitoring.
- *Fuel handling system*
- *Beryllium safety*. It is important to be familiar with beryllium safety procedures because the primary salt and the redox agents may contain it.
- *In-service inspection technologies*
- *Tritium removal system, tritium barriers.* It may be possible to test more than one method of tritium removal/recovery in the FHTR. These methods will need to be tested first in the laboratory.
- *Tribology in an FHR environment* (especially for the fuel pebbles in the salt and valve seats for stop valves).

3.6 FHTR Safety Analysis and Licensing Code Validation

Safety analysis and licensing code validation will be needed for thermal hydraulics, neutronics, and systems response.

3.6.1 Thermal Hydraulics

Transient thermal hydraulic analysis of the FHTR will focus on a set of LBEs that are expected to potentially challenge the system’s ability to meet Regulatory Design Criteria (RDCs). To aid the early FHR development process and guide pre-conceptual design, the first FHR workshop identified a subset of these LBEs, referred to as “characteristic LBEs,” for the initial FHR development effort. This subsection focuses on the similitude assessment between the commercial prototype reactor and the FHTR with respect to these LBEs, followed by a list of existing and FHR-specific validation basis for FHTR licensing codes.

3.6.1.1 Similitude Assessment

Because the FHTR shares the same thermal hydraulic phenomenology with the commercial prototype reactor, both facilities are expected to follow the same operating modes, and the FHTR should be used to analyze the system's response to all characteristic LBEs as detailed in the first FHR workshop white paper. Principal figures of merit that arise from steady-state operation and thermal hydraulic transients in the FHR (both commercial prototype and FHTR) are listed below. Any systems code used for licensing of the FHTR must be validated for this set of LBEs, based on the listed principal figures of merit. After being validated, these systems codes can be used to model both the FHR and the FHTR.

3.6.1.2 Thermal Hydraulic LBEs

A set of bounding events was postulated for the FHR class in the first FHR workshop white paper. These events put severe tests on the reactor safety systems and are all considered to be events in the beyond design basis event (BDBE) frequency range or lower. The same bounding events, except for large loss of primary coolant that is not listed here, apply to the FHTR. Key thermal hydraulic events have been identified as follows:

- *Protected loss of heat sink (LOHS).* From full-power conditions, assume that all cooling via the normal cooling system is lost (loss of intermediate loop). As soon as the reactor protection system detects off-normal conditions, the reactor scrams. Analyze the event for cases where the DRACS heat removal capability is limited for an extended time.
- *Protected loss of forced circulation (LOFC).* From full-power conditions, assume the reactor scrams as soon as the reactor protection system detects off-normal conditions for an LOFC. Analyze the event for two cases:
    - Assume that the pumps trip and begin to coastdown. Assume that the DRACS heat removal capability is limited for an extended time.
    - Assume that the flow through one pump stops suddenly and the others continue to operate normally. Assume that the DRACS heat removal capability is limited for an extended time.
- *Overcooling events.* From full-power conditions, assume loss of heat removal to the power conversion unit. Assume the reactor scrams as soon as the reactor protection system detects off-normal conditions. Assume pumps operate in a configuration that maximizes heat removal from the primary coolant. Assume the normal shutdown cooling system and DRACS loops operate at full capacity. Assume electric heaters are not available. Analyze for 12 hours. Note that the blockage of the DRACS loop for LOHS and LOFC effectively evaluates the potential impact of transients with freezing.
- *Flow blockage.* Assume blockage of flow to or from one fuel assembly (fixed-fuel design) or because of pebble breakage (pebble fuel design).
- *Asymmetric flow transients.* Assume, for instance, that a single DRACS loop is compromised out of multiple DRACSs, a single pump fails, or one out of multiple shutdown cooling systems is lost.

- *Operator-caused LBEs.* In some cases, an operator might cause an initiating event that will exacerbate events which initially required action. For instance, inadvertent insertion of cooling water injection may increase corrosion in the core and materials degradation.

More details about the LBE selection approach are provided in the first FHR workshop white paper.

3.6.1.3 Principal Thermal Hydraulic Figures of Merit

Principal figures of merit that arise from normal operation and key thermal hydraulic transients are as follows:

- Peak fuel element temperature to avoid fuel failure and release of radionuclides (very unlikely to govern any FHR LBE because of the large thermal margin of FHR fuel)
- Peak local power density, which affects particle and element thermal stresses and may be important
- Time at temperature for the fuel, which influences radionuclide release
- Peak bulk coolant outlet temperature, which is a simple metric indirectly related to the structural integrity of the system
- Time at temperature for metallic and ceramic structures for long-term structural materials' creep deformation and degradation
- Peak thermal stress induced in metallic and ceramic structures (requires coupling with a structural mechanics code), including coolant thermal shock, striping, and ratcheting
- Minimum coolant temperature in the DRACS loop to assess importance and duration of a potential overcooling transient, including freezing phenomena in the natural draft heat exchanger, reducing heat removal capacity of a safety-related component
- Temperature difference across the DRACS, which is one of the key parameters associated with passive decay heat removal through natural circulation
- Time to establishment of natural circulation and how long it can be sustained.

Any thermal hydraulic code used for steady-state and transient analysis of the FHTR will be required to accurately predict these figures of merit. The capability depends on the proper accounting of thermal hydraulic phenomena in the system, as detailed in the second FHR workshop white paper.

3.6.1.4 Existing and Required Experimental Basis for Thermal Hydraulic Modeling Validation

As highlighted in the second FHR workshop white paper, although preliminary thermal hydraulic modeling of the FHR has been performed using the RELAP5 systems analysis code, the code, in its current state, is not capable of capturing some of the key FHR thermal hydraulic phenomena. A list of verification and validation efforts needed to increase the reliability of any code to properly model thermal hydraulic phenomena for the FHTR is presented below. Because the commercial prototype FHR and FHTR share the same thermal hydraulic phenomena, this list is similar to that generated for the commercial prototype. More details can be found in the second FHR workshop white paper.

The experimental test program for the FHR will provide empirical data to validate models that predict plant reliability and safety (NRC 1996). After being validated, these models can be used for both the commercial prototype and FHTR. The phenomena identification and ranking table (PIRT) process provides the basis to identify the dominant phenomena that control the system response to a specific reliability or safety-related transient. A PIRT-type exercise should be applied to the FHTR in the short term to that effect. Based on the current point of FHR design, this exercise will help identify thermal hydraulic priority phenomena, those important for transient response of both the FHR and the FHTR to LBEs, those that lack a reliable knowledge basis, and those needed to provide experimental data to validate these models. The hierarchical two-tier scaling methodology, which informs integral effects tests (IETs) and separate effects tests (SETs) required for validation of thermal hydraulic models, is detailed in (Bardet and Peterson 2008) and explained in the second FHR workshop white paper. The following lists the tests applicable to both the FHR and the FHTR:

- *IETs*. Table 3-8 summarizes IETs needed to validate key FHTR thermal hydraulic phenomena. Details about a subset of these IETs are provided in the second FHR workshop white paper.

- *SETs*. The FHR SET experiment program would cover FHR key thermal hydraulic phenomena for which high-quality, experimentally validated models are not yet available. While detailed conceptual design phase PIRTs have not yet been developed, several dominant phenomena have already been identified in FHR modeling efforts where existing experimental data are insufficient. This list is shared with the FHTR SET requirements, because both systems share the same thermal hydraulic phenomena.

    - *Viability phase*. For the viability phase, SET experiments include studies of mixed convection heat transfer in pebble beds and in vertical channels using simulant fluids, where relevant experimental data do not exist in the range of Prandtl and Grashof numbers that would occur in the FHTR.

    - *Performance phase*. During the performance phase, SET data will be collected for prototypical components with the prototypical heat transfer fluid, as detailed in Section 3.5. Adequate instrumentation should be used to collect heat transfer, pressure drop, and other SET data of interest. In particular, these data would address questions about the potential impact of thermal radiation on heat transfer to liquid salts. Very limited data for infrared absorption is available for flibe and the candidate salts for the secondary coolant. Thus, the performance phase is also expected to have SET experiments to measure absorption in the primary, secondary, and DRACS salts. Table 3-9 summarizes SETs needed to validate key FHTR thermal hydraulic phenomena.

Table 3-8. Existing and Projected IETs to Validate Key FHTR Thermal Hydraulic Phenomena*

Test Facility	High Prandtl Number	Multi-Dimensional Porous Media Flow	Natural Circulation, 1Φ, Non-compressible	Freezing and Melting	Conduction in Fuel and Structures	Core Bypass Flow	Status
CIET	x	x	x		x		Operational
CIET 2	p		p			p	Under design/ construction
PREX		x					Operational
APEX			q			q	
HTTF			q			q	
Ohio State DRACS Loop	p		p				Planned
ORNL Salt Loop	p		p	p		p	Under design/ construction
CAS Salt Loop	p	p		p	p	p	Planned
CTF	p	p	p	p	p	p	Future roadmap
2-MW Test Reactor	p	p	p	p	p	p	Future roadmap

* Definitions: x=existing IET providing limited quality/scoping data; q=existing IET providing quality data; p=projected IET

Table 3-9. SETs Required to Validate Key FHTR Thermal Hydraulic Phenomena*

Viability Phase	Heat transfer in pebble beds and vertical channels	x
	Heat transfer in pebbles	-
Performance Phase	Heat transfer at prototypical temperature	x
	Pressure drop at prototypical temperature	x
	Thermal radiation at prototypical temperature	x

* Definitions: -=existing experimental basis; x=additional required SETs

As progress is made in FHTR and FHR development, a more substantial experimental database will become available to validate thermal hydraulic models. To include proper modeling of all FHR key thermal hydraulic phenomena, a simultaneous effort should be made to refine code selection and implementation of new features in the codes. Code-to-code comparison and formal benchmarking exercises can provide an efficient method to detect errors in the way codes model some phenomena. Eventually, code validation will be needed, performing sensitivity analyses to relevant parameters and uncertainty calculations over a range of parameter values that include all modes of operation of the FHTR, down to BDBEs.

3.6.2 Neutronics

Licensing safety neutronic analyses for the FHTR will focus on calculating parameters for startup, normal operation, shutdown, and severe accidents as well as experimental transients. This subsection discusses the figures of merit (i.e., values that must be calculated by a neutronic code), the existing experimental basis for validating neutronic codes for licensing, how to assess the similitude of this experimental base to the FHTR, and finally, the steps forward for code validation.

3.6.2.1 Neutronics Figures of Merit

The evaluation models (EMs) for licensing safety analysis must be developed to calculate all parameters specified in NUREG-1537 (NRC 1996) for a non-power reactor (i.e., test reactor) for transient analysis as well as normal operation. Most of these parameters were catalogued for an FHR in Table 5-1 of the second FHR workshop white paper and can be broadly categorized into four capabilities: high-fidelity criticality analysis, depletion analysis, transient analysis, and sensitivity and uncertainty analysis.

3.6.2.2 Similitude Assessment and Existing Experimental Basis for Neutronic Modeling Validation

The neutronic components of the EM will be validated using existing experimental data from the International Criticality Safety Benchmark Evaluation Project and the International Reactor Physics Benchmark Experiment Project in addition to modern graphite-moderated, coated-particle fuel test reactors: VHTRC, HTR-PROTEUS, HTTR, HTR-10 and ASTRA facilities. A detailed description of these reactors can be found in Appendix D of the second FHR workshop white paper. Only relevant experimental data can be used to validate the components of an EM (Zuber et al. 1998). The similitude of experimental data will be assessed using the sensitivity and uncertainty methods developed at ORNL and previously utilized in some licensing criticality safety analyses (NRC 2005; Broadhead et al. 1999).

The amount of similitude between the existing experimental base and the FHTR will determine the amount of conservatism in the initial startup subcritical neutronic testing procedures. Once some initial experimental data can be obtained from the FHTR, this subcritical experimental data can be used to validate EMs for systems configurations with progressively less conservatism, until the EMs for the FHTR in a critical state and in a transient state are validated.

3.6.2.3 Next Steps for Validation

Based on the expert advice from the participants at the second FHR workshop, the IRP is implementing the Monte Carlo neutron transport code SERPENT for high-fidelity neutron transport and depletion analysis and to develop nuclear data for diffusion neutron transport codes to be used for transient analysis. The IRP must also select a diffusion code package for coupled thermal hydraulic/neutronic transient analysis. In addition, the IRP must select a sensitivity and uncertainty analysis code to assess similitude. The IRP has initially focused on utilizing SCALE’s TSUNAMI code but will transition to advanced sensitivity and uncertainty codes as they become available. The IRP’s validation efforts for neutron transport will focus on validating SERPENT for simulating FHRs.

3.6.3 Systems Response

Coupled and multiphysics modeling may be needed for some FHR steady-state and transient analyses. A detailed discussion on this topic is provided in the second FHR workshop white paper. This subsection focuses on the similitude assessment between the commercial prototype reactor and the FHTR with respect to LBEs involving coupled thermal hydraulic, neutronic, structural mechanics, and chemistry phenomena, followed by information on the FHR-specific validation bases required for FHTR licensing codes.

3.6.3.1 Similitude Assessment

Because the FHR and FHTR share the same thermal hydraulic, neutronic, structural mechanics, and chemistry phenomena, this set of parameters is expected to be the same between the two reactors. Table 3-10 lists parameters that must be exchanged between coupled modeling codes to properly assess the system’s behavior during steady-state operation and transients such as anticipated transient without scram (ATWS).

Table 3-10. Parameters That Must Be Exchanged Between Coupled Modeling Codes to Assess System Behavior During Steady-State and ATWSs

From / To	Neutronics	Thermal Hydraulics	Structural Mechanics
Neutronics	-	T_{fuel}, $T_{coolant}$, $T_{structures}$, pebble packing fraction*, flow-induced pebble re-zoning*	Geometry (pebble motion, control rod channels, etc.)
Thermal hydraulics	Power distribution, DPA (conductivity)	-	Geometry (thermal expansion, etc.)
Structural mechanics	DPA (damage)	T_{fuel}, $T_{structures}$, coolant velocity (flow-induced vibration)	-
Chemistry	Radionuclide source term	Radionuclide transport	Corrosion

* For PB-FHR

Any coupling capability between modeling codes will need proper validation, using a database that has not been adapted to the FHR design to date.

3.6.3.2 Required Experimental Basis for Coupled Modeling Validation

The following coupling and multiphysics validation basis is needed for the FHTR modeling codes:

- *Validation of coupled neutronic/thermal hydraulic and multiphysics modeling*. While considerable efforts have been made in various countries and organizations to develop coupled thermal hydraulic and neutronic codes, as illustrated by the few examples given in the second FHR workshop white paper, these code systems need to be properly validated for use with the FHTR (and eventually the FHR) design. For coupled neutronic and thermal hydraulic codes, the first step is to validate the codes independently, following methodologies presented in Sections 3.6.1 and 3.6.2, further discussed in the second FHR workshop white paper. Experimental data collected in the FHTR will precisely serve as a validation basis for the coupled system, as well as any multiphysics tool used for FHR modeling. As an intermediate step, benchmarks have been developed in international cooperation led by the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development that permit testing the neutronic/thermal hydraulic coupling and verifying the capability of the coupled codes to analyze complex transients with coupled core-plant interactions (Radulescu, Mueller, and Wagner 2008). However, these benchmarks have only been developed for LWRs (one boiling water reactor, one PWR, and one Russian PWR) and are therefore not readily applicable to the FHR technology.
- *Validation of coupled thermal hydraulic/structural mechanics modeling*. Testing of key FHR components in appropriate facilities, as detailed in Section 3.5, can be expected to play a major role in validating coupled thermal hydraulic and structural mechanics models. However, the FHTR will provide the ultimate experimental basis for validation of the coupling capability.
- *Validation of coupled neutronic/structural mechanics modeling*. Fuel irradiation and graphite irradiation tests can be expected to play a major role in validating coupled neutronic and structural mechanics models. However, the FHTR will provide the ultimate experimental basis for validation of the coupling capability.

3.7 FHTR Licensing Framework

The licensing strategy for the FHTR seeks to minimize uncertainty and the time required to complete the licensing process. To meet these goals, the FHTR will try to meet the requirements to obtain a Class 104c license for a non-power reactor through the NRC. This baseline strategy will seek a 20-year operating license, which is standard for reactors of this class.

The content requirements for the Class 104c test reactor safety analysis report (SAR) for the license application and the standard review plan for the NRC staff are detailed in NUREG-1537 (NRC 1996). NUREG-1537 is, in general, technology neutral and provides essential guidance from the NRC on the material required to demonstrate sufficient confidence in reactor safety. Some areas of NUREG-1537 need to be updated, such as the requirements for digital instrumentation and control, but the licensing of the FHTR in this framework fits within the current NRC experience base and therefore reduces some of the complexity in developing a licensing strategy for a new reactor concept.

The primary objective of the SAR is to demonstrate reasonable assurance that the public will be protected from radiological risks resulting from the operation of a reactor facility. NUREG-1537, Part 1 (NRC 1996), includes detailed content requirements for the SAR and should be used as the essential reference for the FHTR licensing strategy. Three critical sections of the SAR that merit further discussion for the FHTR are discussed in the following subsections of this white paper. Subsection 3.7.1 discusses the design criteria of the FHTR structures, systems, and components (SAR Section 3.1). Subsection 3.7.2 reviews the issues for the startup plan (SAR Section 12.11). Finally, Subsection 3.7.3 evaluates the acceptance criteria for the evaluation of a maximum hypothetical accident (MHA) and proposes a set of assumptions that can be used for such a scenario for the FHTR (SAR Section 13.1.1).

Note that the SAR for the FHTR must be developed for a specific and detailed reactor design. Because this design does not yet exist, the discussions in the following subsections are intended to be general discussions associated with the FHR class and the safety requirements for an FHTR in the 10- to 20-MWth power range.

3.7.1 Structures, Systems, and Components Design Criteria

As a first-of-a-kind reactor, the FHTR presents a unique set of challenges in the licensing process that must be met to demonstrate that the facility can be operated with minimal and acceptable risk to the operators and the general public. The FHTR SAR must include a set of design criteria that, when met, should satisfy the regulator that the high-level safety objectives will be achieveable. The design criteria for a reactor built to satisfy Class 104c should include applicable standards, guides, and codes, such as American Nuclear Society (ANS) standards or NRC regulatory guides. A set of ANS standards is currently in development for FHRs and will likely serve as the primary guidance for the FHTR design criteria when completed.

Note that the design criteria for non-power reactors are not as clearly defined as they are for nuclear power plants in 10 CFR 50, Appendix A. Instead, it is the task of the applicant to apply a set of design criteria that are both specific and general. The NRC can evaluate criteria for how well they meet high-level NRC requirements to protect the safety of the public. General design criteria that should be used, with references to current standards outlined in NUREG-1537 (NRC 1996) include the following:

- Cover the complete range of normal reactor operating conditions.
- Cope with anticipated transients and potential accidents.
- Use redundancy to protect against unsafe conditions in the case of single failures of reactor protective or safety systems.
- Facilitate inspection, testing, and maintenance.
- Limit the likelihood and consequences of fires, explosions, and other potential human-made conditions.
- Use quality standards commensurate with the safety function and potential risks.
- Use design bases to withstand or mitigate wind, water, and seismic damage to reactor systems and structures.
- Analyze the function, reliability, and maintainability of systems and components.

The general nature of the top-level design criteria for non-power reactors allows them to be applied in a technology-neutral manner to new reactor concepts such as the FHR. In cases where standards do not exist as a model for the SAR design criteria, applicants may develop their own criteria as needed to demonstrate to the NRC that the safety goals are met. The set of specific design criteria for the FHTR will likely include a high degree of conservatism because the knowledge base for this class of reactors is limited.

As the design of the FHTR matures, it will be of primary importance to integrate the high-level design criteria into the detailed requirements for structures, systems, and components. This effort, combined with the attractive inherent safety characteristics of the FHR concept, should result in a strong case that the risks posed by the operation of the FHTR are acceptable.

3.7.2 Startup Testing Plan and Acceptance Criteria

A detailed startup plan is required as part of the SAR for a Class 104c license and is particularly important for new reactor technologies such as the FHR with a limited amount of neutronic data. The startup plan builds confidence that the operating characteristics of the reactor are well understood and validates the models for the predicted reactor behavior. Initial startup testing for the FHTR would be expected over a period of 12 to 18 months. The neutronic tests for the startup plan are described in Subsection 3.6.2, while this subsection is focused on the acceptance criteria to meet NRC licensing requirements.

The acceptance criteria for the FHTR startup plan must satisfy the following requirements, adapted from NUREG-1537 (NRC 1996), to ensure that the reactor is functioning within the bounds for which it was designed and analyzed and that the license and the technical specifications are satisfied:

- The applicant should have plans for receiving fuel, handling and performing quality assurance checks on the new fuel, and loading fuel used in a critical experiment, in this case, loading fuel into the FHTR.
- The critical mass (number of fuel elements) should be approximately known and should be exactly determined by a systematic approach to loading fuel into the FHTR.
- Neutron detectors of high sensitivity and reliability may be used to supplement the operational instrumentation during subcritical neutron multiplication measurements.
- Measurements should be planned for operational reactor physics parameters, such as shutdown reactivity (to confirm shutdown margin) and reactivity feedback coefficients including temperature reactivity feedback, differential and integral control rod worths, power level monitors, scram and interlock functions, fuel heat removal, and related thermal hydraulic parameters.
- Measured and predicted reactor physics parameters should be compared, and the results of the comparisons should be evaluated against pre-established acceptance criteria.
- The control rods should be calibrated, and excess reactivity should be loaded systematically to obtain accurate values.
- Thermal power of the reactor should be calibrated acceptably and accurately to ensure compliance with the licensed power level limits and any other license conditions, such as pulse characteristics.

- Area and effluent radiation surveys should be conducted to confirm predictions of the radiological status of the facility.
- All instruments and components should be tested before routine operations begin.
- Other systems discussed in the startup plan should be tested and found to be operational before routine operations begin.

This set of acceptance criteria will be used as the basis to determine if the FHTR startup plan meets the NRC requirements for non-power reactors. Note that this set is not comprehensive, and there may be additional requirements that have not yet been identified for FHRs.

Two issues are important to note on the set of acceptance criteria detailed in NUREG-1537. First, because of the stochastic nature of the pebble bed packing, it will not be feasible to know the exact core geometry at criticality. Pebble accounting techniques can be used to determine the number of pebbles in the core, but small variations in the packing may introduce some error in the initial critical configuration. Based on this observation, the FHTR should adopt a strategy to approach criticality by packing a pebble bed and performing tests while withdrawing shutdown rods. This procedure should maintain the overall bed packing and reduce the impact of uncertainties associated with the bed configuration.

The second potential issue for the FHTR concerns the development of pre-established acceptance criteria to compare measured and predicted reactor physics parameters. Low levels of similitude for FHR cores to previous experiments could introduce larger errors in predicted values than for LWR non-power reactors with a much larger experience base. Further clarification is needed to determine what acceptance criteria should be established for comparing these results based on meeting the high-level safety objectives of the system. The licensing strategy for the FHTR should seek to maintain as much flexibility as possible in predicting specific values for reactor physics parameters at startup, while ensuring that the high-level safety functions are met by the design.

The most relevant historical experience for the startup plan of the FHTR is from the MSRE and small pebble bed HTGRs such as the AVR and HTR-10. Startup plans for the gas reactors could not be located, but the MSRE startup plan (Beall et al. 1964) is instructive in what procedures might be followed and how long the process might take. The startup plan for the MSRE involved testing over a period of 14 months and phases of added complexity, eventually leading up to full power operation. Table 3-11 is a preliminary startup plan for the FHTR adapted from that of the MSRE. This plan includes equipment, materials, and instrumentation testing that must be completed before fuel is loaded into the system and before physics testing can proceed. This process includes steps that are specific to the pebble-based FHTR design option, but could be modified as needed for fuel handling in alternative designs.

Table 3-11. Preliminary Startup Plan for FHTR Based on the MSRE Plan

Startup Phase	Months
Reactor installation	0
Operator and supervisor training	1-6
Dry equipment testing • Heaters, thermocouples, instrumentation, and data acquisition systems • Non-salt fluid systems (e.g., cover gas, pump-oil, component-air-cooling, and radiator-cooling systems) • Air compressor, containment ventilation, and electrical systems • Remote maintenance	1-3
Salt loop testing • Primary and secondary salt loading, inventory, and transfer methods • Salt circulating systems • Fuel recirculation system tests with inert (e.g., graphite) pebbles • Salt chemistry and inventory control system • Heat-balance methods, heat-removal systems, and temperature control • Graphite examination	3-6
Precritical shutdown • Partial loading of fuel pebbles • Completion of nuclear instrumentation and control-rod tests • Final maintenance	6-7
Critical experiments • Verification of nuclear instrumentation performance • Loading of fuel pebbles to criticality • Measurement of temperature reactivity coefficients, control rod worth, and response to salt flow changes • Establishment of baselines for determination of effects of power on chemistry, corrosion, and nuclear performance	6-9

Postcritical shutdown • Final seal of the reactor cell • Final check on containment leakage	10
Approach to full power • Heat balances and heat-removal control • Shielding and containment surveys • Measurement of power coefficient, xenon poisoning, salt permeation of graphite, and off-gas composition • Review of reactor control and data acquisition system performance	11-14

3.7.3 Maximum Hypothetical Accident and Acceptance Criteria

Because non-power reactors operate at significantly reduced thermal power compared to commercial plants and may be located in areas with low population density, there is a large reduction in the total source term of fission products and a corresponding reduction in the risk to the health and safety of reactor staff and the general public in the surrounding area. The license application for test reactors builds in conservatism for reactor technologies with minimal prior operational experience, such as FHRs, by requiring the analysis of an MHA. This subsection outlines the acceptance criteria for an MHA and proposes a set of assumptions that can be used for preliminary evaluation of the FHTR MHA.

The MHA is meant to be a conservative bound on the potential release of radioactive material and is a scenario that is not expected to occur. The initiating event for the MHA is typically the failure of some specified fraction of fuel and analysis of the release of fission products into the environment. The applicant can reduce the conservatism in the MHA by performing a sensitivity analysis on assumptions used in the bounding analysis. These scenarios of reduced conservatism can be used in the license application, though they may limit the technical specifications and operating parameters for the reactor. Intrinsic characteristics, such as the solubility of fission products in the FHR coolant, can be credited in the MHA.

The selection of the MHA scenario is specific for each reactor and must be determined from a broad set of accident sequences to determine the most conservative event. For a Class 104c reactor license with a high thermal power, the example scenario specified in NUREG-1537 for the MHA initiating event is described as follows:

> "Fuel cooling is compromised or reactivity is added to the reactor so that a certain amount of fuel melts causing cladding failure. Fission products are released into the reactor coolant and then into the facility air on the basis of conservative analysis, empirical information, or the combination of analysis and data." (NRC 1996).

This example specifies that cladding failure and fuel release occur in the primary coolant. However, for the FHTR, the most conservative case for radioactivity release might be from failures of cooling in spent fuel handling and storage because this case cannot take advantage of the high solubility of many fission products in the primary coolant. Further analysis is required to determine if this scenario is actually more conservative than the in-core fuel failures.

The following sequence and assumptions are proposed as a preliminary MHA for the FHTR, though the specific sequence of events will evolve as the design matures with more detail:

- At the maximum core burnup, a conservative fraction of TRISO particles lose integrity and all fission products from these particles are released into the primary coolant. The actual fraction used will require a technical basis to be developed that is specific to the FHTR. It is expected to be on the order of a few percent of all TRISO particles, which is consistent with the complete failure of one fuel bundle in the MHA for TRIGA reactors.
- Conservative analysis should be used to determine the release of fission products from the primary coolant. Release fractions of fission products from flibe in the maximum credible accident for the MSRE include 10% of iodine, 100% of noble gases, and 10% of solid fission products (Beall et al. 1964). The release of tritium during the MHA may also require consideration as the FHTR design matures.
- The leakage rate of the low-pressure containment is assumed to be a conservative value (1% per hour). This level is consistent with the value used for the MSRE, which included mechanisms to pressurize containment because of salt-water interactions. This leakage rate serves as the basis to determine dose to workers at the reactor site.
- The leakage rate of the building is conservatively assumed to be 10% per hour. This leakage rate, combined with assumptions of atmospheric transport, serves as the basis to determine off-site dose to members of the public.

It is clear from the sequence above that the proposed MHA is not dependent on the operation of any specific FHR systems during the accident. As the design develops, analysis should verify the conservative assumptions of the preliminary MHA analysis and ensure the regulatory dose limits for the class 104c operator license are satisfied.

4 FHTR

This chapter provides an overview of the FHTR mission, functional requirements, scoping analysis, startup testing, and ownership options.

4.1 FHTR Mission

An FHTR will provide the first demonstration and test of a salt-cooled reactor using high-temperature fuel. Because requirements will drive the design, mission, and strategy of the FHTR, a significant effort is being undertaken to define requirements and understand the tradeoffs for a practical design. The top-level requirements include (1) providing confidence that a commercial prototype reactor is warranted and (2) developing the necessary data for a larger, commercial-prototype reactor.

The FHTR also provides an opportunity to study neutronic, thermal hydraulic, and materials response phenomena as well as to gain experience and conduct testing of major reactor systems, which are vitally important to the reactor design and require demonstration:

- *Fuel handling*. That the reactor fuel floats in the coolant uniquely positions the FHTR to understand its implications and successfully demonstrate a refueling system.
- *Tritium control*. Tritium control methods were partially developed but not demonstrated in earlier molten salt reactors. Thus the FHTR needs to complete the process and have a reliable tritium control system to prevent tritium migration out of the reactor system.
- *Salt chemistry control*. A robust system for controlling coolant salt chemistry and salt volume will be necessary to prevent excessive corrosion in salt-facing components, especially heat exchangers. Because the salt coolant contains beryllium (with a possible design alternative of enriched zirconium), significant care should be taken to ensure this system meets the safety standards related to beryllium's high toxicity. The FHTR must be used to codify Hastelloy-N for use in a commercial nuclear reactor by generating the necessary data.
- *In-service inspection*. The FHTR will be needed to test inspection, maintenance, and repair, and demonstrate instrumentation capability.
- *Safety basis*. A final requirement of the FHTR is to demonstrate the safety basis for future FHRs through transient and operational tests.

The primary mission of the FHTR is reactor system performance testing to enable the design and licensing of an FHR commercial prototype. Meeting this mission requires detailed spatial and temporal mapping of temperature and neutron flux. Expanding the mission to a more general FHTR entails design of a high-temperature material testing reactor with fast flux test sections for materials testing and thermal flux test sections for fuel testing. Making the FHTR a general-purpose FHTR increases flexibility. The key questions for a general-purpose FHTR involve how large the reactor user base might be, compared to the user base for an FHTR designed to replicate the thermal hydraulic and neutronic phenomena associated with a specific FHR fuel class (e.g., pebble fuel or fixed fuel).

4.2 FHTR Functional Requirements

This section reviews the major functional requirements for the FHTR systems. In terms of safety functions, the FHTR functional requirements have a significant amount of overlap with those for the commercial prototype FHR, but the design requirements diverge on those functions that are major economic drivers. In contrast to the commercial mission, the primary functional requirements of the FHTR are to perform research and development tasks, within the scope of Section 31 of the Atomic Energy Act (as amended), that address fundamental viability questions around FHR technology. These research functions must be performed while maintaining the health and safety of those people involved directly in those tasks, as well as the general public.

A more specific FHTR design will include scaled systems that mirror most of the systems for the FHR commercial prototype design, while a general-purpose FHTR designed to irradiate fuels and materials may have substantially different system design and licensing needs. The FHTR systems will share those functional requirements relevant to safety, which are described in detail in the first FHR workshop white paper. The demonstration of these systems in prototypical conditions is one of the major research functions of the FHTR. Section 4.2.1 outlines the primary technology demonstration functional requirements for the FHTR systems.

The second major research objective for the FHTR is to provide data to validate models that can be adapted with high confidence for the design and licensing of the FHR commercial prototype. Thus, the FHTR includes a number of additional functional requirements for instrumentation that can provide the required data for a set of benchmark experiments. Section 4.2.2 details the preliminary set of experiments that the FHTR should be designed to perform.

4.2.1 Key System Demonstration Functional Requirements

The systems, subsystems, and components described in this subsection are those that have been identified as priorities for the FHTR to fulfill its mission to demonstrate the basic performance of the FHR concept. Those functional requirements required for safety overlap with those for the FHR commercial prototype and are detailed in the first FHR workshop white paper. This subsection attempts to identify systems, subsystems, and components that have additional functional requirements essential to the FHTR and diverge from those of the commercial design. (This list is not rank-ordered.)

- *Primary pump*. The primary pump must be controllable so that the flow rate can be adjusted for different experiments (see discussion on thermal hydraulic scaling). The FHTR must also have the capability to measure the total flow rate with acceptable uncertainty. Scaling of the FHTR to match the flow regimes of the commercial prototype FHR at full power may not be possible. The primary system of the FHTR, however, must be scaled so that heat transfer within the flow regime for natural circulation in the commercial prototype can be studied.
- *Graphite structures*. Graphite structures must be designed with a geometry that can accommodate the additional instrumentation and test samples for the FHTR.

- *Reactivity control system*. This system must be designed such that control rod position can be adjusted and measured with sufficient precision to meet the requirements of the FHTR experimental program.
- *DRACS*. Demonstration of coupled natural circulation in the primary system and DRACS is a key functional requirement for the FHTR. The DRACS loop will require sufficient instrumentation to measure flow rates and temperatures (both in the coolant and structures) for validation of simulation models.
- *Intermediate heat exchanger*. The FHTR's intermediate heat exchanger should include additional temperature instrumentation, as needed, for validation of FHR simulation models.
- *Balance of plant*. The FHTR will use the atmosphere as the ultimate heat sink. While the FHTR will be licensed with a Class 104c non-power license, it may be feasible to test innovative features of the proposed NACC. To meet the requirement of the Class 104c license, the FHTR must fall within the scope of research and development activities outlined in Section 31 of the Atomic Energy Act (as amended) and satisfy the condition in 10 CFR 50.22 that no more than 50% of the operational costs is devoted to the production of energy for sale and distribution.
- *Coolant chemistry, particulates, and inventory control*. The FHTR must provide a complete demonstration of the coolant chemistry control required for the commercial prototype FHR. The demonstration of this system is essential to show the compatibility of FHR material selection for the commercial design. The performance of this system also has important implications for the reliability of the commercial design.
- *Cover gas chemistry, particulates, and inventory control*. The FHTR must provide a complete demonstration of the cover gas control required for the commercial prototype FHR.
- *Fuel handling and storage system*. The FHTR must demonstrate pebble fuel handling, but the design of this system does not need to match that of the commercial prototype FHR. The demonstration of the fuel-handling system should meet the requirements for pebble accounting and provide data on the burnup distribution of the core. The system should also be able to monitor degradation and erosion for irradiated fuel pebbles.
- *Tritium management*. The FHTR will demonstrate possible methods for monitoring and managing the production, recovery, and release of tritium. Because the power level of the FHTR is much lower than that of the commercial prototype, the tritium production rate will be significantly lower and pose a reduced challenge to worker safety and environmental protection. The tritium management for the FHTR must satisfy the limits of dose to plant workers and the general public specified in 10 CFR 20.
- *Beryllium management*. The FHTR should demonstrate possible methods for monitoring and managing beryllium exposure to plant workers, unless enriched zirconium is used as a substitute for beryllium.

4.2.2 Key Experiments for the FHTR

One high-level functional requirement for the FHTR is the ability to perform a set of experiments that address fundamental questions associated with the FHR commercial prototype. The items below briefly describe the most important experiments for the FHTR that are required to satisfy the mission. Additional tests and experiments will be developed as the FHTR and commercial prototype designs mature and as additional questions arise after startup.

- *Reactor startup physics testing*. The FHTR will provide invaluable neutronic data that will be used to confirm and validate models used for the design and safety analysis of the commercial prototype plant. Reactor physics testing will constitute a significant portion of the startup plan as measurements are made to confirm predicted values for shutdown margin, differential and integral control rod worth, and reactivity coefficients. The primary objective of these data measurements is to validate the safe performance of the FHTR. Numerical errors from the predicted and measured values should be thoroughly reviewed to provide higher confidence in models used for future FHR neutronic design.
- *Steady-state power operation*. Once at full power, the FHTR will perform a variety of experiments to characterize the steady-state behavior of the system. These tests will help to validate simulations used in the design and optimization of the commercial prototype design. Characterization of steady-state conditions should include thermal hydraulic parameters such as flow rate, temperature, and pressure measurements, which will confirm the total power level of the system. These data should be collected at a range of coolant flow rates and temperature differences relevant for the design of the commercial plant. These measurements will also help to ensure that the FHTR is operating within the licensed technical specification.
- *Fuel burnup statistics*. Under normal operation, as pebbles are removed from the defueling chute, they will need to be characterized for burnup and either inserted into the active core or removed to the spent fuel storage facility. Statistics of measured burnup levels and periodic defueling tests will provide a set of data to compare to those statistics generated from the selected granular flow analysis tools. These experimental results will help to address regulatory concerns about the stochastic nature of pebble bed cores and, in particular, the characterization of those pebbles with the longest transient time through the core.
- *IETs*. The FHTR will demonstrate the basic safety case and viability of the FHR concept. One of the most important functional requirements of the FHTR will be to safely perform and measure the system response to a set of characteristic transients. These transients should include reactivity insertion events, LOFC, LOHS, and potentially, ATWSs. The data from these tests must be detailed and of sufficient quality to confirm and validate systems codes used in the safety analysis of the commercial FHR prototype. These data will be essential to both reactor designers and regulators in assessing the fundamental safety basis for FHRs.
- *Structural material testing and evaluation*. The FHTR will provide a demonstration basis for material performance in the commercial prototype environment. Test coupons of metallic and ceramic structural materials should be included to study long-term material interaction and the effects of irradiation. These test samples should be sufficient for

testing during the 20-year license period, with additional samples in place if the original license is renewed. Material tests can also be used to demonstrate the performance of materials such as SiC composites in non-safety-related functions.

- *Fuel testing and evaluation*. Following the initial process to bring the FHTR to full power, the facility may be used to test small numbers of fuel pebbles of interest to designers of the commercial prototype. Such tests would provide valuable data on pebbles in prototypical core conditions for FHRs, which could be used in future fuel qualification. Fuel testing of this nature would require additional safety analysis to ensure that the FHTR will operate within its technical specifications, though it is expected that small quantities of diverse pebbles will not significantly impact the overall core response.

4.3 FHTR Scoping Analysis

This section reviews the scoping analysis performed to date for FHTR neutronics and thermal hydraulics, to give an understanding of how key phenomena change between the test reactor and commercial prototype scales and the major tradeoffs.

As discussed earlier, there are two different test reactor design strategies: (1) a prototypical FHTR and (2) a general-purpose FHTR. Both designs of test reactors are being investigated. The analysis below is for a prototypical FHTR. Because many of the features of a prototypical FHTR are similar to the corresponding commercial reactor, the analysis of this option is further along.

Each option has advantages and disadvantages. Both options provide the data to validate design codes and demonstrate total system performance—the most important results from the FHTR. The general-purpose FHTR with driver fuel creates core conditions similar to the commercial reactor—but the use of driver fuel complicates analysis of some system phenomena.

4.3.1 FHTR Neutronics

The primary purpose of the FHTR is to provide adequate data and operational experience to eventually build, operate, and license a commercial prototype FHR, while operating safely. To that end, the FHTR must also be able to operate with key characteristics reflecting the neutron physics of the FHR to provide the neutronic environment necessary for an experimental basis.

4.3.1.1 Characteristic Neutronic Parameters During Normal Operation

During normal operation of the FHTR, physical similitude to the FHR is necessary for the acquisition of relevant data. Phenomena that have been identified as important in maintaining physical similitude include neutron energy spectrum and fuel and coolant temperature reactivity coefficients. Scoping studies thus far have been limited to reactor core volumes of 1 m^3—a figure chosen to allow the FHTR to reach a power of 20 MWth or lower and still achieve prototypical power density of approximately 20 MW/m^3.

Table 4-1 includes a number of characteristics that all cores in the scoping study have, compared to the FHR. The pebble diameter for the fuel in the FHTR is identical to the fuel in the commercial prototype FHR such that a small amount of test pebbles with fuel loadings identical to the commercial prototype FHR can be run through the FHTR with any variant in core fuel loading.

Table 4-1. Characteristic Operational Parameters Used in the Scoping Study for a Commercial Prototype FHR and an FHTR

Characteristic	Commercial Prototype FHR	FHTR
Power, MWth	900	20
Core active volume, m^3	56.0	1.0
Pebble diameter, cm	3.0	3.0
Enrichment, wt% $^{235}U/U$	19.9%	19.9%

By using identical fuel to that proposed for the commercial prototype FHR optimized for burnup (300 carbon to heavy metal ratio or C/HM) in the FHTR, key core physics will not match. A significant thermalization in neutron energy spectrum can be observed, as well as a shift to very different reactivity feedback coefficients. Scoping analyses at UCB have examined altering the core design to match necessary physical parameters. The shift to a more thermalized spectrum is a result of added moderation, which can be countered by either (1) shifting to lower leakage cores, where effects from the graphite reflector are less prominent; or (2) using lower C/HM-loaded fuels, where less moderation within the pebbles will lead to a shift towards a faster spectrum.

The fuel form modeled in the scoping analysis was an annularly loaded pebble fuel, shown in Figure 3-2. TRISO particles are packed around a porous inert sphere of graphite. The inert graphite sphere changes density for each fuel loading such that the pebble will be able to float in the coolant. The TRISO layer is then coated with a dense layer of graphite, which is the pebble shell. Table 4-2 compares variations in the fuel design with different heavy metal loadings. A number of variables can be changed to accommodate different heavy metal loadings in the pebble, so densities and packing fractions do not necessarily trend with heavy metal loading.

Table 4-2. Variations in Fuel Loading for the FHTR Scoping Analysis

Fuel Characteristics	**Variations**				
C/HM	75	100	200	300	400
Pebble diameter, cm	3.0	3.0	3.0	3.0	3.0
Annular region thickness, cm	0.92	1.08	1.24	1.27	1.25
Inert core radius, cm	0.483	0.318	0.157	0.132	0.150
Density of inert core, g/cm^2	0.5	1.13	1.52	1.61	1.65
Density of outer shell, g/cm^2	1.74	1.74	1.74	1.74	1.74
Fuel particle diameter, μm	810	810	810	810	810
Average temperature of fuel kernel, K	1,002.1	996.0	997.2	1,006.9	1,020.5

The reactor core design for the FHTR differs significantly from the baseline FHR core. The 1-m^3 core of the FHTR has no inner annular region. No blanket pebble layer was modeled in the scoping analysis, so all pebbles in the core would have the same composition. The scoping analysis varied pebble packing fractions at the wall to account for wall effects. The model roughly divides the core into three regions: a main cylindrical region, a converging section (at a 45° convergence), and a defueling chute. The model varies the core dimensions by the aspect ratio: the height of the cylindrical region divided by the radius of the cylindrical region. A summary of various core geometries modeled can be found in Table 4-3.

Table 4-3. Variations in Core Dimensions for the FHTR Scoping Analysis

Core Characteristic	**Variations**			
Aspect ratio	1.3	1.414	2.0	3.5
Cylindrical region radius, cm	58.0	56.8	51.5	43.7
Cylindrical region active height, cm	75.5	80.3	103.1	152.9
Converging region angle, degrees	45	45	45	45
Converging region height, cm	43.0	41.8	36.5	28.7
Defueling chute radius, cm	15	15	15	15

A summary of the results of the scoping analysis can be found in Appendix B. The scoping analysis showed that no single-core design can match the FHTR physics to the commercial prototype FHR. An alternative method is to operate the FHTR with multiple cores. Given the pebble form of the fuel, cores can be changed throughout and during operation of the FHTR, and a transition between cores without a shutdown is feasible. A transition in core dimensions is also possible depending on the ability to access the graphite reflector. As such, a core can be designed to match single phenomena to the commercial-scale FHR core. Table 4-4 matches some representative cores from the scoping analysis to physical parameters of the commercial prototype FHR core, and Figure 4-1 compares the neutron energy spectra between each case. All of these cores have an aspect ratio of 1.414, but as cases B1 and B2 display, multiple cores could be used to achieve the necessary physics to reflect the commercial prototype FHR. Case A uses the same fuel loading as the commercial prototype FHR, Case B attempts to match the coolant void reactivity coefficient, and Case C attempts to match the neutron energy spectrum. Choosing which core configuration to use will again depend on a number of other parameters, including safety and feasibility. Cores B1 and B2, for example, have significantly different fuel loadings, such that B2 is overmoderated, which may have other effects on the core physics and safety.

Table 4-4. Candidate Cores for the FHTR

Core	**C/HM**	**α_{void}, pcm/K**	**α_{fuel}, pcm/K**
Commercial prototype FHR	300	-0.49 +- 0.1	-4.5 +- 0.2
Core A	300	+0.09 +- 0.07	- 1.24 +- 0.06
Core B1	200	-0.25 +- 0.06	-1.58 +- 0.06
Core B2	550	-0.44 +- 0.06	-1.12 +- 0.07
Core C	75	-3.62+- 0.07	-3.61+- 0.07

It is clear from the scoping analysis of the representative FHTR that a single core will not match the physics. At the equilibrium cycle, the neutron energy spectrum and the temperature reactivity coefficients of the fuel and coolant vary significantly with C/HM loading in the fuel. Other cores may also be chosen to match other representative parameters of the commercial prototype FHR physics but have not been investigated.

4.3.1.2 Reactivity Control and Shutdown

The most important feature of the FHTR is to operate with and maintain safety throughout the lifetime of the core. In addition to negative fuel, coolant, and moderator reactivity coefficients, a number of shutdown mechanisms must be available to provide adequate safety mechanisms. Mechanisms to control reactivity in the core must be able to compensate for the following effects: (1) densification of coolant to freezing point and/or ambient temperature, (2) fluctuations in power during normal operation, (3) maintenance of criticality from beginning of

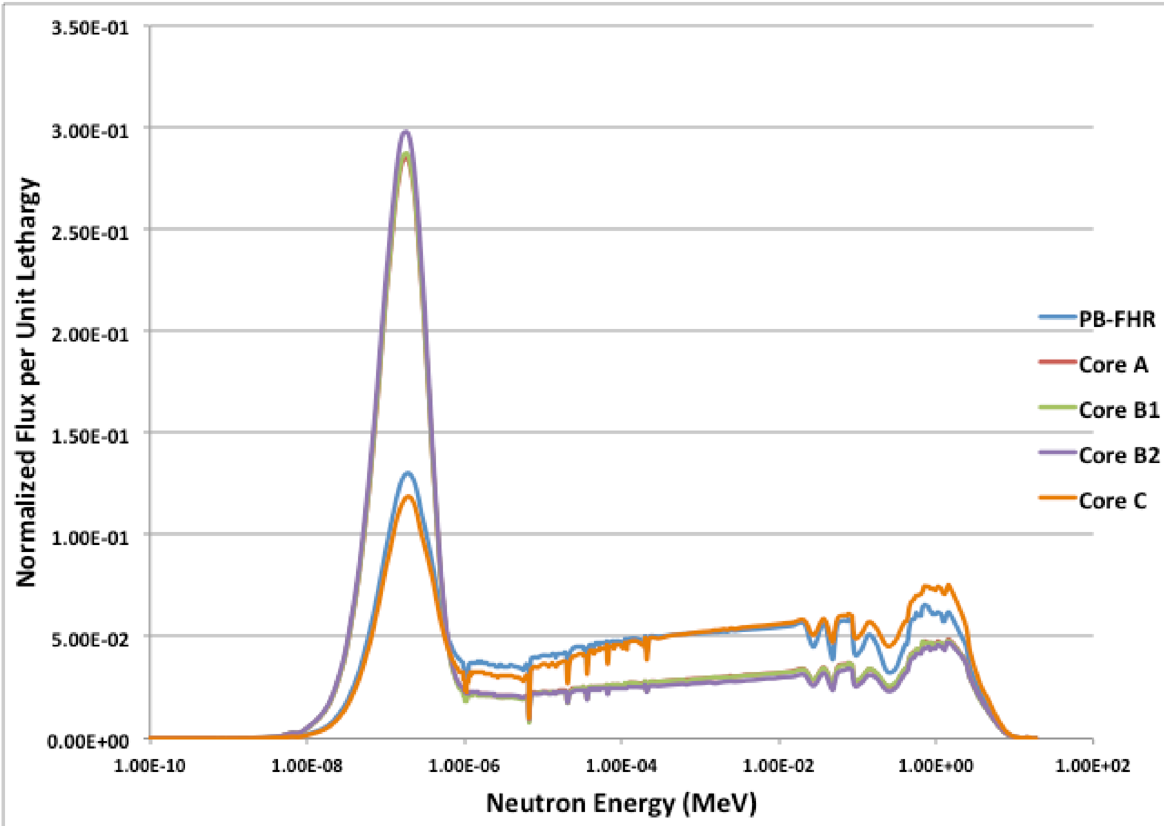

Figure 4-1. Comparison of Neutron Energy Spectrum Between Candidate Core Designs

life to end of life, (4) fission product buildup during operation, and (5) fission product decay post-shutdown.

The primary reactivity control system proposed for normal operation of the FHTR is a series of control rods that can be inserted into channels in the outer graphite reflector. Unlike the commercial prototype FHR, the FHTR would not have an inner graphite annulus, so the control rods must be on the core periphery. For the scoping analysis, the model assumed the reflector had 12 control rod channels. A view of the MCNP model is shown in Figure 4-2.

Another option for the peripheral control rod system is a series of buoyantly driven control rods that can be gradually inserted into coolant-filled channels in the reflector, based on temperature-driven shifts in the coolant density. For the neutronic scoping analysis of the FHTR, these control rods behave similarly to manually driven control rods. Should significant design differences arise between a buoyantly driven control rod and a manually driven rod, the model would need to be changed accordingly.

The scoping analysis (see Appendix B) showed that peripheral control rods do not have sufficient negative reactivity to maintain subcriticality after the decay of xenon post-shutdown. Another system that has been investigated is a series of control blades that can be inserted directly into the pebble bed for shutdown purposes. Because these blades will affect pebble movement in the core, they will not be ideal for use during normal operation. However, the

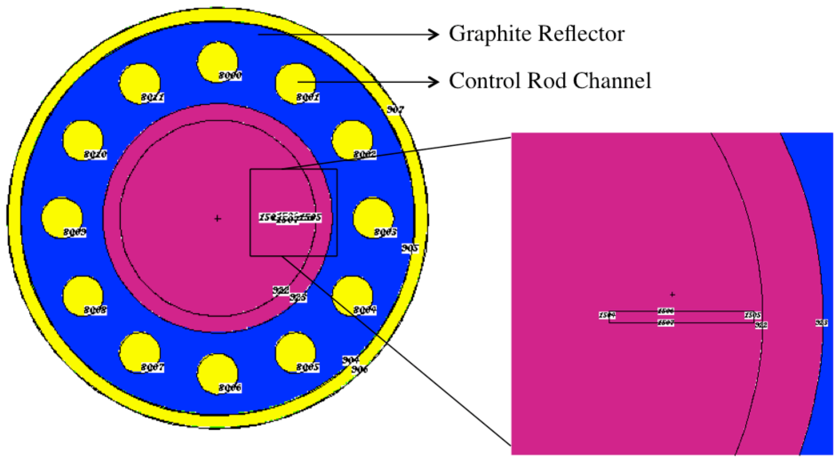

Figure 4-2. Axial Slice of the FHTR with Control Rod Channels and Control Blade Insertion

scoping analysis showed that the control blades have enough negative reactivity to compensate for positive reactivity changes attributed to the solidification of flibe and xenon decay in the core post-shutdown. Experiments are being conducted to investigate the feasibility of inserting such a blade directly into the pebble bed. Figure 4-2 also includes a view of a single control blade insertion. If this option is pursued, the FHTR will likely have more than one blade available for shutdown.

Burnable poisons are another option for reactivity control in the FHTR. However, the scoping analysis did not examine the feasibility. This approach could be included in future scoping analyses, once a decision is made as to which methods will be used to achieve criticality and start up the FHTR.

It is clear that no single mechanism will account for all necessary requirements for reactivity control in the core, so using some combination of these reactivity control systems will be necessary.

4.3.1.3 Future Work

The scoping analysis was limited to matching equilibrium cores of the FHTR to equilibrium cores of the commercial prototype FHR. Given that the neutron energy spectrum, reactivity coefficients, fission product inventory, and other parameters will evolve throughout the FHTR lifetime, the physics must be tracked as a function of core composition, including burnup. Because the coolant reactivity coefficient is heavily dependent on the fuel-to-moderator ratio, the spectrum and reactivity coefficient will also evolve depending on the method used to approach criticality in the core and the transition between different cores. Options for startup testing for the FHTR can be found in Section 4.4. Analyses of these options will be necessary to determine the final design of the FHTR. Future analyses must include options for the FHTR beginning of life, middle of life, and end of life, once a startup procedure is chosen. Studies investigating the sensitivity of the FHTR to a number of variables are also being performed at UCB.

Additional future work could include investigating possible alternatives to fuel forms and types of coolants in the FHTR. However, depending on the FHTR mission, these may need to match the commercial prototype FHR, and thus will depend on the prototypical design. The design of the FHTR will therefore iterate between the desired prototypical design of the FHR and its functional requirements/mission.

4.3.1.4 Applicability of the FHTR as a Neutronic Validation Experiment for the FHR Commercial Prototype

Because of its much lower thermal power and smaller size, the FHTR cannot simultaneously achieve similitude for every neutronic response of interest (e.g., multiplication factor, void and temperature reactivity coefficients, conversion ratio, etc.) with the FHR commercial prototype. However, the FHTR core can be designed to operate with multiple critical configurations that can validate reactor physics models over a relatively wide parameter space. The FHTR must also be designed to reproduce other important phenomena (e.g., thermal hydraulics, structural mechanics, materials, and fuels), and while the overall prioritization will be complex, the prioritization will be easier if the FHTR design can accommodate multiple fuel designs, configurations, and operating conditions. Design parameters (e.g., pebble C/HM, fuel enrichment, core aspect ratio, pebble configuration, and system carbon-to-flibe atom ratio) can be varied so the necessary validation phase is bounded by the range of FHTR configurations. Each configuration shall be safe to operate, with negative reactivity coefficients.

Different configurations of the FHTR core may match the FHR in

- Pebble C/HM
- Fuel, coolant, and reflector flux spectra
- Fuel and coolant temperature reactivity coefficients.

The validation phase is composed of

- Flux spectra within the fuel, pebble, coolant, and reflector
- Neutronic response sensitivities
- Nuclear data uncertainty-weighted sensitivities.

Overall, the core design optimization process will proceed as follows:

- Determine aspects of the validation phase of the FHR commercial prototype configurations (fresh and equilibrium) and existing FHTR point designs
- Identify the aspects that are not covered
- Design new FHTR configurations that fill the un-validated aspects
- Compare phi(E) in fuel/flibe

- Compare phi(E) as f(r).

4.3.2 FHTR Thermal Hydraulics

This subsection reviews results for the FHTR scoping analysis thermal hydraulics and discusses the scaling issues for replicating phenomena important at commercial prototype scale. Because there is no specific design for the complete FHTR at this point, detailed analyses have only been performed for the core, and other components will be modeled in the future.

4.3.2.1 Thermal Hydraulic Performance Under Normal Operations

The FHTR core would be mainly cylindrical, which helps predict the flow dynamics through the core analytically. One of the main drivers for the core design is the neutronics similitude with a commercial reactor, which was discussed in Subsection 3.6.2. Therefore, the thermal hydraulic scoping analysis was based on four different core geometries that were investigated for the neutronic scoping analysis. The global geometry is shared between those designs, with one constant cross-section cylindrical region (the "active region"), one converging region, and the defueling chute, as shown in Figure 4-3. The designs differ through the aspect ratio of the core. These parameters are summarized in Table 4-5.

The FHTR core would have an approximate volume of 1 m^3 and a nominal power of 20 MWth, compared to the target nominal power of 400 MWth for a commercial prototype FHR. Both cores would be axisymmetric. However, the FHTR would not have cross-flow in the core. Indeed, cross-flow is needed in the commercial prototype FHR to reduce the pressure drop across the core, thus enhancing natural circulation under decay heat removal conditions. However, the smaller dimensions of the FHTR would greatly reduce the pressure drop across the pebble bed, hence no cross-flow would be needed in the core. Pressure drop comparisons are provided in Table 4-6 to illustrate this fact. Because of the difference in pressure drop, pumping power in the FHTR will be significantly reduced compared to the commercial prototype FHR, as shown in Table 4-6 for an ideal pump (efficiency $\eta = 1$).

Pressure drop here is calculated using the relationship

$$\boldsymbol{f_K = \frac{K^{1/2}}{\rho_f u_D^2}\left(-\frac{dp}{dz}\right)} \tag{4-1}$$

where *K* is the permeability of the pebble bed (in square meters), defined as

$$\boldsymbol{K = \frac{\varphi^3 d_p^2}{180(1-\varphi)^2}} \tag{4-2}$$

with φ the porosity of the bed and d_p the pebble diameter. ρ_f is the average density of the coolant, u_D is the superficial velocity of the coolant in the core, and f_K is the friction factor, derived from Ergun's correlation for pebble beds (Nield and Bejan 2006)

$$\boldsymbol{f_K = \frac{1}{\mathrm{Re}_K} + \frac{1.75}{\sqrt{180\varphi^3}} \approx \frac{1}{\mathrm{Re}_K} + 0.5156} \tag{4-3}$$

where Re_K is the Reynolds number using permeability of the bed, defined as

$$\mathbf{Re}_{\boldsymbol{K}} = \frac{\rho_f u_D K^{1/2}}{\mu_f} \tag{4-4}$$

with μ_f the average density of the coolant.

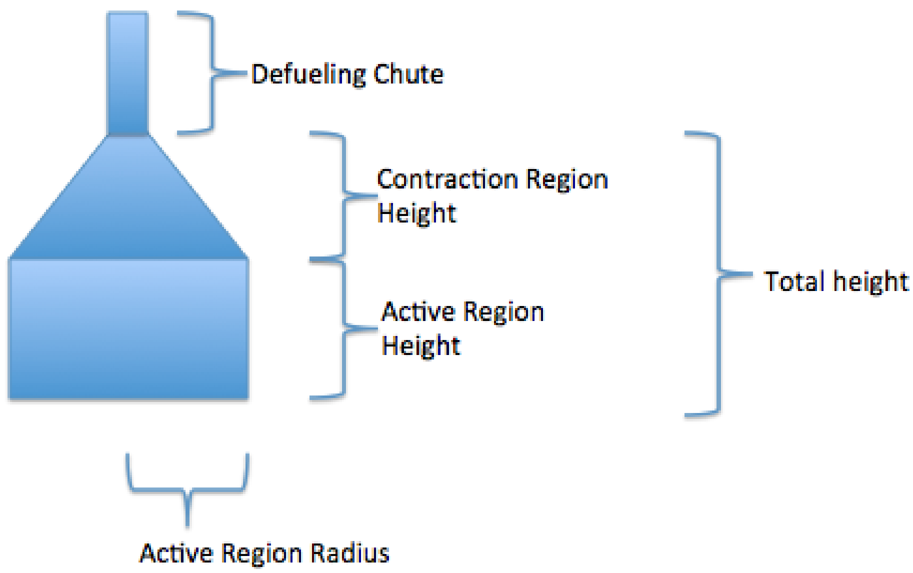

Figure 4-3. Layout of the FHTR Core

Table 4-5. Geometrical Parameters for Four FHTR Designs

Parameters	Core #1	Core #2	Core #3	Core #4
Defueling chute radius, cm	15.0			
Cylindrical region height, cm	75.5	80.3	103.1	152.9
Converging region height, cm	43.0	41.8	36.5	28.7
Total height, cm	118.5	122.0	139.6	181.6
Active radius, cm	58.0	56.8	51.5	43.7
Aspect ratio	1.3	1.414	2.000	3.5

The working fluid is flibe in both the commercial prototype FHR and the FHTR. The coolant is expected to run at an inlet temperature of 600°C and an outlet temperature of 700°C. The pebble fuel is also expected to share the same design, hence the same thermophysical properties between the two reactors. Because the FHTR is expected to run at similar temperatures as the commercial prototype FHR, the mass flow rate of coolant through the core can be decreased in proportion to the total power. One observation of the information provided in Table 4-6 is that the difference in Reynolds number between the two systems will result in a different impact of fluid drag forces on the pebble dynamics, and heat transfer between the coolant and the pebbles will also be affected, as illustrated by the difference between the Nusselt numbers for the two systems, calculated using the Wakao correlation (Wakao and Kaguei 1982)

$$\mathbf{Nu = 2 + 1.1Re^{0.62}Pr^{0.33}} \quad \textbf{(4-5)}$$

and shown in Table 4-6. However, Reynolds numbers between the FHTR and the commercial prototype FHR cores are of the same order of magnitude, which means that flow regimes will be similar in both systems ($Re > 200$ is turbulent flow for pebble bed systems), and no significant distortion is expected with respect to flow dynamics and heat transfer phenomena.

One solution to match the Reynolds number between the FHTR and the commercial prototype FHR would be to reduce the temperature change across the FHTR core and maintain the average coolant temperature by increasing the primary pump power, hence increasing the mass flow rate and Reynolds number in the FHTR while keeping the average Prandtl number constant. This change will impact thermal transients though, and needs further analysis in the future. A larger range of Reynolds numbers can also be covered by varying the FHTR core power, although the nominal value of 20 MWth is expected to be the maximum achievable power generation.

Peak temperatures in the pebble fuel would not be a concern during normal operation of the commercial prototype FHR because of the large thermal margins of the TRISO fuel. In the FHTR, because of a lower Reynolds number in the core, heat transfer between the coolant and the fuel will be reduced, and peak temperatures in the fuel are expected to be higher than in the commercial prototype FHR. However, because heat transfer coefficients are expected to be of the same order of magnitude between the commercial prototype FHR and the FHTR, this small distortion is not expected to affect the behavior of the core significantly. Temperature distribution in the fuel will be further assessed through subsequent analyses, and models will be validated through fuel temperature measurements in the FHTR core. Also, because heat transfer coefficients depend on Reynolds number, which is related to temperature-dependent viscosity, the temperature distribution of the coolant in the FHTR will be analyzed and compared to the commercial prototype FHR in future analyses.

Outside of the core, the precise design of the primary and intermediate loops, DRACS loops, heat exchangers, power conversion system, and other subsystems has not been determined, hence it is difficult to provide a correct thermal hydraulic scoping analysis for these subsystems. Scaling of the components, and their estimated range of operational parameters, will then be determined through validated thermal hydraulic models to replicate the reaction of the system to normal operating events such as power load changes (coupling between the core and the power conversion system).

Table 4-6. Design and Scaling Parameters for the Commercial Prototype FHR and the FHTR

Parameter	Commercial Prototype FHR	FHTR
Working fluid	Flibe	Flibe
Thermal power, MWth	400	20
Geometry	Axisymmetric (cross-flow)	Axisymmetric (axial flow)
Inlet temperature, °C	600	600
Outlet temperature, °C	704	704
Fluid density, kg/m^3	1,990-1,940	1,990-1,940
Pebble density, kg/m^3	1,680-1,810	1,680-1,810
Pebble fluid density ratio	0.84-0.93	0.84-0.93
Pebble diameter, m	0.03	0.03
Core cross-section area, m^2	8.1*	0.60-1.06
Bed packing fraction	0.6	0.6
Mass flow rate, kg/s	1,609	80
Re_d in core	890	340-600
Pressure drop across pebble bed, kPa	17	1.4-8.5
Core (only) pumping power, kW	13.7	0.06-0.35
Nusselt number in the core	160	91-127

* Scaled down from the 900-MWth Pebble Bed Advanced High-Temperature Reactor (PB-AHTR).

4.3.2.2 Thermal Hydraulic Performance Under LBEs

Key thermal hydraulic LBEs were listed in Subsection 3.6.1. The role of the FHTR is to replicate the system's response to those events, through scaled figures of merit. This subsection reviews those LBEs and how the FHTR compares to the commercial prototype FHR in terms of initiating events and parameters that characterize the response of the system to these events. Because thermal hydraulic transient response of the FHTR to LBEs has not been formally modeled to date, this scoping analysis is based on fundamental principles and should be pursued in the future.

Protected LOFC. Protected LOFC occurs when one or more pumps in the primary loop are tripped and the reactor scrams. This event will be replicated with the FHTR by stopping the same number of pumps, starting from normal operating pumping power as shown in Table 4-6. The main requirement is to scale the time to establishment of natural circulation in the primary loop and the DRACS loop between the FHTR and the commercial prototype FHR, based on relative residence time of the coolant in the reactor between the two systems under normal operation, which depends on the design of the primary loop. Further analysis must be performed to calculate this parameter.

The performance of the DRACS for decay heat removal will be assessed with the FHTR. Each DRACS heat exchanger in the FHTR will be downsized compared to the commercial prototype versions to extract 2% of the FHTR nominal power. This performance can be assessed by matching the Grashof number between the FHTR and the design of the commercial prototype system in the primary, intermediate, and DRACS loops:

$$\boldsymbol{Gr} = \frac{\rho_0^2 g \beta \Delta T_{sink}}{\mu_0^2} \cdot \boldsymbol{d^3} \qquad (4\text{-}6)$$

where ρ_0 is the average density, g is the gravity constant, β is the coefficient of thermal expansion, ΔT_{sink} is the temperature difference between the heat exchange surface and the fluid in the loop, d is the hydraulic diameter, and μ_0 is the dynamic viscosity. All parameters are calculated at the coolant average temperature. If flibe is used as the primary coolant at similar temperatures in both the commercial prototype FHR and the FHTR, design of the heat exchangers will set the hydraulic diameter, hence ΔT_{sink} at each heat exchanger, which will ultimately inform the operating temperature conditions in the intermediate loops and the DRACS loops of the FHTR compared to the commercial prototype FHR. This factor will be analyzed when a design is chosen for the intermediate heat exchangers and the DRACS heat exchangers in the commercial prototype FHR. Under natural circulation, all parameters are fixed depending on power, as a solution to the steady-state momentum equation:

$$\mathbf{m^3} = \frac{\rho_0^2 \beta g}{c_p} \frac{\Delta z_{HL} \cdot Q}{\sum_n F_n'(m, T(s))} \qquad (4\text{-}7)$$

$$\mathbf{\Delta T} = \frac{Q}{m c_p} \qquad (4\text{-}8)$$

where m is the mass flow rate, Δz_{HL} is the elevation difference between the centers of the core and the DRACS heat exhanger, Q is the extracted power, c_p is the specific heat capacity, F'_n is a characteristic friction term in each segment of the loop, and ΔT is the temperature rise across the core. All parameters are calculated at the coolant average temperature.

From these equations, based on the proposed geometries of the FHTR core listed earlier, the elevation difference between the centers of the core and the DRACS heat exchanger in the FHTR will have to be 19% to 70% of that of the commercial prototype FHR. As mentioned, one option to match the Reynolds number between the FHTR and the commercial prototype FHR would be to reduce the temperature rise across the core. In these conditions, Δz_{HL} and the DRACS heat exchanger size would have to be increased in the FHTR. If the temperature rise across the core is decided to be similar between the FHTR and the commercial prototype FHR, so will be the temperature difference in the DRACS loop. Otherwise, the operating temperatures in the DRACS loop will be calculated from the scaling between temperature differences in the FHTR and the commercial prototype FHR.

Protected LOHS. Protected LOHS occurs when normal cooling through the intermediate loop is lost. This event will be replicated with the FHTR by stopping the same number of intermediate pumps, starting from normal operation pumping power. Because very little design analysis has been performed to date for the intermediate loops and their coupling to the core and the power conversion system, no scoping analysis has been performed for this transient. One key element will be to scale the time to establishment of natural circulation in the primary loop and the DRACS loop between the FHTR and the commercial prototype FHR, based on relative residence time of the coolant between the two systems under normal operation, as under protected LOFC. Most scaling parameters have been discussed for the protected LOFC case. In addition to those, the normal shutdown cooling system will need to be scaled for the FHTR, as the primary way to extract decay heat from the core if the intermediate loops were to fail. Scoping analyses for geometry of the normal shutdown cooling system and pumping power in the FHTR will be performed when the commercial prototype normal shutdown cooling system is designed.

Overcooling Events. If temperatures in the FHTR loops are similar to those in the commercial prototype FHR, and if the same working fluids are used as expected, then overcooling events should follow the same dynamics in both systems. However, because heat exchangers will be downsized between the commercial prototype FHR and the FHTR, and heat exchanger surfaces are the most likely to freeze, the flow blockage between the FHTR and the commercial prototype FHR will be distorted. As a result, reaction of the FHTR system to overcooling transients can be treated as a conservative case of coolant freezing when scaled up to the commercial prototype FHR.

Large Loss of Primary Coolant. Large loss of primary coolant occurs when there is a large rupture of the reactor vessel, under the assumed pool design. This BDBE will not be tested with the FHTR; therefore, no scoping analysis will be needed for this case.

4.4 FHTR Startup Testing

The startup of the FHTR will have two major phases:

- *Non-fuel startup*. The first phase will be startup with dummy fuel. The goal is to test all system components (pumps, refueling systems, instrumentation systems, reactor inspection systems, salt cleanup systems, etc.). Because of the lack of experience in operating large systems with hot salt, this test program will be more extensive than seen in LWRs. Some systems, particularly the salt cleanup system, may see heavier duty than when the FHTR goes critical.
- *Nuclear startup*. During the startup of the FHTR, a number of subcritical and low-power reactor physics tests will be performed. Initially, the core will be brought to a critical state with a 1/M procedure, by which the system subcritical multiplication is measured incrementally along the approach to criticality. Next, the differential reactivity worth of each control rod will be measured with dynamic low-power supercritical period tests.

Afterwards, integral shutdown reactivity worths can serve as confirmatory measurements. The reactivity deviation of a variety of off-normal static and dynamic configurations will then be measured using the control rod worth curves. Lastly, reactivity worth related to validation will be measured. Characteristics of each of these procedures that are unique to the FHTR, the accuracy of historical startup testing, acceptance criteria and uncertainty quantification, and intrinsic neutron sources are outlined in Appendix C.

4.5 FHTR Ownership

Ownership of the FHTR has implications for the reactor's ultimate mission, funding, and licensing strategy. The options for ownership include the government; the government with international partner;, a university, industry, or a consortium; or a combination of these approaches. This section reviews a handful of research and test reactors including ATR, HFIR, EBR-II, and Jules Horowitz facilities. Table 4-7 contains a list of these and a few other facilities with various owners, operators, and regulators that are included in the International Atomic Energy Agency (IAEA) research reactor database.[1]

Table 4-7. Selected Research and Test Reactors*

Reactor	Owner	Operating Organization	Regulatory Body	International Safeguards	Type	Power, MW	$\Phi_{thermal, max}$/ $\Phi_{fast,max}$	Status
NTR General Electric	GE	Vallecitos Nuclear Center	NRC	NRC	Graphite	0.1	2.5E12/ 5.0E11	O
Dow TRIGA	Dow Chemical Company	Dow Chemical Company	NRC	NRC	TRIGA Mark I	0.3	5.0E12	O
UC Davis/ McClellan Nuclear Research Center	University of California	University of California at Davis	NRC	NRC	TRIGA Mark II	2	3.0E13/ 1.0E14	O
Annular Core Research Reactor	DOE	Sandia National Laboratories	DOE	DOE	TRIGA ACRP	4	4.0E12/ 4.0E13	O
Army Material Research Reactor	Army Materials & Mechanics Research Center	U.S. Army	NRC	NRC	Pool	5	5.0E13/ 1.5E14	D

[1] Information from the IAEA's Research Reactors website at http://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx. Accessed May 24, 2012.

Reactor	Owner	Operating Organization	Regulat-ory Body	Inter-national Safeguards	Type	Power, MW	$\Phi_{thermal, max}$/ $\Phi_{fast,max}$	Status
BAWTR	Babcock & Wilcox	Lynchburg Research Center	NRC	NRC	Pool	6	-/-	D
MITR II	MIT	MIT	NRC	NRC	Tank	6	7.0E13/ 1.7E14	O
ASTR	General Dynamics-Convair	U.S. Air Force	-	-	Tank	10	-/-	D
Fast Burst White Sands	U.S. Depart-ment of Defense	U.S. Army	DOE	DOE	Fast Burst (Air Cooled)	10	-/-	O
NBSR	National Institute of Standards and Technology	National Institute of Standards and Technology	NRC	NRC	Heavy Water	20	4.0E14/ 2.0E14	O
HBWR (Norway)	Institutt for Energitekni kk	Institutt for Energiteknikk	Statens Stralevern	IAEA	Heavy water	20	1.5E14/ 0.8E14	O
EBR-II	DOE	Argonne West National Lab	DOE	DOE	Breeder (Sodium Cooled)	62.5	-/ 2.5E15	D
HFIR	DOE	ORNL	DOE	DOE	Tank	85	2.5E15/ 1.0E15	O
Jules Horowitz (France)	CEA	CEA	ASN	ASN	Tank in pool	100	5.5E14/ 1.0E15	UC
ATR	DOE	Bechtel BWXT Idaho, LLC, ATR	DOE	DOE	Tank	250	8.5E14/ 1.8E14	O
Fast Flux Test Facility	Westinghou se Electric Corporation	Westinghouse Hanford Co.	DOE	DOE	Fast sodium cooled	400	4.6E15	S

*Definitions: D=decommissioned; UC=under construction; O=operational; S=shut down; CEA=Commissariat à l'énergie atomique (France); ASN=Autorité de Sûreté Nucléaire (France)

4.5.1 Advanced Test Reactor

Since 1949, when what is now called INL was established as the National Reactor Testing Station, the site has housed 52 reactors including the ATR built in 1967[1]. ATR currently stands as the largest test reactor in the world at 250 MW. It is owned and regulated by the DOE and operated by Bechtel BWXT Idaho, LLC, the DOE's management and operations contractor at INL. The purpose of the ATR is material and fuel testing, especially simulating long-term radiation exposure. It also produces rare isotopes for medical purposes. In April 2007, the ATR was designated a National Scientific User Facility to increase use by universities, laboratories, and industry. Researchers can submit proposals through a peer-reviewed proposal process, which closes for review twice a year. Judged based on feasibility, technical merit, relevance to DOE Office of Nuclear Energy programs, and cost, accepted proposals award researchers cost-free access to the ATR, PIE facilities, or one of the partner facilities.

4.5.2 High Flux Isotope Reactor

The HFIR[2] was built in the 1960s to produce transuranic isotopes. Its purpose also now entails materials irradiation, neutron activation, and neutron scattering. It hails as one of the sole producers of the isotope ^{252}Cf in demand for cancer therapy and detection of pollutants in the environment and explosives in luggage. The fusion energy program furthers research studies with the reactor as well. Current reactor full power is 85 MW.

The HFIR was built at ORNL in Oak Ridge, Tennessee, at the recommendation of the U.S. Atomic Energy Commission (AEC) to accelerate the transuranic production program. ORNL submitted a proposal to the AEC and gained approval to design a high-flux reactor. The government agency responsible for the reactor shifted after the Energy Reorganization Act of 1974 and the Department of Energy Organization Act of 1977. It now falls under DOE ownership and regulation with operations by ORNL.

4.5.3 Experimental Breeder Reactor-II

The Experimental Breeder Reactor II (EBR-II) operated as an experimental liquid sodium fast breeder reactor from 1964 to 1994 (Walters 2009). It was designed as 62.5-MW reactor that would also produce about 20 MW of electricity. Staff at Argonne National Laboratory's Idaho campus constructed and operated the reactor under DOE ownership and regulation. Initially, the EBR-II successfully demonstrated the breeder-reactor concept (1964 to 1969). It then expanded its mission to include steady-state fuels and materials testing (1970 to 1978) and operational testing (1979 to 1986). Finally slated as the Integral Fast Reactor prototype (1982 to 1994), the EBR-II funding was cut in 1994, 3 years before completion of the Integral Fast Reactor program.

4.5.4 University Reactors

The Massachusetts Institute of Technology Reactor (MITR) currently operates as a research reactor at MIT. It is owned and operated by the university with regulatory oversight by the NRC. Current research includes boron neutron capture therapy, in-core experiments, trace element analysis, and neutron transmutation doping of silicon. It is the second largest university reactor in

[1] Information from the Advanced Test Reactor National Scientific User Facility website at http://atrnsuf.inl.gov/About/tabid/37/Default.aspx. Accessed May 24, 2013.

[2] Information from the Oak Ridge National Laboratory's History of HFIR website at http://neutrons.ornl.gov/facilities/HFIR/history. Accessed May 24, 2013.

the U.S. at 6 MW behind the University of Missouri Research Reactor at 10 MW and plans to convert from high-enriched uranium to low-enriched uranium fuel.[1]

The University of California at Davis's McClellan Nuclear Research Center (MNRC) reactor was originally developed and owned by the U.S. Air Force to detect corrosion and defects in aircraft structures using neutron radiography. Currently it's owned and operated by the University of California at Davis. Beginning operation in 1990, it holds claim as the newest research reactor in the U.S.[2] It is located on a former U.S. Air Force base that is now an industrial park.

4.5.5 Jules Horowitz Reactor

The Jules Horowitz reactor will serve as a European research reactor for an international consortium of research institutes (CIEMAT in Spain, SCK in Belgium, NRI in the Czech Republic, VTT in Finland, the CEA in France, and IAEC in Israel), utilities and industry partners [Electricité de France (EDF), AREVA, and VATTENFALL], two associate partners (DAE in India and the Japan Atomic Energy Agency), and the European Commission.[3] The reactor will be located in France at the Cadarache site with expected completion in 2014 (Gaillot et al. 2010). It will operate as a 100-MW reactor primarily for materials testing but also for fuel testing and the production of radioisotopes for medical purposes. The JHR consortium finances construction and guarantees members access to the facility for proprietary experiments. Additionally, a joint program for international collaboration promotes research opportunities. It will be operated and owned by CEA, which is the French counterpart to the DOE.

4.5.6 Ownership Options

Based on the history of research and test reactors, several options for FHTR ownership—each with its own set of advantages and disadvantages—emerge according to the purpose and mission of the FHTR as well as the licensing and funding strategy.

A privately owned test reactor, by a university, industry, or a consortium, requires licensing by the NRC. If the ultimate goal is solely to make viable a commercial prototype FHR, then a benefit to private ownership would be up-front interaction with the NRC. The NRC interaction would be less than if licensing a commercial reactor and would be licensed by a different group within the NRC, but it would keep the NRC informed and more familiar with the FHR design. Private reactors tend to maintain very specific purposes [e.g., neutron activation analysis (NAA) or radioisotope production] and to operate at low power. The costs of construction and licensing were considerably less when companies like Dow Chemical and GE built their reactors. Thus, private ownership may also require public funding or a partnership funding strategy.

No test reactor has ever been operated and owned by a university, and economically this is an unlikely option because the large financial costs associated with test reactors exacerbate a

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[1] Information from the MIT Nuclear Reactor Laboratory website at http://web.mit.edu/nrl/www/reactor/reactor.html. Accessed May 24, 2013.
[2] Information from the UC Davis McClellan Nuclear Research Center website at http://mnrc.ucdavis.edu/about.html. Accessed May 24, 2013.
[3] Information from the Jules Horowitz Reactor website (English version) at http://www.cad.cea.fr/rjh/index.html. Accessed May 24, 2013.

university budget. As seen in Table 4-7, any reactor with power levels greater than 10 MW was developed with governmental support.

If the FHTR is under DOE ownership, then it can be built on an existing DOE site. This location can be a cost savings because there is no need to go through a reactor siting selection process or to acquire land. Also, some DOE facilities have existing infrastructure for reactors, including access to out-of-core examination facilities and hot cells. DOE funding carries the burden of being beholden to the congressional budget cycle and political process. Cost and funding factor heavily in determining the ownership of the FHTR.

Private ownership would suit the sole mission of paving the way for a commercial prototype FHR. On the other hand, a potentially more attractive mission is that of a general-purpose FHTR, which expands the options for funding and ownership. Securing government ownership and/or funding or a consortium agreement between many parties may necessitate a broad mission to attract enough interest and sources of funding. Support from industry decreases as the return on investment lengthens, thus a general-purpose FHTR secures interest with more near-term application.

The option of having DOE ownership with international partners is a distinct possibility. Several countries are building different types of test reactor systems, including the CAS FHR test reactor. This option of international support for the FHR could lead to an agreement between one or more countries in which the expense and utilization of the FHTR would be shared. The European JHR provides as an example of such a model. A consortium of partners, with or without international components, appeals for cost sharing the construction, maintenance, and operation and for increasing reactor application beyond testing FHR functional requirements.

Lastly, combining several of these options warrants consideration. Benefits of the different ownership options can be combined. For example, the DOE has land and existing infrastructure to save costs, and licensing the reactor under the NRC could help with future licensing of a commercial prototype reactor by having the NRC involved early in the project. This combination could also include private ownership and funding to circumvent potential congressional budget shortfalls and political issues. With more interests in the FHTR, it can conceivably be reasoned the more attractive the project and the less likely it is to fail.

5 FHR Commercialization Roadmap

This chapter presents the foundation of the FHR commercialization roadmap. The starting point is asking the question: "What is the market for the FHR?" The traditional answer to that question has been the production of baseload electricity. The unique technical characteristics of the FHR, which enable it to be coupled to a NACC, allows a different answer. That answer, as discussed in Chapter 2, is the production of baseload electricity, peak electricity, electricity regulation, and process heat. It is a radically different strategy (a break with history), with many unknowns but also the potential to substantially increase plant revenue and thus fundamentally improve nuclear plant economics. Added details are provided below. The requirements of the nuclear plant operator were addressed in the Electric Power Research Institute's (EPRI) Utility Requirements Document for Advanced Light Water Reactors (EPRI 1999), which was reviewed to understand the desirable functional requirements in terms of maintenance, operability, and economics from the perspective of utility operators. Section 5.1 presents the justification and trade-offs associated with various design choices with respect to economics and safety, Section 5.2 discusses utility requirements, and Section 5.3 lays out desirable features for an FHR commercial prototype.

5.1 Overview of Open-Air Brayton Power Conversion for FHRs

The potential to use a NACC is a key characteristic that differentiates FHRs from LWRs, HTGRs, and LMRs. This section reviews key issues for the development of NACC for FHRs (particularly achieving a small pinch-point temperature difference in salt-to-air heaters and tritium management).

The reasoning behind using a NACC coupled to an FHR for power conversion includes several technical and economic arguments. One of the most attractive reasons is that most components are largely commercially available and well developed (advances in gas power turbines and aero-derivative turbines) (Eldrid, Kaufman, and Marks 2001). This availability in turn allows rapid construction with modular components. Moreover, having multiple manufacturers (GE, Siemens, Alstom, and Mitsubishi) ensures fair market prices. One exception is the coiled-tube air heater (CTAH) salt-to-air heat exchanger, which is explained in a later section, and another is the modification of the turbine design to enable one or more stages of reheat.

Another beneficial aspect of the NACC is the feasibility of using a conventional combined-cycle configuration, with a Rankine bottoming cycle or other combined heat application. This combination implies efficiencies in the range of 44% to 48% and increased power conversion efficiency by 40% to 50% compared to LWRs. Furthermore, natural gas, hydrogen, or other fuels can be injected directly into gas turbines for co-firing, which yields increased temperatures and output for peaking operation.

In lieu of a steam bottoming cycle, the high-temperature exhaust of the gas turbine, ranging anywhere from 300°C to 500°C or more, can be used as process heat for distillation technology, hydrogen production, or other chemical processes. For example, low-temperature process heat can be used for thermal distillation of saline brines or seawater through advanced multi-effect distillation (Peterson and Zhao 2006).

In brief, a NACC coupled to an FHR becomes attractive to an electric utility because of low fuel costs, high efficiency, load-following ability, peaking power production, spinning reserve, and black-start services.

5.1.1 Basic Configuration

The planned basic configuration is a NACC with hot air exhausted to a Rankine steam bottoming cycle similar in design to an NGCC plant. The output of commercial gas turbine systems is limited. To build plants of different capacities while minimizing costs, utilities couple one to three gas turbines to a single steam bottoming cycle. Depending on the size of a commercial FHR, the same top-level layout of the power cycle would be used.

5.1.1.1 Brayton Cycle

The design of the Brayton cycle has many variations. To determine a baseline, the options of two, three, and four expansion stages were explored.

At present, the most attractive setup seems to be the two expansion stages with a single reheat stage because of its simplicity compared to using larger numbers of expansion stages, and to the limited pressure ratio of modern industrial gas turbines. This setup has only two CTAHs, as opposed to three or four. The arrangement reduces plumbing and CTAH surface area by providing a larger log-mean temperature difference as a result of the larger turbine stage expansion ratios (ER) and lower turbine exit temperatures, overall pressure losses, and circulating pumping power requirements. After a primary assessment, the best setup is for both expansion stages to have similar ERs as opposed to a high ER for the last stage. The higher ERs from only having two turbine stages coupled to a high compressor pressure ratio (PR) imply that flow at the turbine-inlet nozzle is choked for both stages. This setup also has a lower and more constant specific CTAH size per unit of power output, while keeping both CTAHs similar in size. Additionally, with the similar ER setup, the power output is distributed approximately in a two-to-one ratio between the air and steam cycles, as opposed to a more lopsided ratio for cases with more than two expansion stages.

The other prominent feature of the NACC for the FHR is that it is not recuperated for a number of reasons. First, the arrangement allows the steam cycle to be decoupled from the nuclear heat supply source. Closing the cycle would increase the complexity in implementing an air venting system that allows the combined cycle to accommodate loss-of-load transients without generating a rapid transient in the nuclear heat supply system. Second, recuperation would cannibalize heat from the steam bottoming cycle by reducing HRSG inlet temperature, having a negative impact on the overall cycle efficiency. Finally, co-firing would become considerably more complex in a closed cycle.

A supplemental consideration is the selection of a single or multi-shaft design for the Brayton cycle. With two expansion stages, the power balance requires that a single shaft design be used.

A few turbine designs are good candidates to be coupled to the FHR. Table 5-1 shows specifications for 60-Hz turbines from several manufacturers.

Table 5-1. Commercially Available Gas Turbine Specifications*

Manufacturer	Model	Power Output, MW	Mass Flow, kg/s	PR	Compressor Rows	Turbine Stages	Exhaust Temp, °C
Alstom	GT11N2	115.4	400	15.9	14	4	526
Alstom	GT24	230.7	505	35.4	22	4	597
GE	7FA	171.7	436	15.5	18	3	602
GE	7FB	184.4	436	18.5	18	3	
GE	7H	400	565	23	18	4	566
Siemens	SGT6-5000F	208	503	17.4	16	4	582
Siemens	SGT6-8000H	274	600	20	13	4	620
Mitsubishi	M501F	185.4	459	16	16	4	613
Mitsubishi	M501G	272	599	20	17	4	614

*All models have frequencies of 60 Hz.

5.1.1.2 Rankine Cycle

The Rankine cycle may be set up with reheat and multiple steam pressures, as is conventional in modern NGCC power plants. A reasonable baseline design would use a triple-pressure recovery steam generator and steam turbine assembly. The added combined cycle performance of a triple-pressure system is significant when weighed against the capital cost increase, which is moderate relative to total plant cost. In terms of design, an HRSG approach temperature of air/saturated steam of 15°C to 25°C pinch-point temperature difference is reasonable.

Overall, the combined cycle can be optimized based on the steam bottoming cycle. Many studies outline optimization techniques based on an energy analysis and minimization of the various components of the Rankine cycle (Franco and Casarosa 2002; Xiang and Chen 2007). This component can be optimized once the baseline design for the NACC is complete.

5.1.1.3 Co-firing

Co-firing is attractive for the FHR because the auto ignition temperature of natural gas is 580°C, well below design CTAH outlet temperatures of 660°C. Co-firing capability is placed after the final CTAH and before the final expansion stage inlet. Co-firing can increase the turbine inlet temperature to 900°C, resulting in increased power output of about 150% of base and efficiencies in the range of 48% to 50%. Inlet temperature into the HRSG ranges between 580°C and 640°C (depending of last stage ER), which is an optimal temperature according to Franco (2002). Increasing temperature of co-firing any higher would increase power output and efficiency further, which in return might require active blade cooling for the turbines as well as push inlet temperatures for the HRSG beyond 700°C, which is a material constraint. Exceeding this limit is possible by adding a radiant section at the front end of the HRSG, which can push temperatures to 900°C to 1,000°C, but adds a further complication to the design.

The operating cycles for peak power units vary from less than 100 hours per year to a few thousand hours per year. The lower incremental capital costs of an FHR peaking unit implies an economic advantage in the market for peaking units operated a few hundred hours per year where fuel costs are not important. For peaking units that operate more hours each year, the cost of fuel becomes important. The preferred peaking unit is then a tradeoff between capital costs and fuel (natural gas) costs. An FHR peaking unit is more efficient than a stand-alone open-air Brayton cycle. Initial estimates put the peaking efficiency of the FHR power conversion in the 52% to 60% range. The top end combined-cycle gas turbines have plant efficiencies of 56%. If the FHR is cost competitive for baseload electricity, the FHR peaking units could dominate this market.

However, a question that emerges is whether the last turbine stage and the HRSG can tolerate a rapid ramp in the air inlet temperature as a result of sudden co-firing. Moreover, additional requirements, tolerances, and operating limits must be laid out to better define the co-firing capability.

5.1.2 Assumptions and Limitations

For the NACC to be able to couple to the FHR, a set of design constraints and parameters need to be met. The most important one is the design and manufacture of the CTAH. The CTAH has two main functional requirements. Firstly, it needs to achieve an air outlet temperature of 660°C, which is a 20°C pinch from the 680°C inlet temperature of the flinak on the salt side. Secondly, flinak freezes at 454°C, which imposes another temperature constraint on the salt outlet temperature of 550°C to prevent freezing with a comfortable safety margin.

Additionally, for the NACC to be economical, the CTAH needs to have minimal pressure losses across the air side (reduces air circulating power requirements), high thermal effectiveness (measure of approach to true counter-flow), a relatively small size with a high surface area-to-volume ratio, the ability to drain the salt for maintenance and inspection (excludes certain types of heat exchangers), and a tritium diffusion barrier that can be placed on the air side (Al_2O_3 or other). All these functional requirements will require a high amount of custom design and manufacturing for the CTAH to effectively couple a secondary loop to the open-air Brayton cycle.

5.1.3 THERMOFLEX

The THERMOFLEX software has been instrumental for optimization and sensitivity studies involving the combined cycle power conversion setup; however, several points can be made about the software’s limitations and shortcomings. First, the models use efficiency data for specific turbomachines that may not be correct. Second, piping losses and losses at elbows, joints, valves, and other plumbing are not included. Moreover, the CTAHs are simulated by a thermodynamic one-dimensional model without any detailed thermal hydraulics or heat exchanger layout. Instead, nominal pressure drop and heat loss are added across the CTAHs, an approach that might not be completely realistic. Finally, there is no dynamic simulation and therefore all results are steady state.

To model the baseline FHR cycle, ISO 3977 ambient conditions were used (ambient conditions of 15°C, 1 atm, 60% RH), which is the standard environmental design point of any gas turbine system (Brooks). This standard is divided into nine subsections and covers procurement, design requirements, installation, and reliability. By using this standard, however, performance and efficiency take a significant penalty at higher ambient temperatures. On the other hand, in very cold climates performance is better than the International Standards Organization standard.

The NACC was modeled as a compressor with no intercooling and a single reheat stage (two expansion stages). The exhaust was modeled to go into a triple-pressure HRSG before exhausting to the atmosphere. The Rankine cycle was modeled as a closed loop system dumping heat to atmosphere through a dry cooling tower. The co-firing was modeled as a flow of natural gas into a combustor, which raised the temperature to 900°C before the last expansion stage. The analysis assumed compressor efficiencies of 85% to 89%, turbine efficiencies of 87% to 91%, and a CTAH nominal pressure drop of 2% of the total system pressure drop. Schematics of parts of the THERMOFLEX model are shown in Figure 5-1 and Figure 5-2. Indicative results for the cycle can be found in Appendix D.

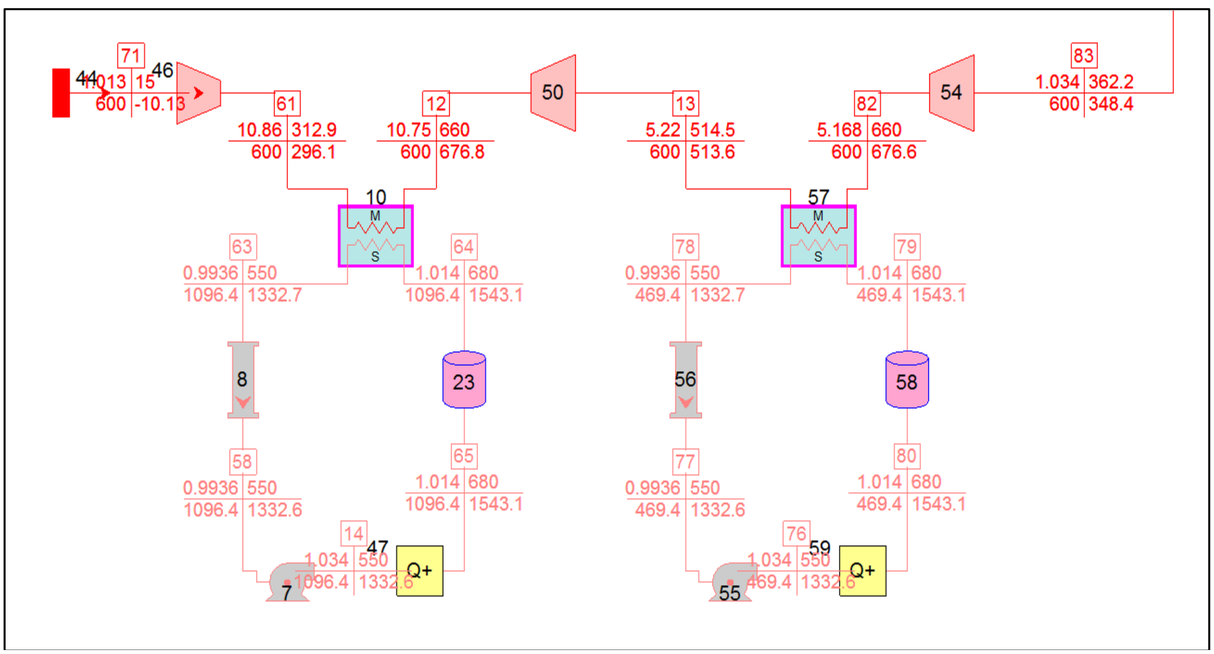

Figure 5-1. Open-Air Brayton Cycle and Intermediate Loop Modeling Configuration

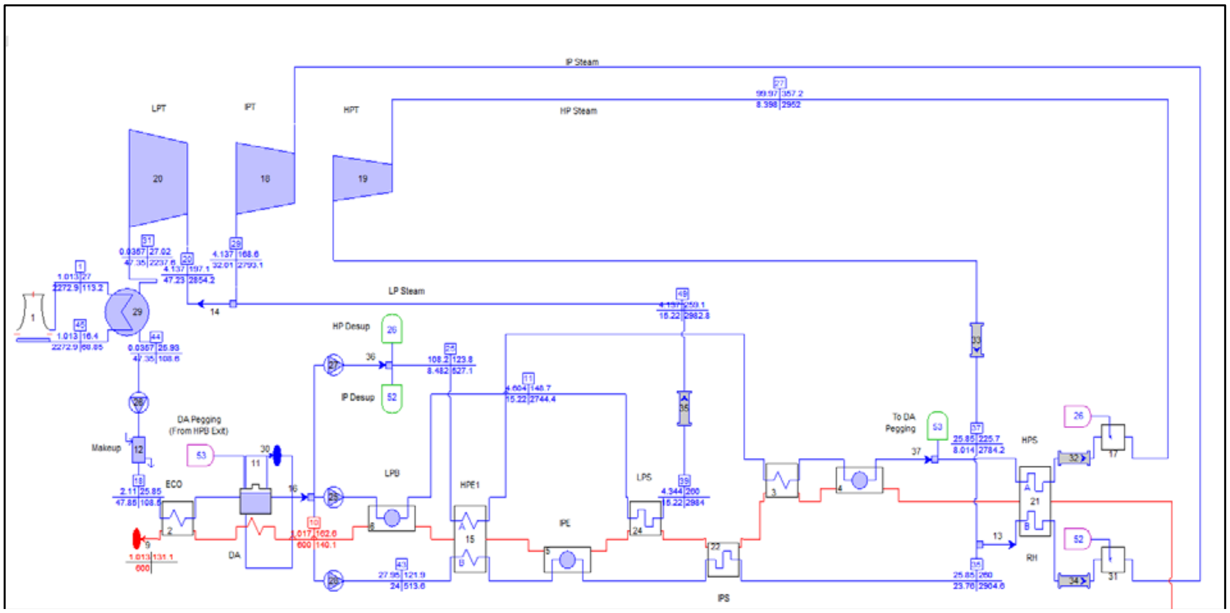

Figure 5-2. Rankine Cycle Modeling Configuration

5.2 Utility Requirements for FHRs

EPRI's ALWR Policy and Summary of Top-Tier Requirements (EPRI 1999) lists what the U.S. electric utilities desire for next-generation nuclear power plants – including core, as well as balance of plant – based on over 40 years of domestic and international operation of LWRs. These functional requirements were developed based on a set of 14 ALWR policies:

- Simplification
- Design margin
- Human factors
- Safety
- Design basis versus safety margin
- Regulatory stabilization
- Standardization
- Proven technology
- Maintainability
- Constructability
- Quality assurance

- Economics
- Sabotage prevention
- Good neighbor.

Note that there is a key difference between the commercialization approaches of FHRs compared to the commercialization approach of an ALWR. Because the FHR is a new reactor technology, many components and systems without proven technology solutions will be tested. However, whenever appropriate the FHR commercial prototype can take advantage of existing technology.

Another key point is the design basis versus safety margin approach proposed by EPRI. This approach utilizes a deterministic framework for licensing safety analysis (design basis) and a best estimate probabilistic framework taking advantage of non-safety-related systems for investment protection (safety margin).

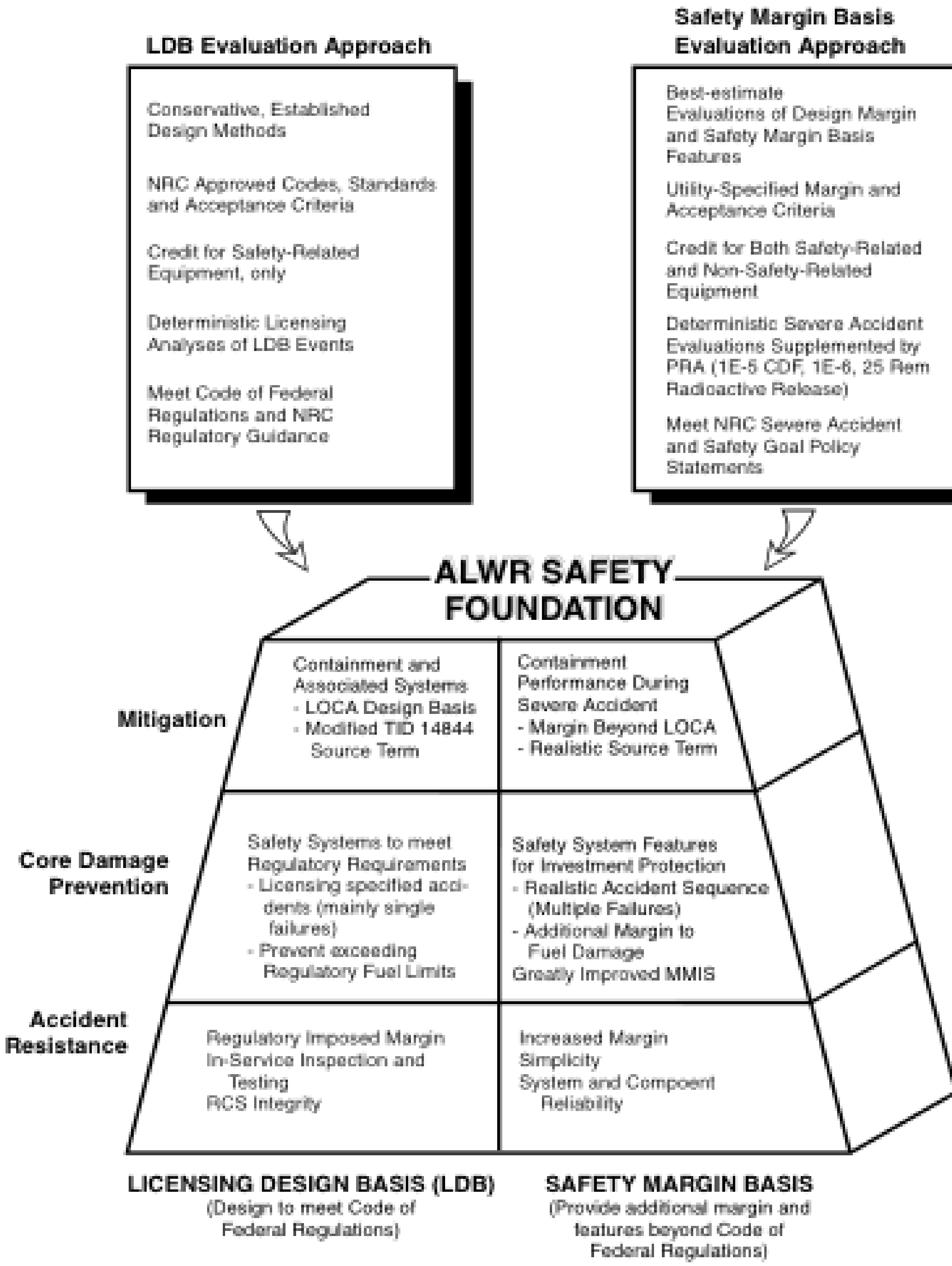

Figure 5-3. ALWR Safety Foundation (EPRI 1999)

The requirements presented in the ALWR Utility Requirements Document (EPRI 1999) were developed for evolutionary LWRs and passive LWRs. Given that one of the defining features of the FHR class of reactors is the DRACS loop for passive decay heat removal, the passive LWR requirements will be applied to FHRs. These requirements were divided into five high-level functions: safety, power plant availability, radioactive waste, maneuvering, and construction time. The most important top-tier functional requirements are summarized in Table 5-2. A full mapping of the ALWR utility requirement to FHR design are presented in Appendix E of this white paper.

Table 5-2. Highlights of EPRI's ALWR Top-Tier Requirements (EPRI 1999)

Area	Requirements
Safety	Fuel design margin of 15% above regulatory requirements*
	Safe shutdown earthquake 0.3 *g*
	Power reactivity coefficient negative under all conditions
	72 hours to respond to an accident
Power plant availability	60 year life of plant
	24-month or longer shutdown frequency
	17-day or less nuclear outage
	87% or greater availability over life of plant
Radioactive waste	Wet storage for 10 years of operation
	Dry storage for all the fuel consumed over life of plant
	Radioactive waste produced consistent with the top 10% of efficient LWRs**
Maneuvering	Startup from cold zero power to hot full power within 24 hours
	Rampdown from 100% to 50% and rampup of 50% to 100% power within 2 hours***
Construction time	36- to 60-month construction time including 6 months for inspections, tests, and analyses, ensuring acceptance criteria are met****

* For high-temperature coated-particle fuel, it is unclear what EPRI is referring to as "fuel design margin."
** FHRs produce tritium during normal operation so it is important to understand targets in terms of tritium release.
*** This maneuvering requirement should consider electric power produced by natural gas heat source in addition to the nuclear heat source taking the thermal efficiency.
**** This construction time should be reduced to reflect the lower power rating for the commercial prototype FHR.

5.3 Desirable Features for an FHR Commercial Prototype

This section discusses the basis for selecting an initial FHR commercial prototype power level and other key design parameters/features. The very first commercial prototype FHR cannot practicably be a multi-gigawatt reactor, even though large FHR reactors may be the most desirable commercial technology in the long term. This section is divided into three components: an simplified economics cash flow model, the temperature limits in FHRs, and the physical and economic tradeoffs when selecting the size and power density of an FHR with respect to transportability and power conversion systems. The final subsection discusses what costs do not scale with size of the commercial prototype FHR.

5.3.1 Simplified Economics Model

This subsection presents a conceptual model of the economics of an FHR commercial prototype. Developing an economics model is an important component of the development of a conceptual design of a nuclear power plant because it gives the plant designer a means to perform economic trade studies on different design options.

The economics model was developed based on the cash flow of a proposed FHR project and used to determine the required capital investment and rate of return on investment over the life of the FHR project. This economics model will be used in the subsequent sections as a basis to qualitatively assess the economic impacts of various design choices, by determining to what degree specific design decisions affect each of the 106 components of this simplified economics model:

- *Construction costs:* The economics of an FHR system were analyzed for advanced high-temperature reactor systems (Holcomb, Peretz, and Qualls 2011) using the international Economic Modeling Working Group Generation IV Excel Calculation of Nuclear Systems (G4-ECONS) model (Economic Modeling Working Group 2008) and Energy Economics Data Base (Delene and Bowers 1986); a similar study was performed by the NGNP program for an HTGR (Gandrik 2012). These capital costs were divided into direct capital costs, indirect capital costs and other capital costs. Direct capital costs include structures and improvements, reactor plant equipment, turbine-generator equipment, electrical equipment, heat rejection equipment, miscellaneous equipment, special materials, and simulator. Indirect capital costs include design, quality assurance, project management, temporary facilities, insurance, taxes, permits, and other costs associated with the construction site. Finally, other capital costs include land, licensing, studies and reports, staff recruitment, training, shipping and transport of components and spare parts, and possibly the startup core's fuel inventory.
- *Nuclear fuel cycle costs:* The nuclear fuel cycle costs for FHRs include uranium mining, uranium conversion, uranium enrichment, tails disposal, fuel fabrication, spent fuel storage, and disposition (Economic Modeling Working Group 2008; Delene and Bowers 1986; Gandrik 2012).
- *Natural gas costs:* Some PB-FHR power plants utilize a natural gas firing stage before the final turbine stage to provide additional peaking power depending on electricity demand and spot prices of electric power. For these PB-FHR power plants, the cost of natural gas becomes another variable cost.
- *Non-fuel operations and maintenance:* The AHTR Systems and Economic Analysis (Holcomb, Peretz, and Qualls 2011) lists (non-fuel) operations and maintenance costs as staffing, pension and benefits, consumables, repair, services and subcontracts, insurance,

regulator fees, radioactive waste management, capital replacement, and other general and administrative costs.

- *Decommissioning costs:* Decommissioning and decontamination costs cover costs after the power plant can no longer be used.
- *Revenue from electricity sales:* The FHR will generate revenue via the sales of electricity. Future FHRs may produce process heat for industrial applications such as oil production, hydrogen production, desalination, etc. (Shropshire et al. 2007; Forsberg 2008b; Greene, Flanagan, and Borole 2009; Konefal and Rackiewicz 2008).
- *Construction time:* The time it takes to construct the power plant; conduct inspections, tests, and analyses; and ensure that acceptance criteria are met plays a significant role in the economics of the FHR project because interest accrues over this period of time without any revenue.
- *Shutdown period:* This is the length of time nuclear outages last. The FHR will periodically need to shut down to perform maintenance and change out replaceable components. Note that the EPRI ALWR Policy and Top-Tier Requirements document proposed a minimum refueling period (i.e., shutdown period) of 24 months (EPRI 1999).
- *Nuclear outage time:* This is the period when the nuclear heat generator is shut down. During this period, no heat is being produced from the nuclear core, so the natural gas heating section of the power conversion system will operate differently if it operates at all. The length of the nuclear outage time and shutdown period combine to determine the capacity factor of the power plant. Note that the EPRI ALWR Policy and Top-Tier Requirements document proposed a minimum nuclear outage time of 17 days (EPRI 1999).
- *Life of plant:* This is the length of time the project can operate. The life of plant is determined by how long the large capital investments (reactor vessel, graphite outer reflector, heat exchangers, etc.) can be maintained safe and serviceable. Note that the EPRI ALWR Policy and Top-Tier Requirements document proposed a minimum life of plant of 60 years (EPRI 1999).

One important source of uncertainty for any FHR capital model is the costs of fluoride-salt production. The flibe is likely to be a significant capital expenditure because of uncertainty of the costs of lithium depleted in ^{6}Li and safety concerns associated with beryllium handling. In this model, flibe is considered part of the construction costs.

Fuel fabrications have a major impact on the fuel-cycle costs in FHRs, driving the baseline fuel design to as high enrichment as possible to reach as high burnup as possible, reducing the fuel fabrication per unit energy produced. An open topic of discussion is how the power plant designer can leverage the enhanced safety features of FHRs to reduce quality assurance standards (one of the biggest drivers in coated-particle fuel fabrication costs) and reduce the overall fuel cycle costs.

The electricity market to which the FHR would supply power will determine how much revenue the utility will receive. Several Regional Transmission Organizations and Independent System Operators are responsible for managing electricity markets in North America and will serve as markets for electricity generated in an FHR.[1] Each of these markets has its own set of market rules, but can be generally classified as energy only markets – where electricity generators only earn revenue from electricity sales – and markets where utilities can sell ancillary services such as reserve capacity. In the California Independent System Operator group, spinning and non-spinning reserve capacity is the portion of unloaded capacity that can deliver energy to the grid within 10 minutes .[2] In general, the price of peak power is more volatile in energy-only markets – this price volatility makes the peaking power attractive even without payments for reserve capacity services.

The structure of specific electricity markets will determine how advantageous a natural gas co-firing is to the economics of FHRs. The following cases provide additional detail.

5.3.1.1 Case A: Nuclear Power Only

In this economics case, the nuclear power plant only generates heat. All the capital costs (*construction costs*) are discounted to an initial time. After an initial *construction time*, electricity can be generated and revenue (*revenue from electricity sales*) is collected. While electricity is being generated, *operating and maintenance costs* and *nuclear fuel cycle costs* are also accrued. Periodically (*shutdown period*), the nuclear reactor must be shut down for maintenance and during this *nuclear outage* only maintenance costs (*operating and maintenance costs*) are accrued, though not necessarily at a rate consistent with the operation and maintenance costs during normal power production. Finally, after the life of plant, the nuclear reactor must be decommissioned (*decommissioning costs*). Figure 5-4 shows a stylized graphic of the cost flow for Case A; the detailed cost analyses have not been performed so the relative costs (y-axis dimension) are not intended to reflect actual values.

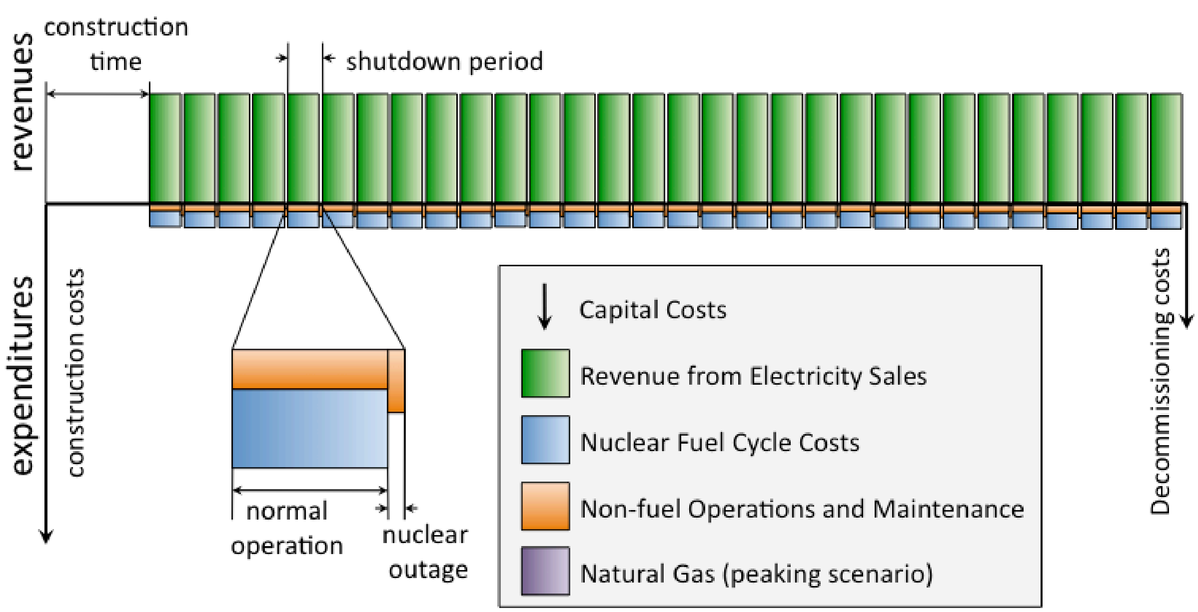

Figure 5-4. Cost Flow of Case A: Nuclear Only Commercial FHR

[1] Information from the ISO/RTO Council website at http://www.isorto.org. Accessed May 24, 2013.
[2] Information from the California Independent System Operator's "Market Products and Services Help Meet Demand" website at http://www.caiso.com/market/Pages/ProductsServices/Default.aspx. Accessed May 24, 2013.

Based on assumptions of the expected costs and revenues for the commercial FHR power plant project, the capital investment required for the PB-FHR and the rate of return on this investment can be estimated.

5.3.1.2 Case B: Nuclear Power with Simple Gas Turbine Peaking

In Case B, a natural gas heater is used to provide additional heat to produce more power when electricity demand is highest and thereby sells for a higher cost. Load following was one of the desirable features presented in EPRI's Advanced Light Water Reactor Utility Requirements Document (EPRI 1999). Preliminary analysis shows a co-firing commercial FHR can produce an additional 180 MWe with a thermal efficiency of approximately 58% to 62% for power produced from the natural gas heater. The cost flow for Case B is essentially the same as Case A, but the construction costs include additional costs to accommodate this peaking power production (natural gas heater, infrastructure to handle natural gas, higher rated electric transmission equipment, etc.), additional revenue from electricity sales to reflect increased energy production at peak power, additional operation and maintenance (operating and maintaining the natural gas heater, changes in costs relating to higher-capacity power conversion system, etc.), and costs of natural gas.

Figure 5-5 shows a stylized graphic of the cost flow for Case B; the detailed cost analyses have not been performed so the relative costs (y-axis dimension) are not intended to reflect actual values.

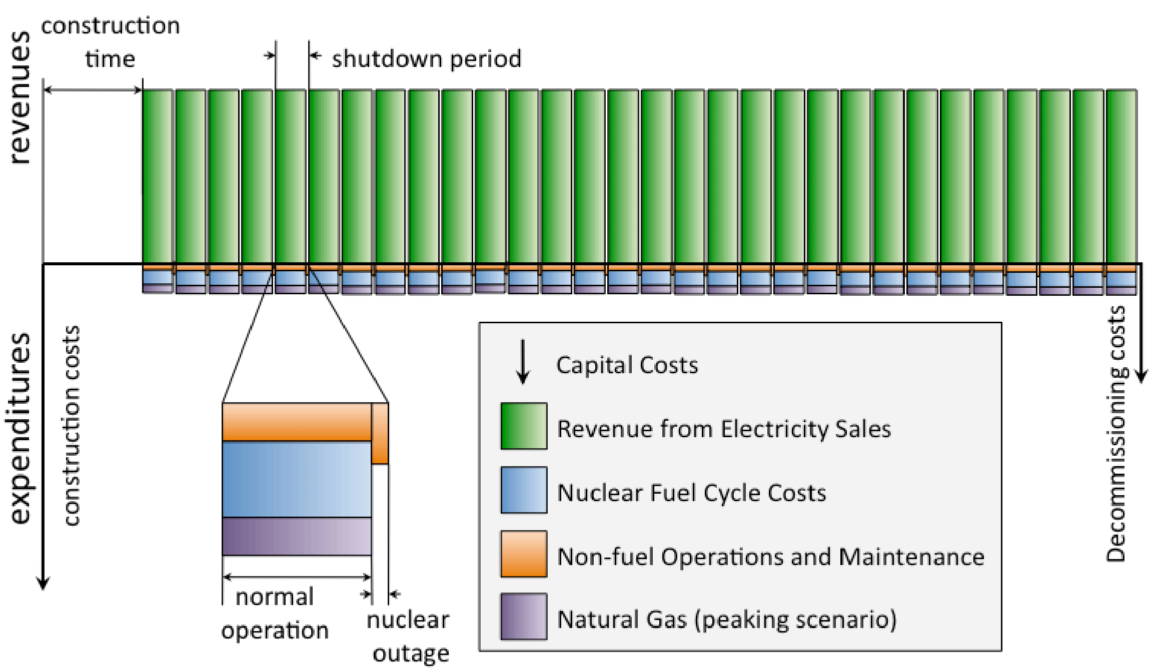

Figure 5-5. Cost Flow of Case B: Nuclear Power with Simple Gas Turbine Peaking

5.3.2 Temperatures

This subsection presents the physical constraints and tradeoffs that limit the operating temperatures of the commercial FHR. Five main concerns limit the coolant temperatures:

materials qualification limits, high-temperature creep, accident response, thermal hydraulics, flibe's freezing point, corrosion, thermal efficiency and process heat applications.

The three candidate materials for metallic innards in an FHR – Alloy N, 316 SS, and Alloy 800H – have normal operation temperature limits of 704ºC, 750ºC, and 800ºC, respectively; note that Alloy N is reasonably well proven at temperatures up to 704ºC (Koger 1972); however, it has not been codified into ASME Division 5 (high-temperature materials) Section III Boiler and Pressure Vessel Code. Therefore, FHRs – AHTR, PB-AHTR, and small modular AHTR (SmAHTR) – have traditionally been designed with a maximum coolant bulk temperature of 700ºC.

Another consideration on the operating temperatures is thermal creep. The FHRs metallic innards will be exposed to temperatures where the time-dependent structural properties of materials are important. Division 5 of Section III presents the means to analyze these time-dependent effects.

Moreover, the coolant temperature sets the initial conditions (coolant temperature and fuel kernel temperature) of an accident sequence. The metallic structural materials must be able to behave gracefully during accidents – the most severe being an unprotected loss of cooling accident. Under such a transient, the temperature fuel and coolant equilibrate to a characteristic temperature between the normal operating condition of the coolant and normal operating temperature of the fuel kernels (Cisneros et al. 2012). Therefore, the structural materials must survive these temperatures for a brief period of time without catastrophic failure; however, precise limits for these excursions have yet to be defined.

In addition to the outlet temperature, the temperature rise across the core has a significant effect on the thermal hydraulics of the FHR core. For a core with a constant power, the temperature rise and coolant mass flow rate are linearly dependent (flow regimes). However, increasing the coolant flow rate increases the pumping power requirement, thereby penalizing the power plant's thermal efficiency. The temperature rise is limited by the freezing point of flibe – 460ºC. The pebble bed fuel geometry continuously disrupts the coolant's flow, enabling smooth transitioning from laminar to transition to turbulent flow regimes without instabilities. Cores with channel flow paths or rather cores with prismatic- or plate-type fuel are susceptible to instabilities transitioning from laminar to turbulent flow in their heat removal systems. To protect against over-cooling transients, a significant margin between the lowest operating temperature and this freezing point is desirable.

Increased temperature may also have an effect on the corrosion rate of the components in the various high-temperature salt loops (Williams, Toth, and Clarno 2006). More work must be done to understand the corrosion degradation mechanisms and develop quantitative and measurable corrosion limits.

The FHR has the ability to deliver heat at high temperatures to the power conversion system. Delivering heat at different temperatures expands the potential to utilize multiple power conversion technologies including steam cycles – the baseline for the AHTR (Varma et al. 2012) – and supercritical CO_2 cycles – the baseline for the SmAHTR (Greene et al. 2010). Figure 5-6 shows the thermal cycle efficiencies over operable temperature ranges for various power conversion technologies. Preliminary thermal cycle analysis has been performed assuming heat can be delivered to the turbomachinery at ~650ºC.

An open-air Brayton cycle has been selected as the baseline power conversion technology for the FHR because of the existing technical base, availability (turbomachinery for a Brayton cycle can be procured as an “off-the-shelf” item, rather than a “one-off” item), and potential for added revenue from peak power and grid stability operations. Additionally, the NACC would provide the ability to exhaust heat even directly to the atmosphere in the event of a malfunction in the power conversion system. Furthermore, the open-air working fluid of this cycle reduces the feedwater requirements for this power plant relative to one with a steam cycle, thereby opening more potential sites.

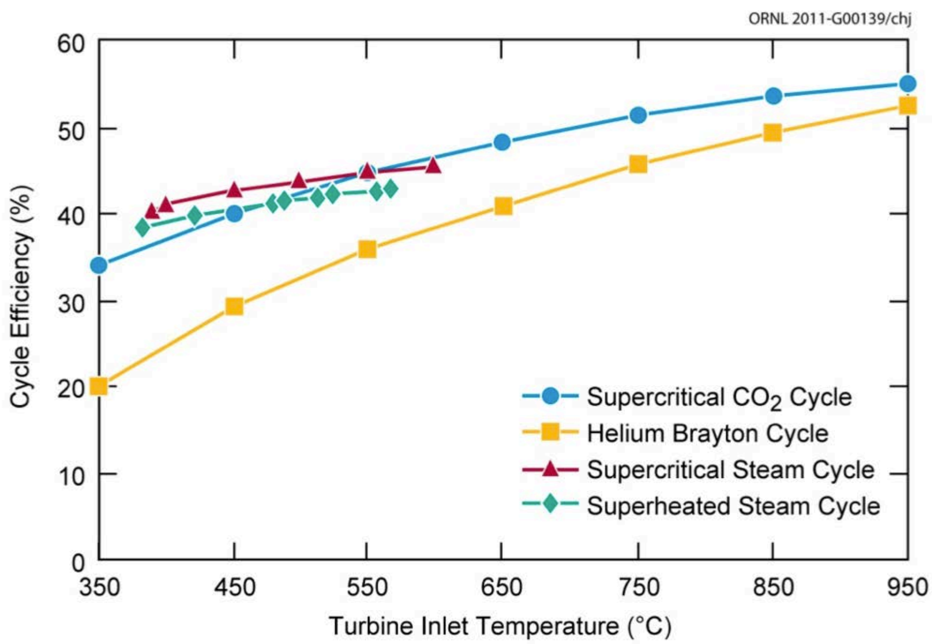

Figure 5-6. Comparison of Thermal Cycle Efficiencies for Various Power Conversion Technologies (Dostal 2000)

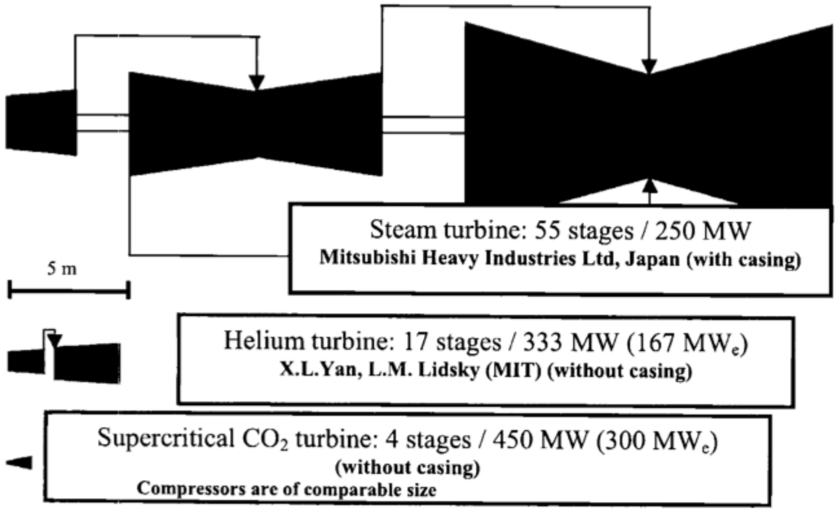

Figure 5-7. Comparison of Physical Volume of Various Power Conversion Technologies (Dostal 2000)

Supercritical CO_2 thermal cycles are an intriguing long-term technology option for FHRs (Cisneros and Qualls 2010). However, these cycles were not considered because of the near-term timeline for implementation of FHR technology and the strategy to minimize risk by selecting an established power conversion technology.

In future applications of FHR technology, the temperature at which the heat can be delivered will also influence the applications for which FHRs can produce process heat. Figure 5-8 presents the requisite temperature and potential size of an energy market for some proposed process heat applications of nuclear technology.

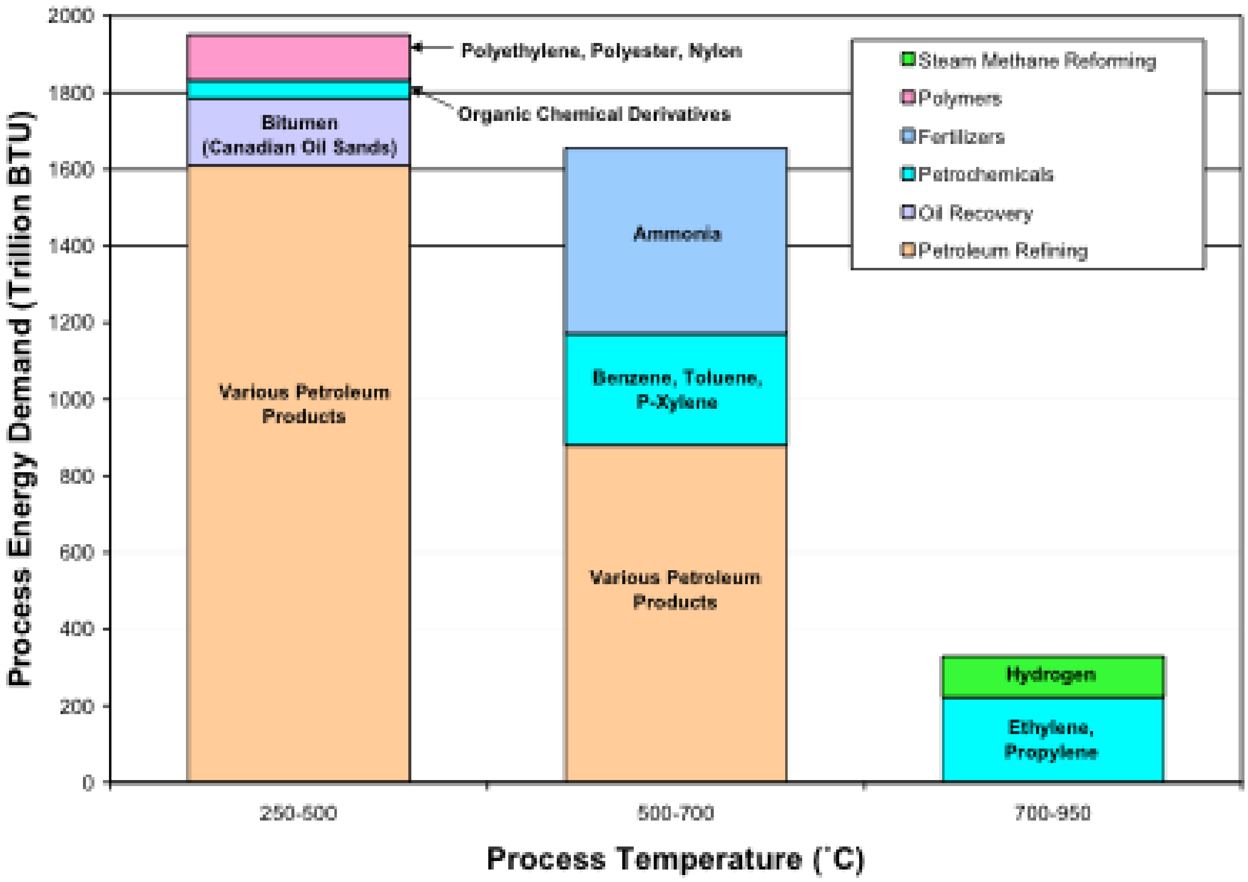

Figure 5-8. Temperature and Power Requirements of Proposed Process Heat Applications of Nuclear Power (Konefal and Rackiewicz 2008)

Based on the physical limitations and implication of the operating temperature (maximum outlet temperature and temperature rise across the core) of the FHR, the sensitivity of each component of the simplified case flow model was qualitatively assessed. The results of this assessment are presented in Table 5-3.

Table 5-3. Impact of Coolant Temperature on the Economics of the FHR Power Plant Project

Cash Flow Model Component	Impact
Construction costs	High*
Nuclear fuel cycle costs	Moderate**
Natural gas costs	Low
Non-fuel operations and maintenance	Low***
Decommissioning costs	Low
Revenue from electricity sales	Moderate**
Construction time	Low
Shutdown period	Low***
Nuclear outage time	Low***
Life of plant	High****

* Materials qualification presents a feasibility issue on which structural materials can be used for the metallic innards, heat exchangers, and reactor vessel. Depending on which material is selected and at which temperature the core is operated, the thickness of components must be determined such that the requisite strength can be guaranteed throughout the life of plant. This thickness translates into the volume of material needed for structural components, which should be related to the costs of these components.

** The outlet temperature and temperature rise in the core affect the thermal efficiency of the power conversion system. This change in efficiency effects amount of electricity produced and that directly affects the revenue collected from electricity generation. Generating more thermal power can offset a reduction in thermal efficiency; however, this exchange will redistribute the costs of the loss of efficiency from the revenue collected from electricity to the nuclear fuel cycle costs.

*** The temperature at the structural component/coolant interface under normal operation conditions determines the rate at which structural components corrode in the primary, DRACS and intermediate loop. Many components can be replaced periodically during the nuclear outage. However, replacing degraded components could increase the amount of maintenance required during operation and possibly during the nuclear outage. The cost of replacement of components is part of the operation and maintenance costs. Furthermore, degradation of structural materials could necessitate a reduced shutdown period.

**** Degradation from chemical attack or thermal creep can affect components intended to last the life of the plant – heat exchangers, fixed outer solid graphite reflector or the reactor vessel – and could limit the operating life of the plant.

5.3.3 Reactor Power Limit

This subsection presents the physical constraints and tradeoffs that limit the total power, power density, and size of the commercial prototype FHR. Six main concerns limit the size and power of this reactor: transportability, turbomachinery, heat exchanger design, radiation damage, neutron leakage, and safety.

5.3.3.1 Transportability

The plans for the commercial prototype FHR are targeting an integral vessel design that is rail transportable. This approach limits the diameter of the core to 3.5 m and the height of the core to 23 m. The diameter limitation leads to a cigar-shaped core that extends axially to increase active volume in the core or volume of heat exchangers, as shown in Figure 5-9.

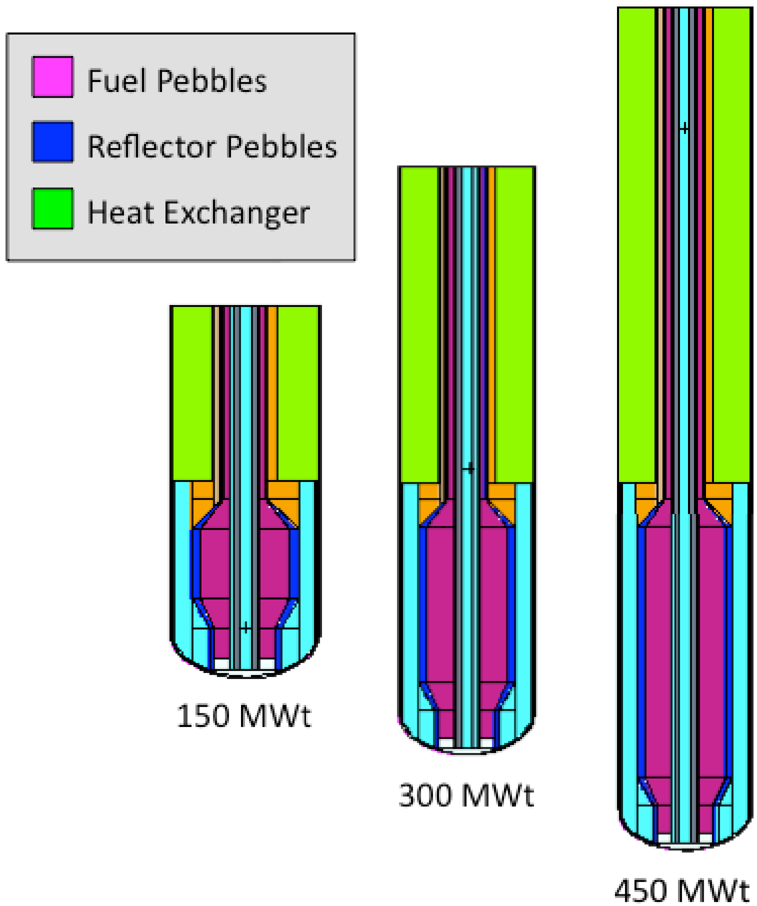

Figure 5-9. FHR Core Geometry Assuming 20-MW/m^3 Core Power Density and 10-MW/m^3 Heat Exchanger Power Density at Different Power Ratings

5.3.3.2 Turbomachinery

The strategy for the design of the power conversion system is to utilize “off-the-shelf” turbomachinery. However, this strategy will limit the total thermal power produced by the FHR nuclear heat source to discreet points corresponding to these “off-the-shelf” components. The estimated limit of maximum combined thermal power from one of these gas turbines with a steam bottoming cycle system is approximately 650 MWth, with 350 MWth and 300 MWth of heat produced from fission energy and natural gas, respectively.

5.3.3.3 Heat Exchanger Design

The baseline, compact design of the FHR requires high power densities in the intermediate heat exchanger and the DRACS. The power densities in the intermediate heat exchanger and DRACS heat exchanger (shell and tube heat exchangers) in the SmAHTR can be back-calculated to be 20 MW/m^3 and 14 MW/m^3 (normal operating power/volume), respectively. The power densities of these heat exchangers are strong drivers of the axial height of the FHR core because these heat exchangers must be housed inside the reactor vessel above the converging region of the core; Figure 5-10 shows how the geometry of the FHR core changes with different power densities of the intermediate heat exchanger and DRACS heat exchanger. However, it is expected that friction losses in these heat exchangers will be unacceptable without the implementation of twisted tubes or enhanced surface features. As core size and power drop, the coolant flow velocity (i.e., superficial velocity and Reynolds number) will drop, challenging the heat removal capacity of the intermediate heat exchanger; Figure 5-11 shows a small section of a twisted tube.

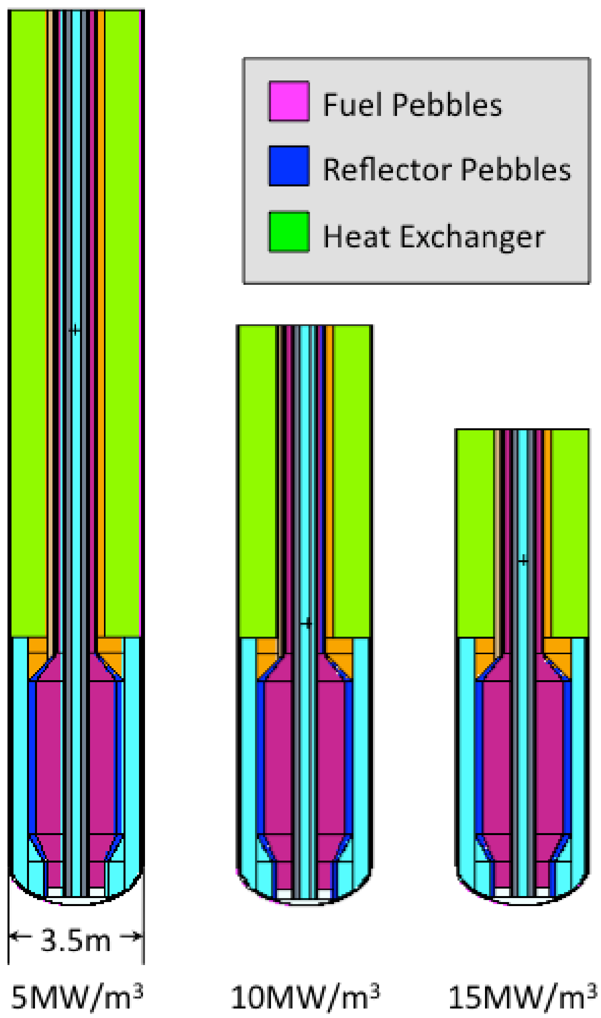

Figure 5-10. 300-MWth FHR Core Geometry with Different Power Densities of the Intermediate Heat Exchanger and DRACS Heat Exchanger, Assuming a Power Density of 20 MW/m^3

Figure 5-11. Twisted Tube

5.3.3.4 Radiation Damage

Radiation damage limits the power density of the FHR and thereby, its total thermal power rating (see the third FHR workshop whitepaper for more information). Radiation damage causes degradation of the thermal and mechanical properties in the structural materials. The reactor vessel, intermediate heat exchanger, and outer solid graphite reflector should last the life of plant before reaching a radiation damage limit. However, the inner solid graphite reflector and pebble separator can be changed out during a nuclear outage. Table 5-4 presents the radiation damage limits being utilized for conceptual reactor design; detailed reactor design will model how these property changes effect the ability of the reactor components to perform their functional requirements.

Table 5-4. Preliminary Radiation Damage Limits for Different Structural Components in the FHR

Structural Component	Radiation Damage Limit	Service Lifetime
Inner graphite reflector	15 DPA	One shutdown period
Outer graphite reflector	15 DPA	Life of plant
Physical pebble separator	10 DPA	One shutdown period
Reactor vessel	$1.*10^{20}$ n/cm^2	Life of plant
Intermediate heat exchanger	$1.*10^{20}$ n/cm^2	Life of plant
DRACS heat exchanger	$1*10^{20}$ n/cm^2	Life of plant

5.3.3.5 Neutron Leakage

The neutron leakage (or non-leakage probability) will change with the physical geometry of the FHR core. Equation 5-1 shows the form for non-leakage probability, and Eq. 5-2 shows the form for the Buckling factor for a finite cylinder, which should qualitatively show the same trends as the FHR core's actual geometry. In these equations: *P* is the non-leakage probability; *B* is the geometric buckling factor; *L* is the neutron length; *R* is the radius of a cylindrical core; and *H* is the active height of a cylindrical core.

$$P = \frac{1}{1 + B^2 L^2} \qquad (5\text{-}1)$$

$$\boldsymbol{B^2} = \left(\frac{\mathbf{2.405}}{\boldsymbol{R}}\right)^{\mathbf{2}} + \left(\frac{\boldsymbol{\pi}}{\boldsymbol{H}}\right)^{\mathbf{2}} \qquad (5\text{-}2)$$

For smaller FHR cores, an increase in the non-leakage probability will penalize the neutron economy, ultimately resulting in a lower discharge burnup. To get a sense of how this leakage was calculated for reactors with different power ratings, see Figure 5-12.

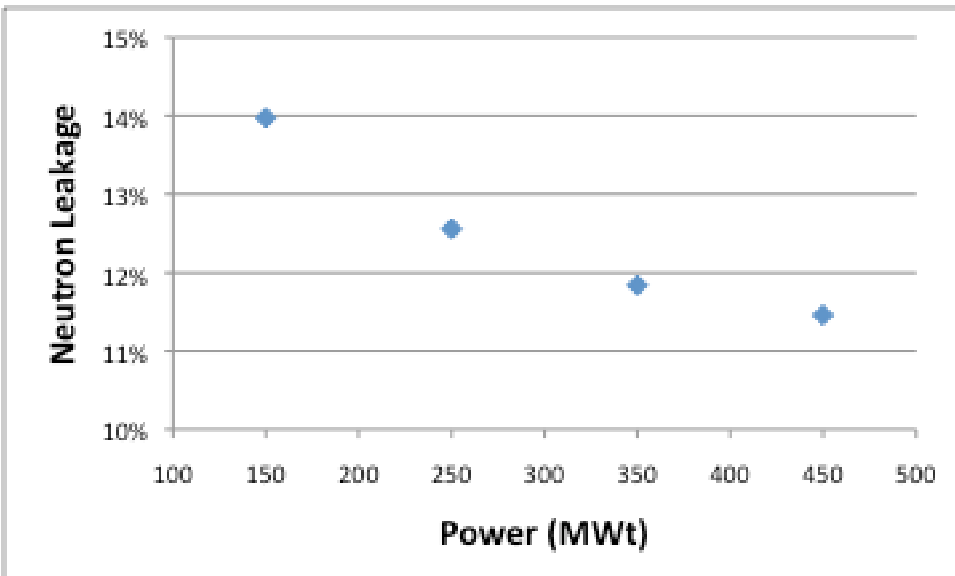

Figure 5-12. Neutron Leakage (assuming beginning-of-cycle fuel) as a Function of Power of the FHR with a Constant Power Density of 20 MW/m^3

The relationship between core size and leakage also influences the coolant's temperature reactivity feedback. As coolant voids, the neutron leakage out of the core should increase. However, this leakage could also result in more thermalized neutrons returning to the core (with a net lower non-leakage probability) partially offsetting the loss of moderation from the coolant as a result of voiding. Figure 5-13 shows the effect of reduced leakage on the coolant reactivity coefficient.

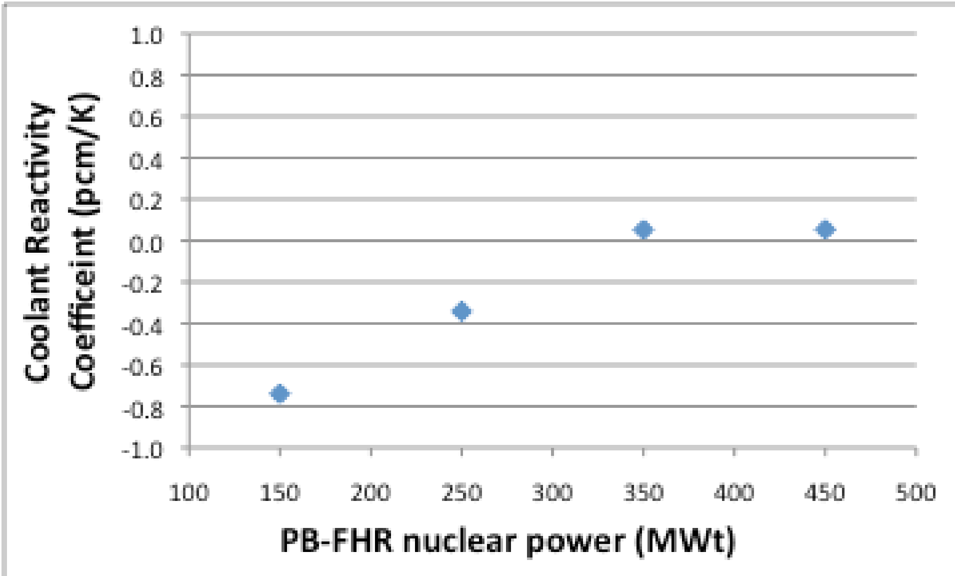

Figure 5-13. Coolant Reactivity Coefficient (assuming beginning-of-cycle fuel and ^{6}Li at 10 molar pcm) as a Function of Power of the FHR with a Constant Power Density of 20 MW/m^{3}

5.3.3.6 Safety

Safety and investment protection must be guaranteed to begin construction of the FHR power plant. The first FHR IRP workshop developed a set of characteristic design basis accidents that would challenge the safety of the FHR. When the primary cooling mechanism is lost, the establishment of natural circulation and effective heat removal in the DRACS loop are the criteria for safety, so long as neutron multiplication is shut down by engaging reactivity control elements. The DRACS heat exchanger must be properly sized in terms of surface area and elevation to remove the decay heat. However, if the primary cooling mechanism is lost and control elements are not engaged (i.e., ATWS), neutron multiplication will be halted by the core's inherent reactivity feedback mechanisms. The temperature of the coolant will rise to an equilibrium state where only decay power is produced in the fuel, and the positive reactivity inserted by reducing the fuel temperatures is balanced by negative reactivity inserted by increasing the temperature of the coolant. The elevated temperature of the coolant challenges the normal operating temperature limits for the reactor vessel, intermediate heat exchanger, and DRACS. It can be shown that the maximum temperature the coolant can reach in this class of severe accidents is limited by the characteristic normal operating temperature of the fuel in the FHR, which scales with the power density of the core.

Table 5-5 provides the results of an assessment of the impacts of reactor power on the simplified case flow model.

Table 5-5. Impact of Reactor Power on the Economics of the FHR Power Plant Project

Cash Flow Model Component	Impact
Construction costs	High *
Nuclear fuel cycle costs	High *
Natural gas costs	Medium **
Non-fuel operations and maintenance	Medium ***
Decommissioning costs	High *
Revenue from electricity sales	High *
Construction time	Medium ****
Shutdown period	Medium †
Nuclear outage time	Medium ***
Life of plant	Low

* Revenue should be directly proportional to the power rating of the FHR. The burnup should not be very sensitive to the power level of the reactor; therefore, the fuel cycle costs should also scale proportionally with power level. The construction and decommissioning costs will increase with power rating of the FHR, but the costs of each system, service, and/or component will change by different scaling factors,

** Think of the FHR as a pre-conditioner for a natural gas heater. Increased power might translate into increased capacity to produce power from natural gas. Therefore, increased natural gas consumption is needed in systems with peaking power produced from a natural gas heater.

*** As the size of the FHR increases, the number and/or size of components should increase the operations and maintenance costs and possibly lead to longer nuclear outages.

**** The construction time should scale mostly with the height of the reactor and the reactor building, which is not directly proportional to the power level of the FHR.

† The shutdown period is limited by a number of constraints, one of them being the radiation damage to structural components, which in turn should scale with the power density of the FHR core (to the inner graphite reflector) and the neutron shielding (outer reflector, heat exchangers, and pebble separator).

5.3.4 Systems Insensitive to Size

Note that the costs of many systems will remain constant independent of other design decisions. A preliminary list of these system components are as follows:

- Reactor licensing analysis
- Fuel handling system

- Monitoring
- Instrumentation and control
- Physical protection and safeguards
- Operations staffing
- Emergency planning
- Electricity transmission infrastructure
- Rail and/or shipping infrastructure.

Scaling of many of these systems is also a topic of concern for some PWRs. The DOE is working with the NRC to potentially resolve a number of policy and licensing issues related to such PWRs (Borchardt 2010).

The revenue from electricity sales required to cover these fixed costs in addition to the variable (with respect to total power) costs will set a minimum power for the commercial prototype FHR.

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Appendix A Fluoride-Salt-Cooled High-Temperature Reactor: Fuel Options and Implications for Test Reactors

The FHR is a new reactor concept based on combining a high-temperature fuel with a high-temperature low-pressure fluoride salt. There are several candidate fuels. The near-term option is the graphite-matrix coated-particle fuel because of its (1) demonstrated high-temperature capability and (2) compatibility with high-temperature salts. It can be fabricated into many different forms. In the longer term, there are other candidates that use silicon carbide (SiC) as a major structural material. Last, there are long-term metal-clad fuel options—but with the caveat that such fuels may not be as robust at high temperatures as the alternatives.

The IRP of MIT, UCB, and UW is developing a pre-conceptual design of a commercial prototype of an FHR and an FHTR. The pre-conceptual design of the commercial prototype will be based on pebble-bed graphite-matrix coated-particle fuel. However, the design of the FHTR should consider both near-term and long-term testing needs. In the early development of the LWR, there was rapid evolution of fuel types. The parallel history occurred with high-temperature reactor fuel. It would not be surprising if FHR fuels have a similar evolution and thus the need to consider future fuel options when considering design of a test reactor.

This appendix describes the basis for selection of the pebble-bed graphite-matrix coated-particle fuel as the mainline option, alternative fuel forms in terms of materials, and alternative fuel forms in terms of geometries.

A.1 Basis for Selection of Pebble-bed Graphite-matrix Coated particle fuel

The baseline commercial FHR fuel is the pebble-bed graphite-matrix coated-particle fuel (Figure A-1). This fuel has been demonstrated as a reliable, high-temperature fuel in helium-cooled pebble-bed reactors in Germany and China. Work by the Department of Energy's NGNP program has dramatically improved graphite-matrix coated-particle fuel performance.

The coated particle fuel consists of small particles of uranium oxocarbide covered with layers of carbon and SiC. These multiple layers are the clad that prevent the fission products from escaping into the coolant. The particles are the size of grains of sand. The particles are embedded in a graphite matrix that is the physical fuel element that is loaded into the reactor. The fuel element form can take many shapes. Reactors have successfully operated with the fuel form being pebbles, prismatic blocks, and cylinders.

The FHR baseline fuel is the pebble-bed graphite-matrix coated-particle fuel because it appears today to meet all the requirements for a large power reactor:

- *Chemical compatibility:* Graphite is chemically compatible in a radiation environment with high-temperate salts as was demonstrated in the MSRE (fuel dissolved in salt).

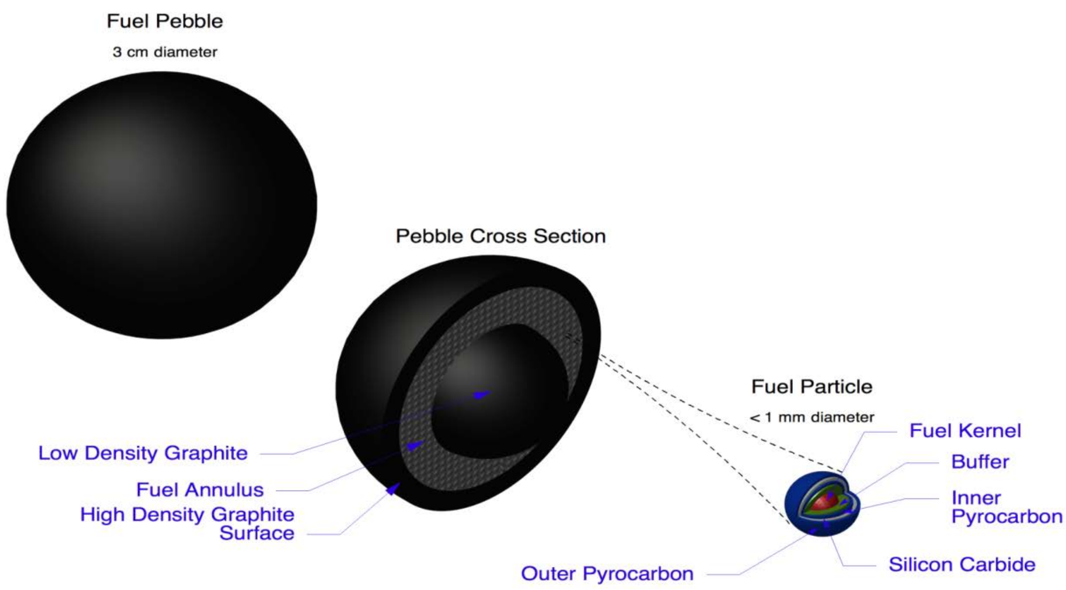

Figure A-1. FHR Base-Case Pebble-Bed Graphite-Matrix Coated-Particle Fuel

- *High-temperature capabilities:* The coated-particle fuel is today the only demonstrated high-temperature fuel
- *Refueling:* Graphite floats in liquid salts. In pebble-bed reactor the pebbles cycle through the reactor once a month. Refueling in a FHR is simplified because the pebbles float to the refueling machine (Forsberg 2008a; Forsberg 2006).

The other demonstrated coated-particle fuel is the prismatic-block coated-particle fuel used in the Fort St. Vrain reactor (see below). This fuel has the advantage of allowing a wider variation of fuel versus moderator versus coolant ratios in the reactor core. However, there are complications in refueling.

- *High-temperature fuel transfer*. The reactor core must be kept under liquid salt at >500°C to avoid overheating caused by decay heat and thus the mechanical parts of the refueling machine must operate at these temperatures.
- *Block stacking*. With current graphite technology the length of the prismatic block is limited to one to two meters. That implies that in a large reactor the reactor core is several fuel blocks high. While refueling a stack of blocks is simple in a gas-cooled reactor (gravity allows stacking of blocks) and was done without difficulty at the Fort St. Vrain reactor, refueling in liquid salt where the graphite wants to float would be mechanically difficult. This difficulty is what led to the ORNL proposal for a plank fuel (discussed later)—a graphite fuel assembly the height of the reactor that avoids the block stacking challenge and allows very conventional core designs.

The short height of the prismatic fuel is not a constraint for a small special purpose or test reactor where the reactor core is one block high. It is possible to add materials to the prismatic block to boost its density so it is heavier than the salt coolant.

A.2 Fuel Materials

There are near-term and longer-term fuel material options

A.2.1 Graphite-Matrix Coated-Particle Fuel

Graphite-matrix coated-particle fuel is the leading option for FHR fuel as it is the only fuel with demonstrated high-temperature capabilities and near-term availability (Forsberg et al. 2012). In this fuel type, fuel microspheres (TRISO particles) are encased in a graphite matrix, which can be shaped into many forms including pebbles, plates, and cylindrical compacts. The graphite serves as both a structural component and as the primary neutron moderator. Modern versions of this fuel have evolved from similar fuels that have been demonstrated in several helium-cooled high-temperature reactors. Peach Bottom Unit 1, the Fort St. Vrain Reactor (FSVR), the Arbeitsgemeinschaft Versuchsreaktor (AVR), the Thorium High-Temperature Reactor (THTR), and—more recently—the Chinese HTR-10 pebble-bed prototype reactor and the Japanese High-Temperature Test Reactor (HTTR) have all successfully demonstrated variations of graphite-matrix coated-particle fuel.

A.2.2 SiC-Matrix Coated-Particle Fuel

Silicon-carbide-matrix (SiC_m) coated-particle fuel (Figure A-2) is a variation of graphite-matrix coated-particle fuel that replaces the graphite matrix with SiC. The coated-particle fuel is unchanged. Only limited work has been done on this advanced fuel at Oak Ridge National Laboratory. This fuel was originally proposed as a new matrix fuel for accident tolerant LWR fuels but more recently has been proposed for the FHR (Forsberg et al. 2012).

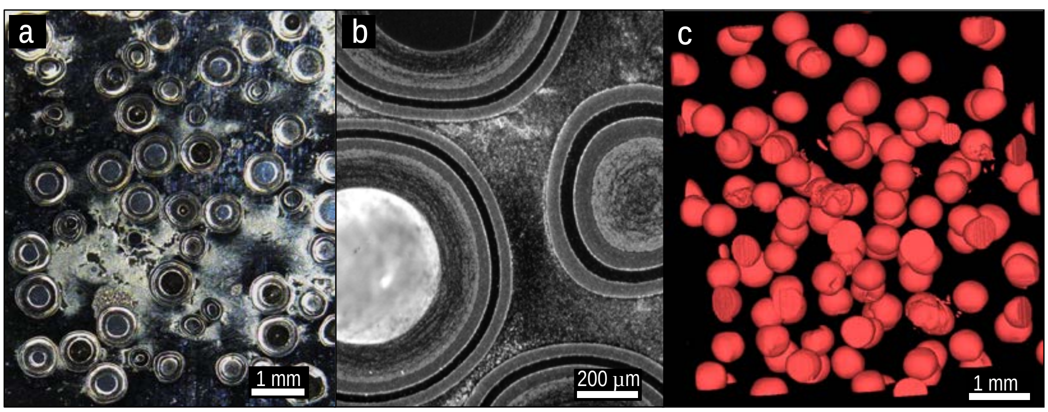

Figure A-2. (a,b) Optical Micrograph of Coated-Fuel Particles Dispersed in Nano-powder Infiltration and Transient Eutectoid SiC matrix, (c) X-Ray Tomography of Fuel Kernels Inside Coated-Fuel Particles Embedded in SiC Matrix Pellet

The objectives of this substitution are to exploit SiC's resistance to radiation damage and to create a fuel form that is more robust under accident and repository conditions. In a reactor, graphite first shrinks and then swells as a function of fast neutron fluence. SiC_m fuel provides much more dimensional stability under irradiation, as can be seen in Figure A-3.

There has been a large amount of work done on SiC for fusion and helium-cooled high-temperature reactors. However, there is almost no experience with SiC exposed to salt in radiation environments. SiC is used in extreme non-nuclear environments such as the after burners of military jet engines. It is also used in the chemical industry for extremely corrosive environments due to its resistance to corrosion.

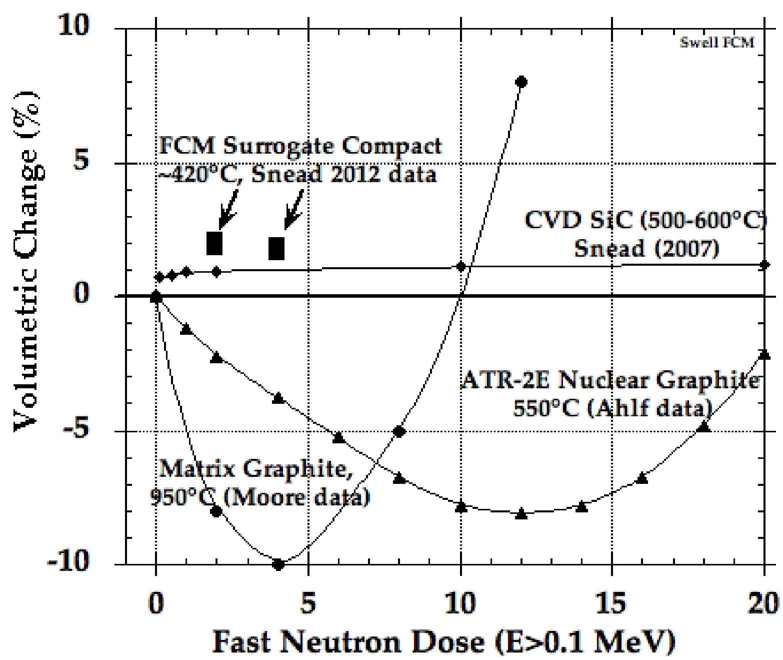

Figure A-3. Dimensional Change Contrast of CVD SiC and FCM Fuel with Nuclear Graphite ATR-2E and Matrix Graphite

The neutron moderation capability of SiC_m is a little less than graphite. SiC_m fuel has a significantly higher density (3.21 g/cm^3) than carbon (1.3 g/cm^3) but the carbon density in SiC (~0.96 g/cm^3) is somewhat less than graphite (1.3 g/cm^3). Consequently, it provides less neutron moderation than graphite-matrix fuel; however, there is also some neutron moderation from the silicon. If the power density of the TRISO particles and loading are unchanged, the core power density will be slightly lower if SiC_m fuel is used instead of graphite-matrix fuel. Additional moderation can be achieved in SiC_m fuel by increasing the thickness of the graphite layers in the TRISO fuel kernels and/or embedding graphite microspheres into the SiC_m.

The use of SiC_m in place of graphite-matrix fuel will also require changes to certain reactor systems, particularly the refueling system, as the density of SiC_m coated-particle fuel is greater than the density of the salt, so the fuel will not be buoyant.

The thermal conductivity of the SiC is less than graphite. This is a constraint in some reactor systems where the graphite thermal conductivity is used to remove decay heat.

A.2.3 Silicon-Carbide Pin Fuel

The longer-term option may exist to create an FHR with a pin-type fuel assembly. The graphite and moderator would be separated from the fuel. Conceptually this would be similar to the British Advanced Gas-Cooled Reactors with graphite matrix and pin type fuel assemblies.

The fuel would be in pellet form such as UO_2 rather than a coated-particle fuel. This could potentially be a lower cost fuel to fabricate and the pin design opens up a wider set of design options.

This is a longer-term option because the SiC pin must not only provide structural support but *it must be a sealed container for fission product gases*. While SiC has been developed as the cladding for coated particle fuel, the joining technology has not been fully perfected for sealing tubes.

Major work is underway to develop this type of clad for LWRs to replace zircalloy. Clad has is being irradiated at MIT under PWR conditions (pressure, temperature, chemistry control). If successfully developed for LWRs in the next decade, it would become a candidate for an FHR. One important factor is that for both LWRs and FHRs it is important for corrosion control to minimize free silica in the SiC. These parallel requirements suggest that development of this clad for LWRs will address most of the challenges in using it for FHRs.

A.3 Fuel Geometry

There are multiple possible fuel geometries for FHRs. The preferred option depends upon goals and time frame.

A.3.1 Pebble Fuel

The pebble fuel geometry is the preferred near-term form for a commercial FHR based on available technologies.

Pebble fuel is comprised of coated-particle fuel kernels embedded in spherical graphite elements. The pebble diameter for helium-cooled reactors is typically about 6 cm. Pebbles are added to the core until criticality is reached. Successful use of pebble fuel has been demonstrated in several high-temperature helium-cooled reactors: the Chinese HTR-10 and the German AVR and THTR. The Chinese FHR test reactor will use pebbles (Figure A-4) that are almost identical to those used in the Chinese helium-cooled HTR-10.

The proposed commercial U.S. FHR design (Figure A-1) has 3-cm pebbles with a somewhat different arrangement of coated-particle fuel in the graphite matrix to allow higher power densities. The smaller pebbles result in more surface area for heat transfer per unit volume of the reactor core. The pebbles have no fuel in the central region; the coated particle fuels are in the outer shell of the pebble. This reduces the temperature drop inside the pebble and allows operation at higher volumetric power levels.

Pebble fuel has several advantages over other geometries: (1) lower fabrication costs than other geometries because there is no complex geometry internal to the fuel; (2) ability to perform on-line refueling; and (3) less complex, and likely lower cost, refueling systems. These advantages—primarily the less-complex refueling requirements—led to the inclusion of pebble fuel in the baseline design of the commercial FHR.

Preliminary economic analyses indicate that particle fuels will have high fabrication costs than traditional fuel pins with UO_2 pellets. Continuous refueling may be able to offset high fabrication costs by maximizing burnup (i.e. decreasing the amount of fuel required for producing a given amount of energy). Pebble fuel is the only fuel geometry under consideration that allows continuous refueling, which allows improved neutron economy and higher burnup because the reactor can operate without any excess reactivity.

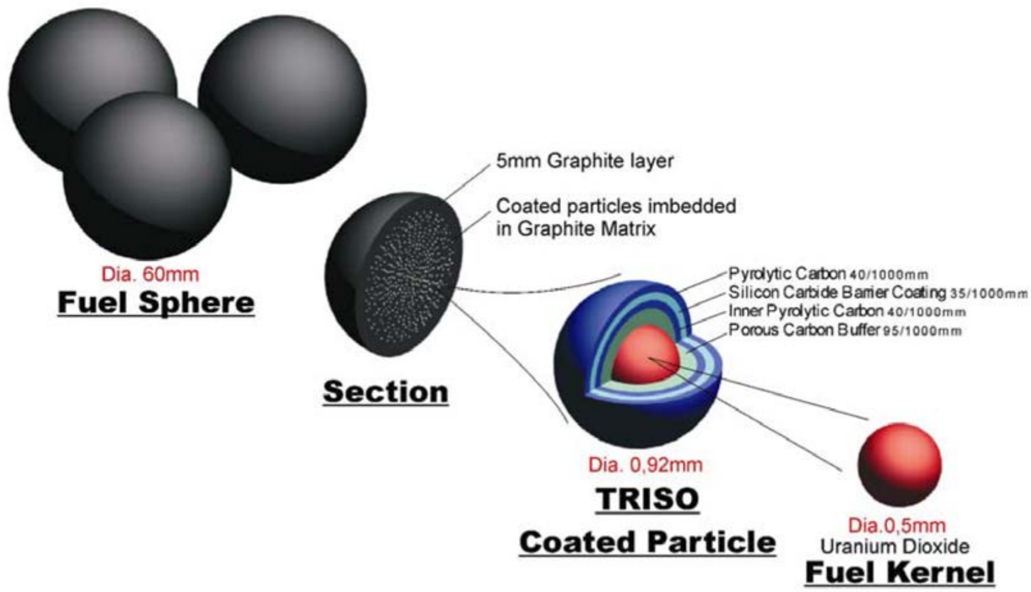

Figure A-4. Illustration of the Composition of Pebble Fuel

The same property of pebble fuel that enables continuous refueling is not without drawbacks. The stochastic movement of pebbles implies that instrumentation cannot be attached or inserted directly into the fuel. Additionally, if graphite-matrix fuel is used, the pebbles will float freely in the coolant salt because the density of the fuel is lower than that of the salt. In contrast, the fuel in all of the other geometries is static, meaning that the fuel can be prevented from floating by adding weights or directly attaching the fuel to the surrounding structures (Forsberg 2008a).

Advantages

- Demonstrated technology
- Pebble bed cores make it nearly impossible to have major coolant channel blockages because there are no narrow cooling channels.
- Continuous and less complex refueling
- Potentially higher capacity factor with online refueling
- Higher burnup than in other fuel geometries

Disadvantages

- Low flexibility in fuel-to-carbon-to-coolant ratio. The high coolant fraction in the core requires the use of a salt coolant with a very low nuclear absorption cross section to avoid a positive void coefficient.
- Instrumentation cannot be located directly in the fuel

- High coolant inventory with associated costs and limited choices of coolants because the coolant-to-sphere ratio is fixed.

A.3.2 Prismatic-Block Fuel

The prismatic fuel block is a near-term fuel option for small FHRs (test reactors or special purpose reactors). It is a less likely option for a commercial reactor because of the difficulty in refueling a floating fuel that must be stacked like a set of blocks in the reactor.

Prismatic-block fuel was originally developed for use in HTGRs. In this implementation, fuel microspheres are encased in a graphite matrix, which is shaped into the form of fuel compacts. The more recent HTTR (Japan) fuel incorporated annular fuel compacts. The annular compacts were designed to decrease the peak fuel temperature by keeping the center of the fuel element, where the peak temperature would otherwise occur, free of fuel.

In the Japanese HTTR (Figure A-5), the compacts are loaded into hollow, cylindrical sleeves that are placed in channels in large prismatic blocks of graphite. Coolant flows between these sleeves and the graphite block. At FSVR, these compacts (Figure A-6) are loaded into fuel channels in the graphite block. Coolant flows through separate channels in the graphite blocks. In this design the size of coolant hole is independent of the diameter of the fuel compact.

Coolant channels in a salt-cooled reactor can be made smaller than in gas-cooled reactors because of salt's superior heat transport properties relative to gas. Prismatic-block fuel has a large base of operating experience (in Peach Bottom Unit 1, the FSVR, and the HTTR) and the technology is considered fully developed for use gas-cooled reactors (Casino Jr. 2006; Forsberg 2008a).

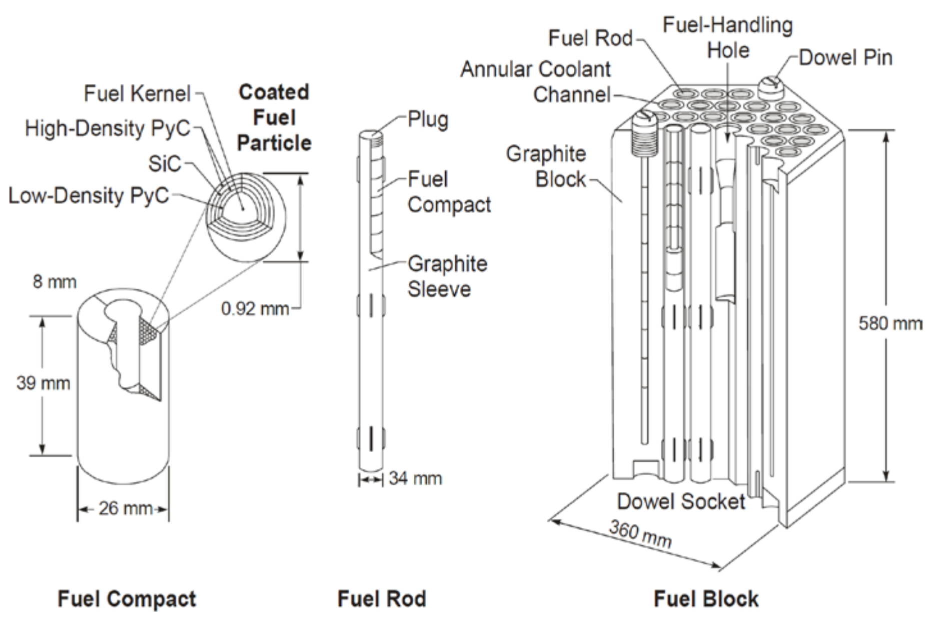

Figure A-5. HTTR Prismatic-Block Fuel

If it is required that the fuel element be more dense than the coolant to simplify refueling or other requirements, there is the option to add dense materials to the prismatic block by inclusion at the bottom of fuel holes a dense material. If only a small increase in density is required, the obvious material is SiC because of its high temperature and high resistance to radiation damage. This option may be viable for small reactors that are one-block high but more difficult to implement if the reactor core is multiple blocks high.

There are other fluoride salt options beyond flibe. The ability to vary the fuel to moderator (carbon) to coolant ratio may allow a wider choice of fluids beyond Flibe. Lower coolant fractions in the reactor core imply less parasitic neutron absorption by the coolant.

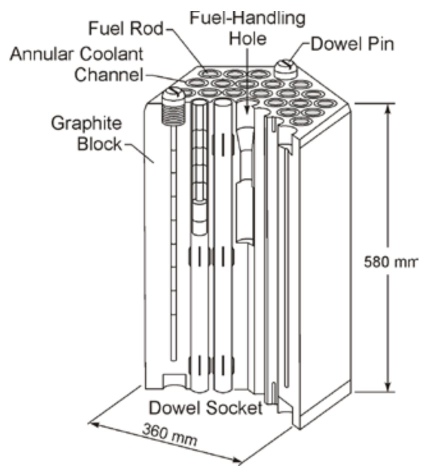

Figure A-6. Fort St. Vrain Prismatic-Block Fuel

There is a significant difference in the manufacturing process for pebble bed fuel versus prismatic block fuel. As a consequence, the prismatic block fuel in many cases is more robust. The pebbles and the annular fuel compacts used in prismatic block are made by mixing coated-particle fuel with binder and graphite. The mixture is compressed into the appropriate form and heated to consolidate the material into the pebble or annular compact form. The processing of the matrix material is constrained by the requirement not to damage the coated-particle fuel.

In the prismatic block form, a graphite prismatic block is made without fuel and holes are drilled in the graphite for coolant channels and for fuel channels. The production process to make the graphite block includes bake out at temperatures far above the failure temperatures of the coated-particle fuel and thus produces a superior graphite material. The block has superior properties because there are no temperature or other processing limits in its production.

Advantages

- Large base of operational and fuel-fabrication experience
- High flexibility in fuel-to-carbon-to-coolant ratio
- Very robust prismatic block
- Instrumentation can be located directly in the fuel

Disadvantages

- Block height is limited. As a consequence, refueling is more difficult for a large commercial FHR where there are multiple fuel blocks stacked on top of each other.

A.3.3 Plate Fuel

Graphite-matrix coated-particle plate fuel is a fixed (static) fuel that is under development at Oak Ridge National Laboratory. It is being developed to address the complications of prismatic fuel block fuel in a commercial FHR. This fuel variant is composed of plates (slabs) of graphite-matrix coated-particle fuel arranged into hexagonal fuel assemblies. The fuel in each plate is located in fuel stripes on both sides of the plate. This results in lower peak temperatures relative to a plate of uniform fuel distribution. The space between the plates provides low-resistance channels for coolant flow, which enable increased passive cooling by natural circulation of the liquid salt during loss of forced circulation scenarios.

Plate fuel uses the same materials as pebble fuel, but arranged into a different geometric form. The similarity between pebble and plate fuel appears to indicate that plate fuel should not be significantly more difficult to manufacture than pebble fuel, but unlike pebble fuel, plate fuel allows high flexibility in the fuel-to-carbon-to-coolant ratio. However, because plate fuel is fixed in place, lower burnup at the top and bottom of each fuel plate contributes to a lower average burnup relative to pebble fuel, which achieves uniform burnup through the continual movement of the fuel through the core (Casino Jr. 2006).

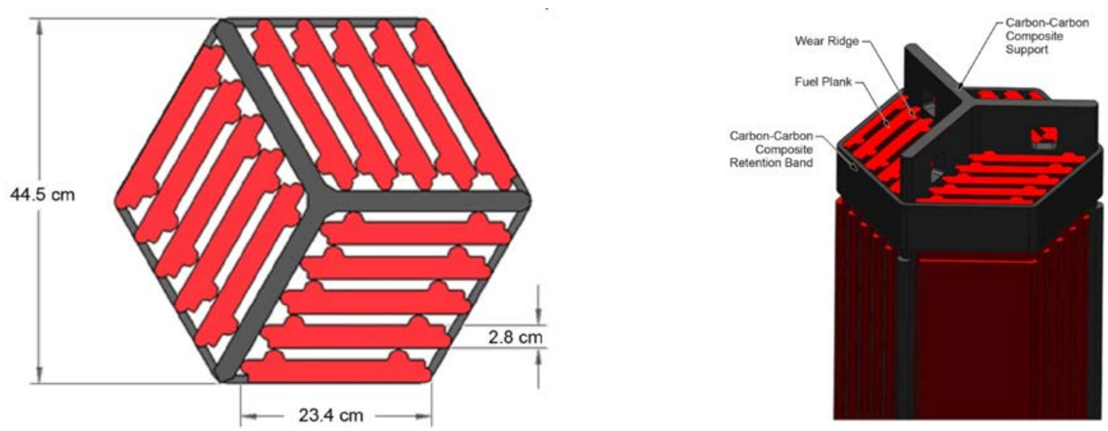

Figure A-7. Illustration of a Hexagonal Plate Fuel Assembly

Advantages

- High flexibility in fuel-to-carbon-to-coolant ratio
- Instrumentation can be located directly in the fuel
- Enhanced passive cooling capabilities
- Enables refueling with an assembly-type fuel

Disadvantages

- No prior experience with manufacturing plate fuel

A.3.4 Pin Fuel

Pin-type fuel (Figure A-8), comprising UO_2 fuel pellets surrounded by SiC or high-nickel alloy (e.g. Alloy-N, also called Hastelloy®-N and INOR-8) cladding, has been considered for use in FHRs (Casino Jr. 2006). Fuel pins are typically arranged in clustered assemblies, in which multiple concentric rings of fuel pins and a single central support pin are held in place by a grid structure and are moved as a single unit. The core is arranged as a square lattice array of clustered fuel assemblies surrounded by blocks of graphite moderator.

This arrangement of clustered assemblies and graphite moderator has been commercially deployed in the United Kingdom's fourteen commercial Advanced Gas Reactors (AGRs) that used carbon dioxide gas as the coolant. The AGRs refuel at high temperatures when the reactor is operating—a remarkable capability given that this reactor operates at high pressure.

In AGRs, stainless steel is used for cladding and for the grid structure that holds each assembly together. For use in FHRs, these components could be constructed of SiC or metal. The British did limited work on using SiC for this application. The challenges include sealing pins to hold in fission gases and demonstrating the performance of SiC in this application. If metal is used, it would require a high-nickel alloy clad for corrosion resistance in liquid salt. Historically high-nickel alloy clad has not been considered viable because of neutron absorption that creates alpha particles resulting in helium inclusions that weaken the clad. Recent advances in centrifuge technology may allow isotopic separation of nickel isotopes to create nickel-alloy clad without helium generation and buildup. However, no significant work has been done to develop such nickel cladding.

Either option involves major technical uncertainties and a long development program. The SiC option is preferred because of its much higher temperature capability. SiC is being considered as an advanced cladding material for LWRs. If it is successfully developed for this application, it could become a candidate for an FHR with pin fuel assemblies.

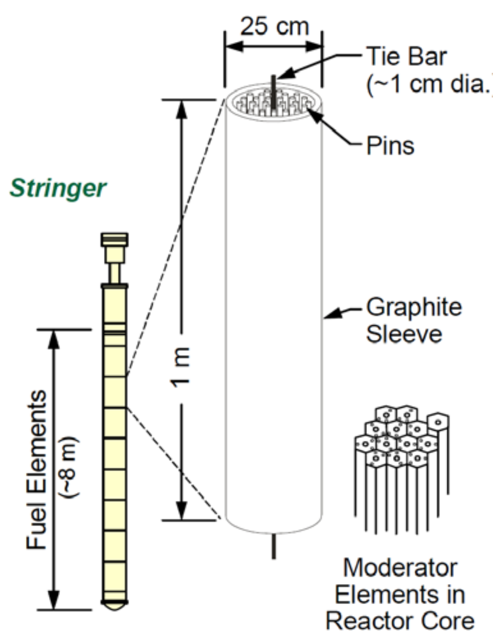

Figure A-8. Pin-Type ("stringer") Fuel

The primary potential advantage of using pin-type fuel is improved economics because of the lower cost of fabricating fuel pellets and through the use of existing fuel fabrication infrastructure. Commercial facilities have decades of experience in producing pin-type fuel assemblies at predictable costs. There is no equivalent infrastructure for any other fuel form (Greene et al. 2010; Forsberg 2008a).

The second major incentive for using pin-type fuel is the physical decoupling of fuel and moderator in the core. Graphite lifetime in a reactor is limited by fast neutron fluence, but in typical FHR configurations, the fuel lifetime in the reactor core is much less than the graphite lifetime. This results in unnecessary graphite waste production in configurations in which the moderator and fuel are inseparable (as is the case with pebble bed, plate, and prismatic-block fuel). The separation of fuel and moderator also simplifies refueling operations. The moderator blocks can be fastened in the core to counteract their buoyancy, thereby enabling maintenance and refueling without the complication of fuel-block movement (Greene et al. 2010).

The second potential option is use of SiC moderator blocks to replace the graphite moderator. The major advantage is the much greater capability to resist radiation damage. In this application, the lower thermal conductivity of SiC relative to graphite would not likely be a significant constraint because the blocks are only heated by gamma and neutron heating—no fuel in the blocks. The long-term endurance of SiC in fluoride salts has yet to be demonstrated. Such options have not been examined.

Advantages

- Low fabrication cost
- Lower graphite waste production
- High flexibility in fuel-to-carbon-to-coolant ratio
- High flexibility in axial fuel enrichment and radial fissile loadings
- Instrumentation can be located directly in the fuel

Disadvantages

- No operational experience with pins in high-temperature fluoride-salt environment
- Major R&D to demonstrate pin feasibility.

A.4 Summary

The near-term fuel option for a commercial reactor is the pebble-bed graphite-matrix coated-particle fuel. In the longer term, there may be additional options as progress continues on alternative fuels with different materials and geometries. The development of alternative fuels will be partly dependent upon developments in materials and what are the reactor goals: commercial electricity, heat for industry, actinide waste burning, small reactors for remote sites, and other applications. A partial summary is shown in Table A-1 that shows there is no universally preferable option. It is entirely possible that if all the fuel forms were fully developed and investigated that the first fuel form, the pebble, would remain the preferred option.

Table A-1. Commercial Fuel Types for FHRs (1 Best; 4 Worst)*

Fuel Geometry	Technology Status	Freedom in Core Design	Refueling	Waste Volume
Pebble	1	4	1	2
Plate	3	3	2	2
Hexagonal Block	1	2	4	2
Pin	4	1	2	1

* The prismatic fuel form may be more robust because there are no constraints on processing conditions to create the block because the fuel is added latter.

Appendix B Summary of the Scoping Analysis for an FHR Design to Test Specific Concepts

Section 4.3.1 addresses the complexity of attempting to match physics of the PB-FHR during operation in a FHTR core. A sampling of cores that reflected characteristic parameters of core physics under normal PB-FHR operation were presented. However, a larger phase-space was investigated for purposes of the scoping analysis. Here a summary of that scoping analysis is presented.

Each candidate FHTR was analyzed at its equilibrium state and compared to the PB-FHR at its equilibrium state. As such, each core is analyzed at a different burnup, which it achieves at its equilibrium cycle. Because the FHTR will not be optimized for fuel burnup, analyzing the core physics at comparable burnup states was not a priority. At high burnups, however, spectral effects do not change significantly for a single core design, as seen in Figure B-1.

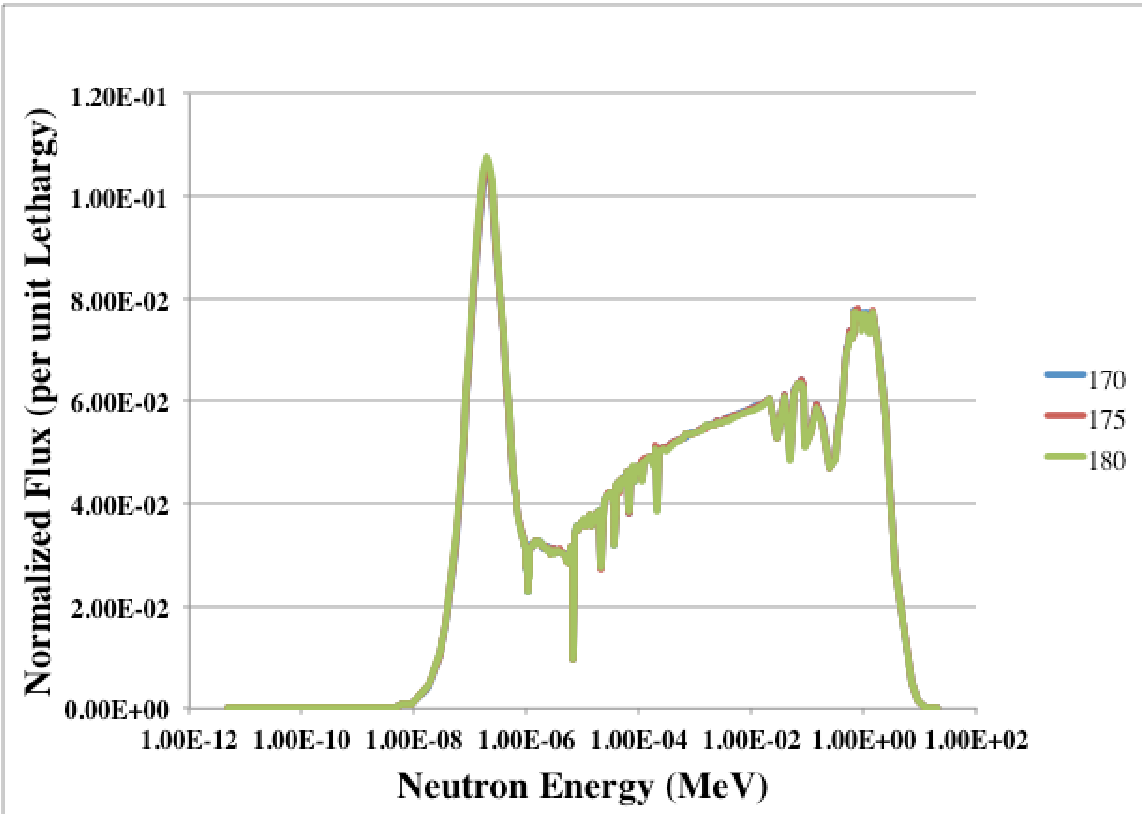

Figure B-1. Effect of Burnup on Neutron Energy Spectrum (2.0 Aspect Ratio, 75.0 C/HM)

The parameters for the core design in Figure B-1 were chosen because the higher C/HM loadings lead to harder neutron energy spectra, which exhibit a higher sensitivity to spectral softening at these burnup steps than do softer spectrum cores. The burnup ranges chosen here encompass all equilibrium burnup states achieved by FHTR cores in the scoping analysis. Even with a more sensitive core, the effect from increasing burnup to changing the neutron energy spectrum is small compared to the differences in neutron energy spectra caused by differences fuel loading or core geometry, which can be found in subsequent pages.

B.1 Changing the Aspect Ratio

The aspect ratio of the core is defined in Section 4.3.2. Initially, the scoping analysis aimed at changing (1) the core geometry, or (2) the fuel loading to see their effects on core reactor physics. The neutron balance in both systems is shared between the fuel, graphite (in the pebbles and in the reflector), and in the coolant. By changing the aspect ratio, changes in the leakage also affects other core physics. Figure B-2 shows that as aspect ratio is increased, the neutron energy spectrum hardens. Table B-1 includes the effect of changing aspect ratio on the temperature reactivity coefficients in the coolant and fuel. In general, the coolant temperature reactivity coefficient becomes more negative as the aspect ratio is increased, and the fuel temperature reactivity coefficient reaches its most negative point at the lowest leakage core, with an aspect ratio of 1.414 (optimized for the lowest surface area to volume ratio). Note that all of these cores have the same fuel loading, 200 C/HM.

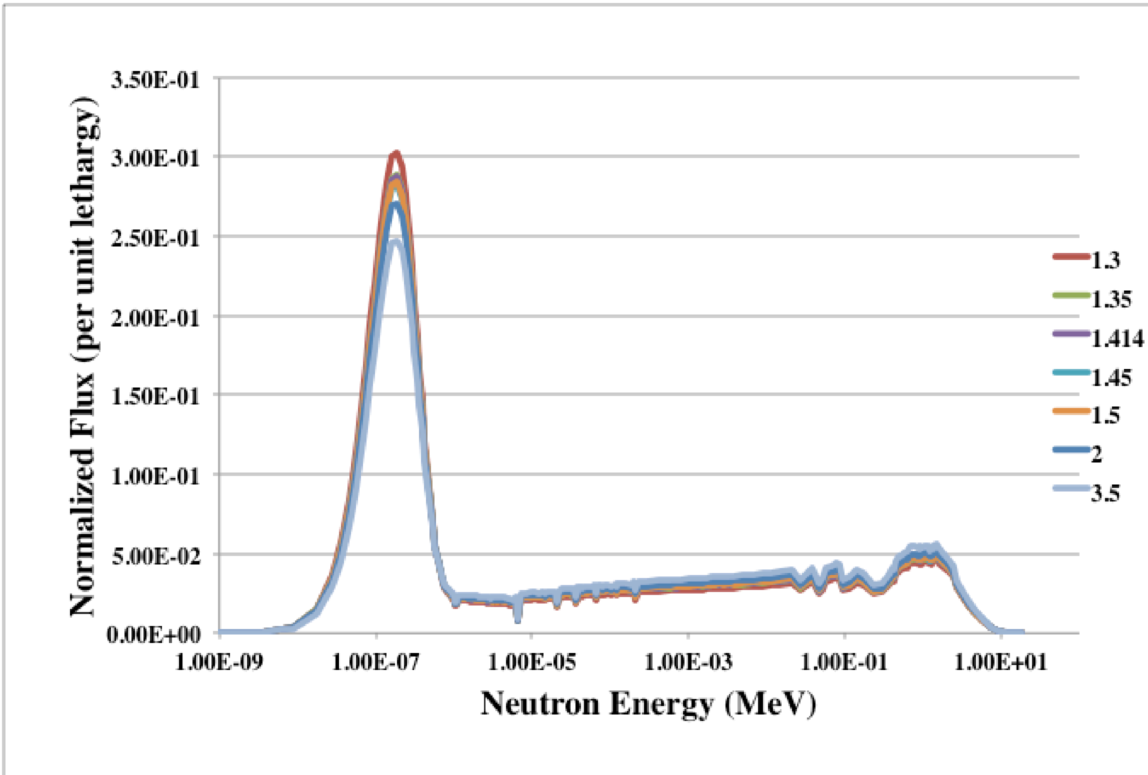

Figure B-2. Effect of Aspect Ratio on Neutron Energy Spectrum at Equilibrium for FHTR Cores

Table B-1. Effect of Aspect Ratio on the Temperature Reactivity Coefficients of the Coolant and Fuel for FHTR Cores

Aspect Ratio	Coolant TRC, pcm/K	Fuel TRC, pcm/K
1.3	6.33E-02	-1.01E+00
1.414	-4.88E-01	-1.68E+00
1.5	-3.94E-01	-1.48E+00
2	-6.223E-01	-1.411E+00
2.5	-9.095E-01	-1.602E+00
3	-9.804E-01	-1.534E+00
3.5	-1.467E+00	-1.560E+00

B.2 Changing the C/HM

As mentioned in the body of the whitepaper, the C/HM loading of the fuel is a variable that can easily be changed under normal conditions, during steady state operation. This enables the test reactor, should an aspect ratio to be fixed, to be flexible in the physics that it can reflect from the PB-FHR. Figure B-3 shows that as the C/HM is increased, the spectrum initially softens, but then hardens slightly at high C/HM loadings. This is likely due to competing effects in the physics of the core, which have yet to be characterized. Additionally, the ranges between neutron energy spectra of candidate FHTR designs are more sensitive to C/HM loading than to variations in the aspect ratio.

To further illustrate the complexity of increasing C/HM on the FHTR core reactor physics, Table B-2 includes the temperature reactivity coefficients of the coolant and the fuel. Unlike changing the aspect ratio, where a trend in the TRCs were clear, the very low and very high C/HM loadings exhibit the most negative TRCs.

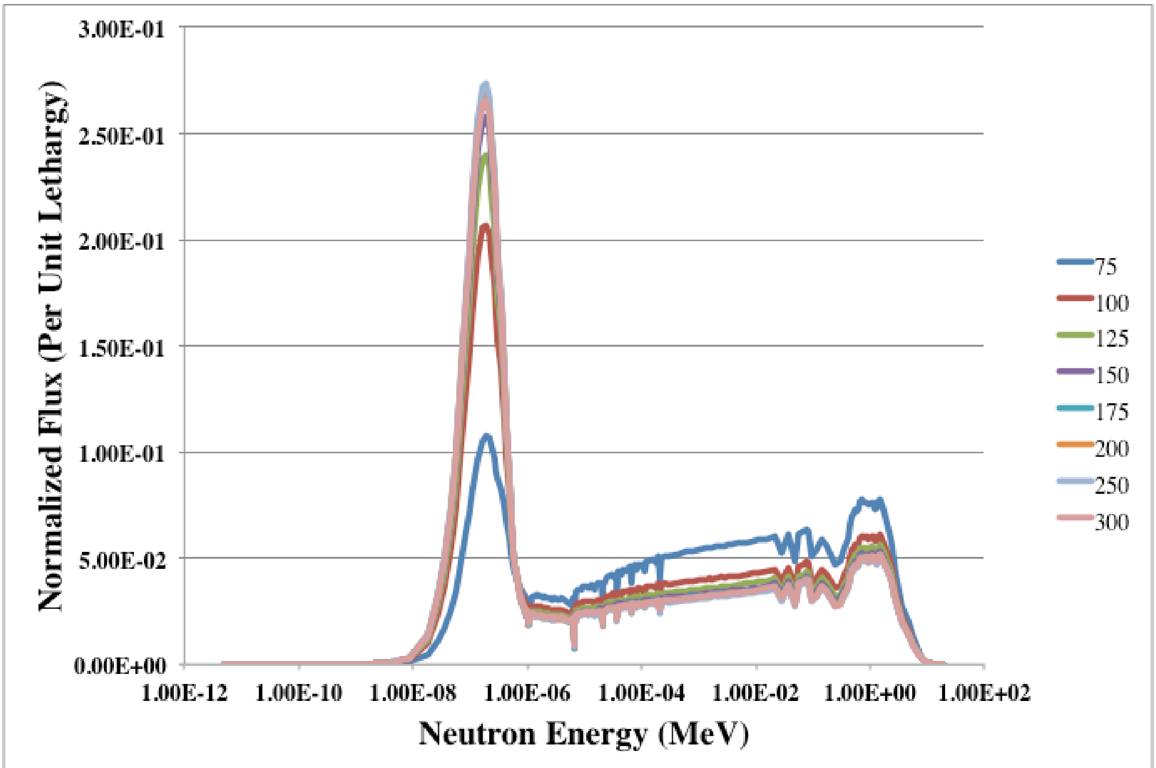

Figure B-3. Effect of the C/HM on the Neutron Energy Spectrum at Equilibrium for FHTR Cores

Table B-2. Effect of C/HM Loading on the Temperature Reactivity Coefficient in the Coolant and Fuel for FHTR Cores

C/HM	Coolant TRC, pcm/K	Fuel TRC, pcm/K
75	-3.62E+00	-3.61E+00
100	-2.11E+00	-1.98E+00
125	-1.95E+00	-1.48E+00
150	-1.56E+00	-1.51E+00
175	-1.43E+00	-1.53E+00
200	-2.25E-01	-1.59E+00
300	8.73E-02	-1.24E+00
450	-3.76E-01	-1.03E+00
550	-4.40E-01	-1.12E+00

The scoping analysis showed that the FHTR cores were much more sensitive to C/HM fuel loadings to shift core physics than to aspect ratios. However, it should be noted that there are a number of competing effects that change the core physics when the C/HM is altered in a pebble. For example, a 75 C/HM pebble may achieve a significantly different burnup than the baseline PB-FHR fuel, at 300 C/HM. While the spectra are different between these two, so are the relative amounts of fission products in each pebble. Because such a large range of C/HM loadings were sampled, it is possible that leakage effects may be more dominant for one range of C/HMs, and spectral effects my dominate for another. It should be noted that these scoping studies are still in progress, so not all effects have been characterized fully.

B.3 Fission Product Decay

In addition to ensuring that candidate cores have negative coolant and fuel temperature reactivity coefficients, ensuring that cores can maintain subcriticality post-shutdown is also a priority. Consequently, the shutdown systems must have enough negative reactivity insertion to accommodate for the densification and cooling of the salt, the cooling of the fuel, and of the fission product decay.

By changing aspect ratio in the system, slight spectral effects were observed, as well as relative changes in the neutron absorptions in the graphite and the fuel. As a result, different equilibrium burnups were observed, as well as different equilibrium Xe concentrations in the core. Table B-3 includes burnup and Xe data with aspect ratio for cores with 200 C/HM loaded fuel. The >2000 pcm difference in reactivity due to Xe decay is substantial, and cannot be discounted when designing the shutdown systems.

Table B-3. Burnup and Xenon Data with Aspect Ratio for Cores with 200 C/HM-Loaded Fuel

Aspect Ratio	Equilibrium Burnup, MWd/MWth	Reactivity Insertion From Xe Decay, pcm
1.3	1.868E+05	2.35E+03
1.35	1.879E+05	2.34E+03
1.414	1.873E+05	2.34E+03
1.45	1.876E+05	2.33E+03
1.5	1.870E+05	2.34E+03
2	1.859E+05	2.31E+03
2.5	1.835E+05	2.29E+03
3	1.825E+05	2.28E+03

Aspect Ratio	Equilibrium Burnup, MWd/MWth	Reactivity Insertion From Xe Decay, pcm
3.5	1.804E+05	2.27E+03

The equilibrium Xe reactivity insertion due to fission product decay was calculated using MCNP.

B.4 Summary of Scoping Analysis Results

The scoping analysis is still in progress, so no FHTR core has been identified. However, it is evident that no single FHTR core can simultaneously reflect the PB-FHR spectrum, reactivity coefficients, and fuel loading. However, it is not physically possible to change the aspect ratio of the FHTR during normal operation, while the C/HM can be adjusted at any time due to the form of the recirculating pebble fuel. Thus, choosing a candidate aspect ratio core is of greater importance than choosing a single C/HM fuel loading. Prior to identifying a single candidate FHTR core, it will be necessary to determine the competing effects that cause variations in the trends with C/HM and with aspect ratio, as the results indicate that a number of factors are likely affecting the reactivity coefficients and neutron energy spectra.

B.5 Future Work

In addition to determining a characteristic test reactor core that can operate safely at equilibrium, it will be necessary to find a core that can start-up and transition to equilibrium while maintaining a safe criticality. Therefore, once a candidate FHTR core is chosen, startup options and the transition to equilibrium must be analyzed. Startup cores may include, but are not limited to: inserting inert graphite pebbles, inserting pebbles loading with burnable poisons, achieving criticality with a shorter bed height, and inserting pebbles loaded with fertile fuel. The transition to equilibrium may also be achieved using some combination of these startup options.

Appendix C FHTR Startup Testing

C.1 Necessary and Likely Subcritical and Low-Power Neutronics Startup Tests

In this section, the subcritical and low-power neutronics startup tests that are necessary and likely to be performed are described. Emphasis is placed upon characteristics that are unique to the FHTR.

During the 1/M approach to criticality, the subcritical multiplication of an external neutron source will be measured for a number of subcritical configurations that are progressively closer to critical. Simple simulations have shown that a traditional geometry is appropriate for the FHTR: a neutron source will reside in the outer reflector at the mid-plane of the core and neutron detectors will measure neutron flux at multiple locations within the opposing side of the outer reflector. As the system nears a critical condition, the neutron source will be multiplied to a larger degree over successively more generations and the flux will approach the fundamental mode; the external source will be but a small perturbation upon the fundamental mode long before the system is critical. This procedure will be repeated, if multiple cores are envisioned for the FHTR.

Differential reactivity worths of control positions are not measured directly. Instead, during the low-power excursions that results from a reactivity insertion, the stable supercritical period is measured and, knowing the average neutron lifetime, the reactivity worth is calculated. The system is first granted sufficient excess reactivity to be critical with a control rod fully inserted. The inserted control rod is quickly withdrawn a small amount, the resultant stable period is measured, and using a second control rod the system is returned to a critical state at its initial power. The procedure is repeated until either control rod is inserted or withdrawn to its full extent. The reactivity inserted and removed from each control rod maneuver is equal, so worth curves for two control rods are measured at the same time. Because these control rod worth measurements are the basis for all subsequent reactivity worth measurements (reactivity worths from a variety of experiments are inferred from differences in the summed worth of all control rod positions) they must be performed slowly and carefully to minimize error and uncertainty. These worth curves will vary with pebble loading.

Some examples of off-normal conditions for static reactivity measurements are various isothermal conditions, various pebble configurations, and mocked lodging of pebbles. The former will be used to measure isothermal temperature reactivity coefficients.

Every measurement performed on the FHTR will be used in the neutronics simulation validation portfolio, but some tests will be performed solely for the purposes of validation. Multiple core configurations will be assembled, with varying pebble carbon-to-heavy-metal atom ratios, fuel enrichments, core aspect ratios, pebble configurations, and system carbon-to-flibe atom ratios. The bulk of data will likely be made up of reactivity worth and critical state measurements, but may also be supplemented by flux magnitude traverses and NAA of metallic films.

Whether or not flux diagnostics are a reliable indicator of core power during operation is the subject of noise measurements. During these measurements, the temporal behavior (the harmonic modes) of flux magnitude instruments is considered with stagnant and flowing coolant, and with stagnant and flowing pebbles. The signal-to-noise ratio may be a constraint defined by the technical specifications.

Dynamic reactivity tests are performed at full power, where thermofluid interactions can take place. Full-power operation produces a significant amount of fission products, whose decay masks flux measurement at low-power, and whose parasitic absorption changes reactivity characteristics. These tests should be performed only after low-power tests are complete. In addition to traditional transients such as LOFC, LOHS, and ATWS, some scenarios specific to pebble-bed FHRs may be considered, such as bed fluidization and reserve shutdown SCRAM.

C.2 Historical Accuracy of Startup Testing Predictions

Review of the accuracy of historical physical benchmarks can provide some useful context for the bias that will inevitably exist between simulated and experimentally measured results in startup testing.

The Molten Salt Reactor Experiment (MSRE) was a reactor that underwent startup testing in 1965 (Prince et al. 1968; Engel and Prince 1967). While the concentration of ^{235}U necessary for its initial criticality was predicted within experimental uncertainty, the differential reactivity worth of the concentration was in error by 5%. The integral control rod worths were 10% off, the temperature reactivity coefficient was mis-predicted by 20%, and the power reactivity coefficient was predicted to be -0.007% δk/k/MW, but was measured as +0.001% δk/k/MW—a bias of 800%!

After the MSRE and in preparation for the Molten Salt Breeder Reactor, a critical experiment was performed in the High Temperature Lattice Test Reactor facility in 1971 (Lippincott 1972). The infinite multiplication factor was mis-predicted by 2% (2000pcm) and the difference between cold- and hot-zero power multiplication factors was mis-predicted by 70%.

From 1998-2000, the HTTR gas-cooled reactor performed various startup tests and reactor physics measurements (Bess et al. 2009; Bess et al. 2011). keff was over-predicted by 2% and excess and shutdown reactivity measurements differed from predictions by 7% and 10-20%, respectively. Isothermal temperature reactivity coefficients deviated by 20-40%, while experimental uncertainty was 20-30%.

The initial criticality of the HTR-10 took place in 2000 (Terry et al. 2007). A collection of simulations, with varying degrees of model complexity and method rigor consistently over-predicted the initial criticality by ~1%.

The ASTRA critical experiment took place during 2003-2004 (Ponomarev et al. 2008). Simulations tended to under-predict keff by ~0.3%.

Reactor physics tests for the initial criticality of Taiwan Unit 1 (a VVER) in 2005 considered reactivity characteristics with varying boron poison concentrations (Astakhov et al. 2006). The critical boron concentration and integral control rod worth, were predicted with a bias of <2%

and 6%, respectively. The coolant temperature reactivity coefficient with varying boron concentration was predicted with 10-20% bias.

C.3 Startup Testing Acceptance Criteria

During startup testing of the FHTR, a number of safety parameters—including the initial critical configuration, shutdown reactivity margin, reactivity coefficients, and differential and integral control rod worths—will be measured and the bias that simulated predictions of these parameters possess will be realized (NRC 1996). Those biases must remain within the pre-established acceptance criteria and operating limits as established by the technical specifications and those limits will be based upon safety requirements. An example limit is that the overall power reactivity coefficient remain negative.

Although the operating limits will be safety-based, some consideration of simulation uncertainty might be required. It will be important to know how likely it is for safety parameters (which are predicted with some uncertainty) to violate those limits. Comprehensive uncertainty quantification of each safety parameter will achieve this. Traditional sensitivity-based propagation of nuclear data uncertainty will be combined with stochastic variations in materials, dimensions, and pebble bed arrangements. Deficiencies in nuclear data covariances (e.g., thermal capture in natural carbon) can also be supplemented when needed.

Additionally, a number of critical experiments have been performed with TRISO fuel, carbon moderator and flibe moderator. The bias that might occur in an FHTR can be estimated by quantifying the bias of simulations with experimental measurements for neutronically similar systems. With educated guesses for which nuclear data is responsible for the largest biases, validation experiments can be focused upon them to reduce bias efficiently.

C.4 Sources of Experimental Uncertainty for Context and Avoidance

The initial criticality will be subject to a number of epistemic experimental uncertainties due to stochastic variation in the geometry and configuration of structures and pebbles, manufacturing variance in component dimensions, densities, and the composition of flibe and graphite. These could either be measured explicitly for the given system or statistically characterized over their probable distributions. After each adjustment during the 1/M approach to criticality, it will take an increasing amount of time for the system to asymptotically reach subcritical steady-state. If control rod position is used to approach criticality, there will be depressions that alter the flux distribution from the fundamental mode. If pebble bed height (a somewhat ambiguous metric) is used to approach criticality, the fundamental mode will move spatially. It will be important to locate the flux magnitude in regions that have gentle gradients so they are less sensitive to variations like these.

For control rod worth measurements, there is noise in the flux magnitude detectors and uncertainty in the delayed neutron precursor yields, branching ratios, and half-lives. In the absence of temperature feedback, it takes a while to determine if a system is low-power critical preceding a supercritical excursion (especially with flux monitor noise). During an excursion, sufficient time must also elapse for the stable period to emerge.

C.5 Discussion of Intrinsic Neutron Sources

Enrichment of ^{235}U in natural uranium, increases the abundance of ^{234}U (a much shorter-lived isotope) as well. In the MSRE, α-decay of ^{234}U and subsequent (α,n) reactions on flibe constituents produced ~1.5×10^6 n/s/kg-^{234}U (Haubenreich 1963). This was a non-negligible uniform neutron source (good for 1/M), which is comparable to the specific neutron source rate from ~2.3×10^6 n/s/kg-^{252}Cf. In the α-decay occurred within flibe; in FHR, α's will be deposited in the TRISO fuel, so intrinsic neutron source rates will be much lower.

Appendix D Preliminary PCU Model Results

This appendix lists graphical results from a two expansion stage open air combined cycle described in the main white paper.

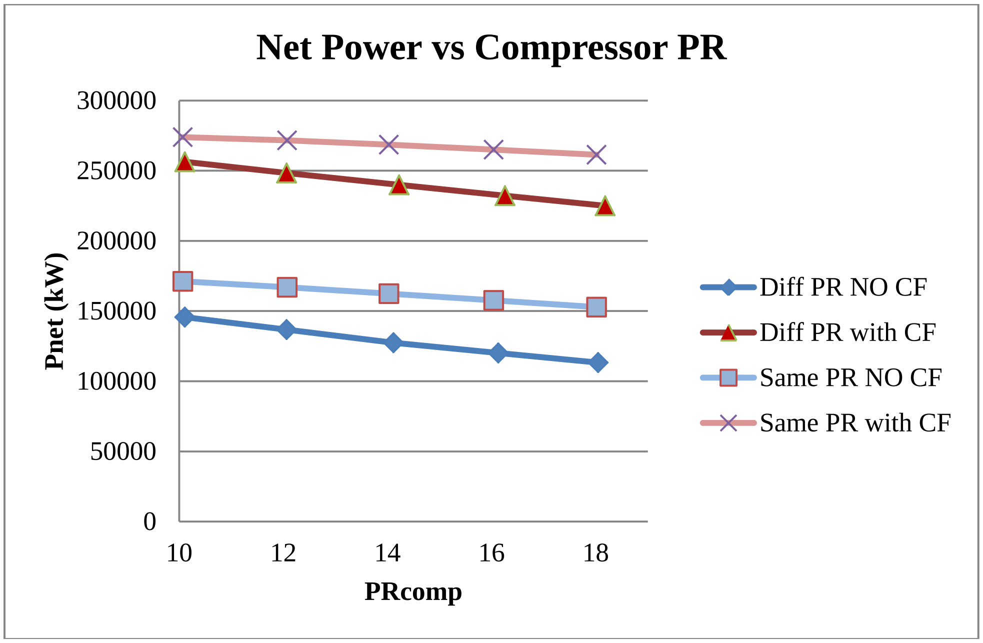

Figure D-1. Net Power vs. Compressor PR

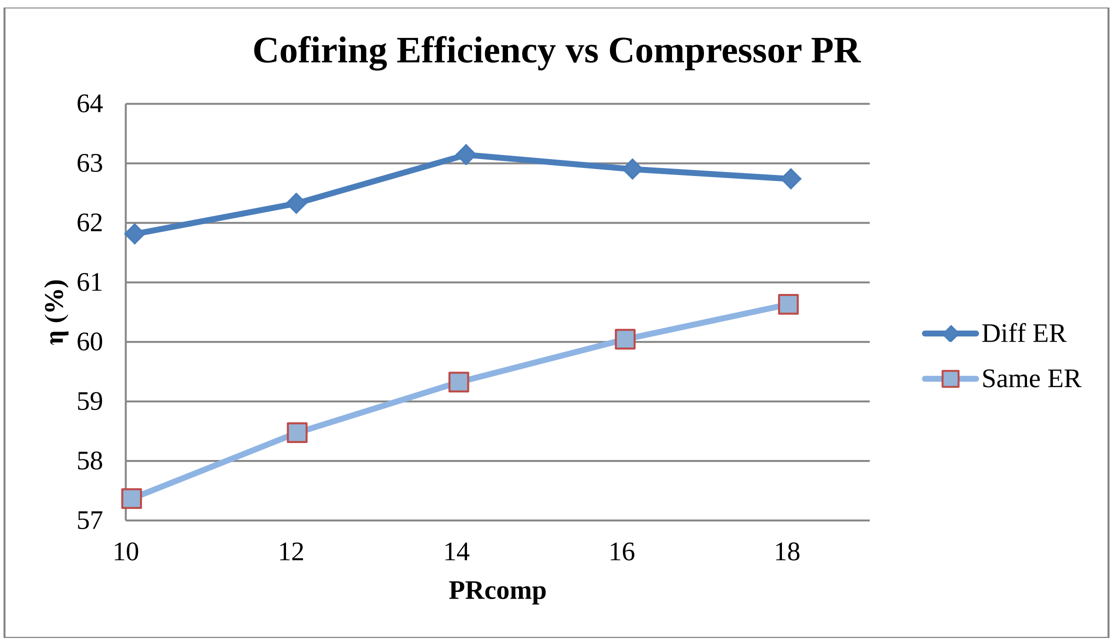

Figure D-2. Cofiring Efficiency vs. Compressor PR

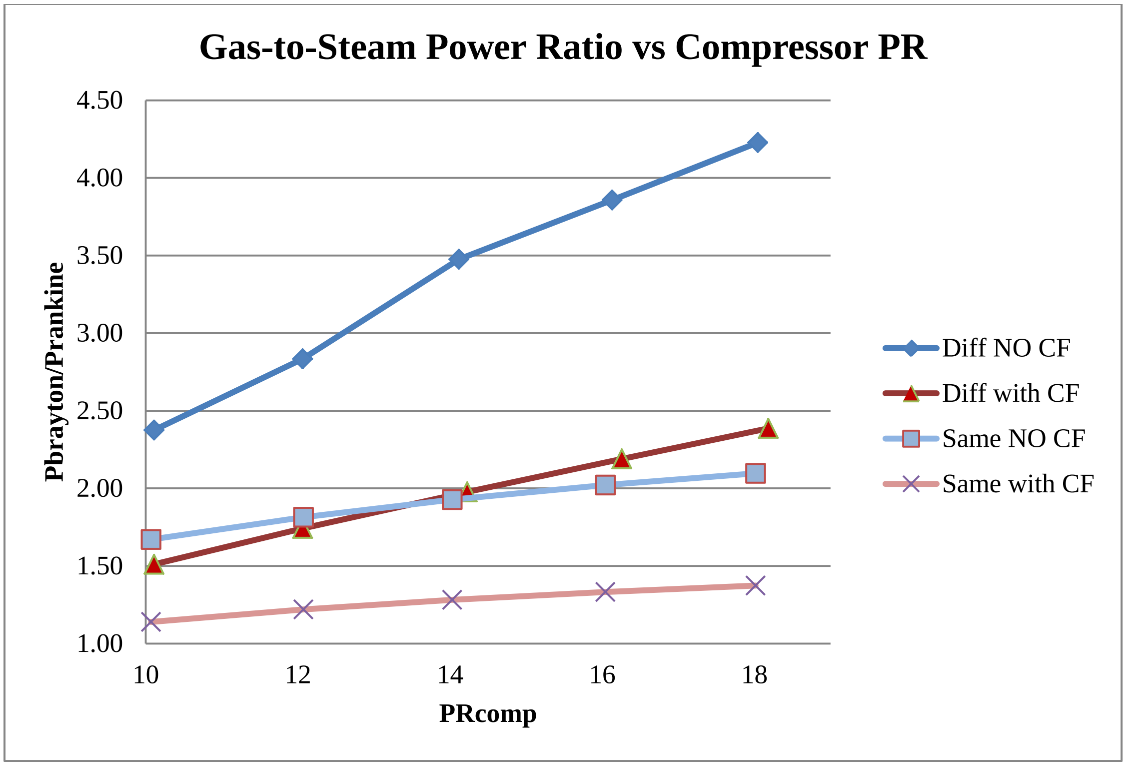

Figure D-3. Gas to Steam Power Ratio vs. Compressor PR

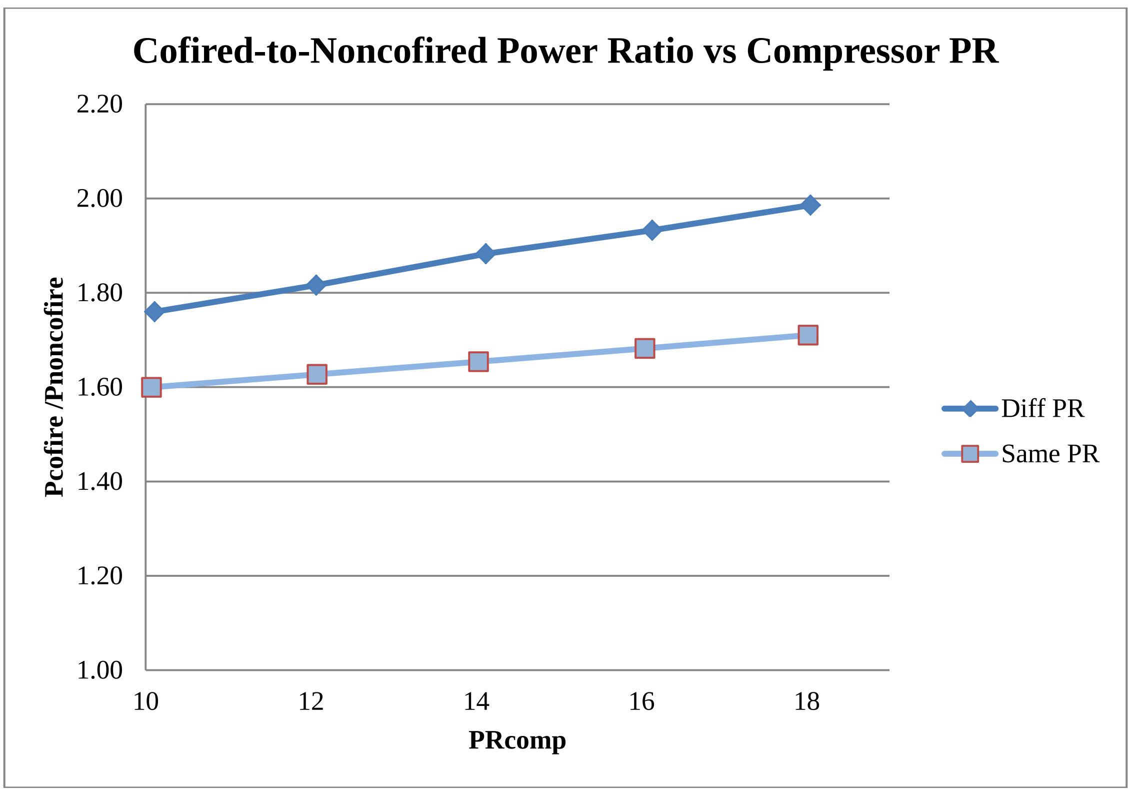

Figure D-4. Cofired to Noncofired Power Ratio vs. Compressor PR

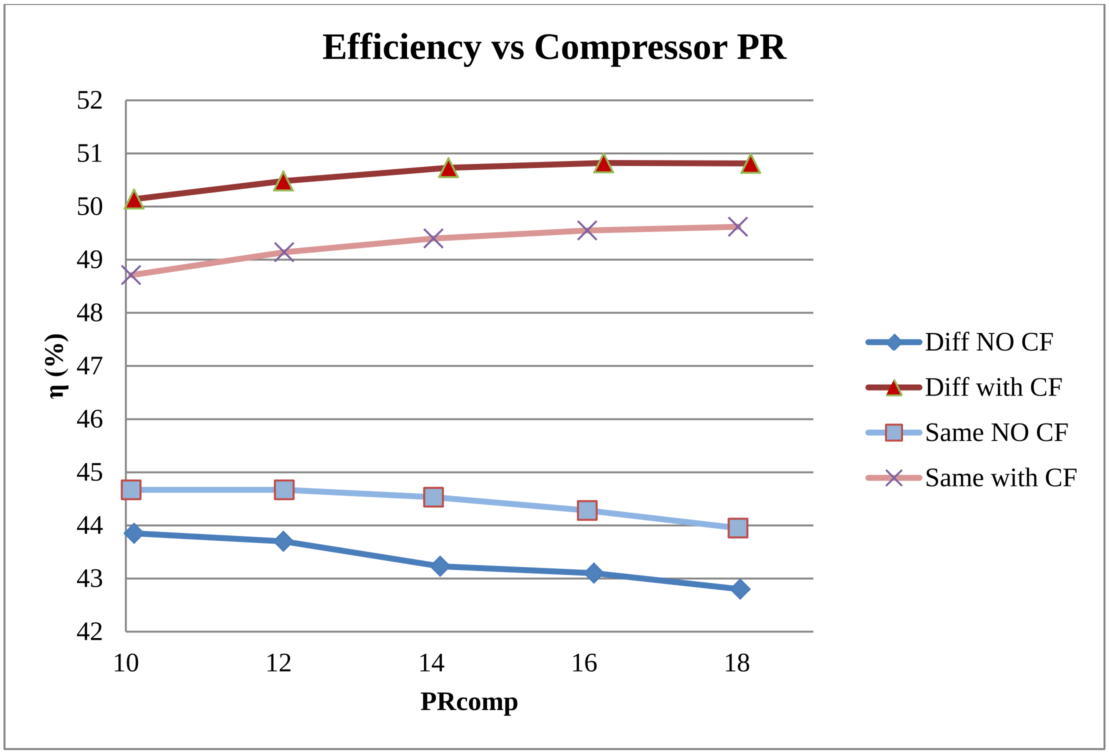

Figure D-5. Efficiency vs. Compressor PR

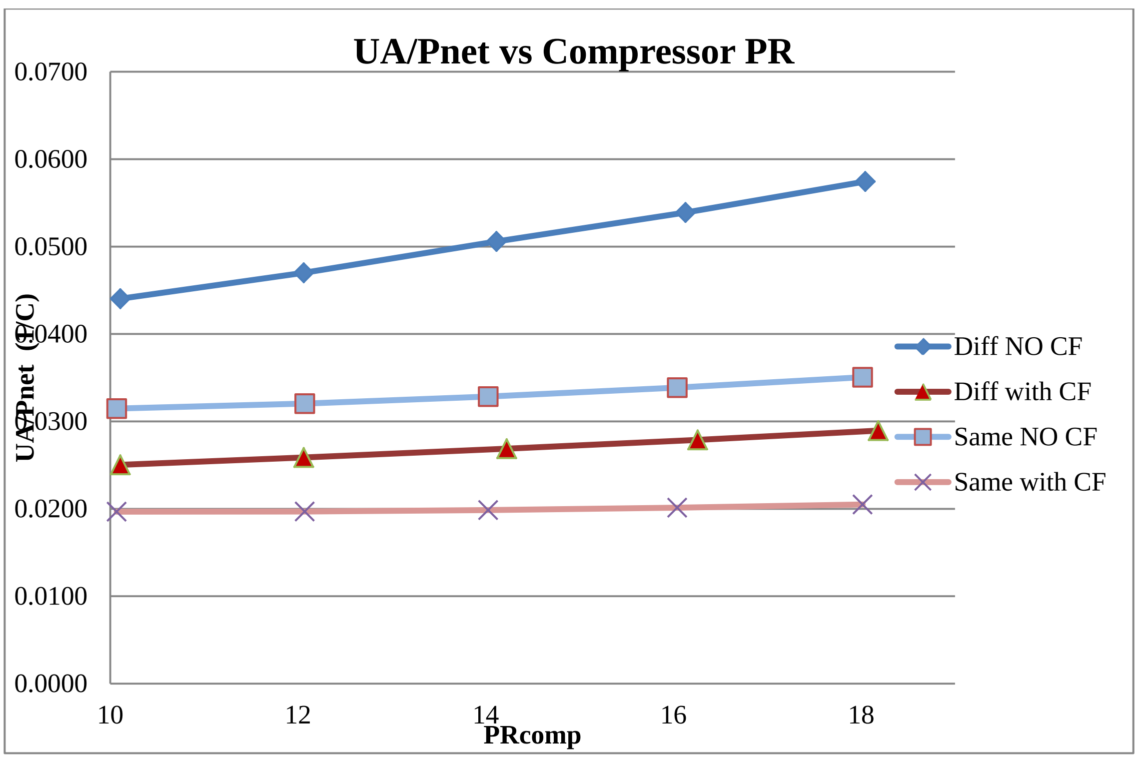

Figure D-6. UA/Power vs. Compressor PR

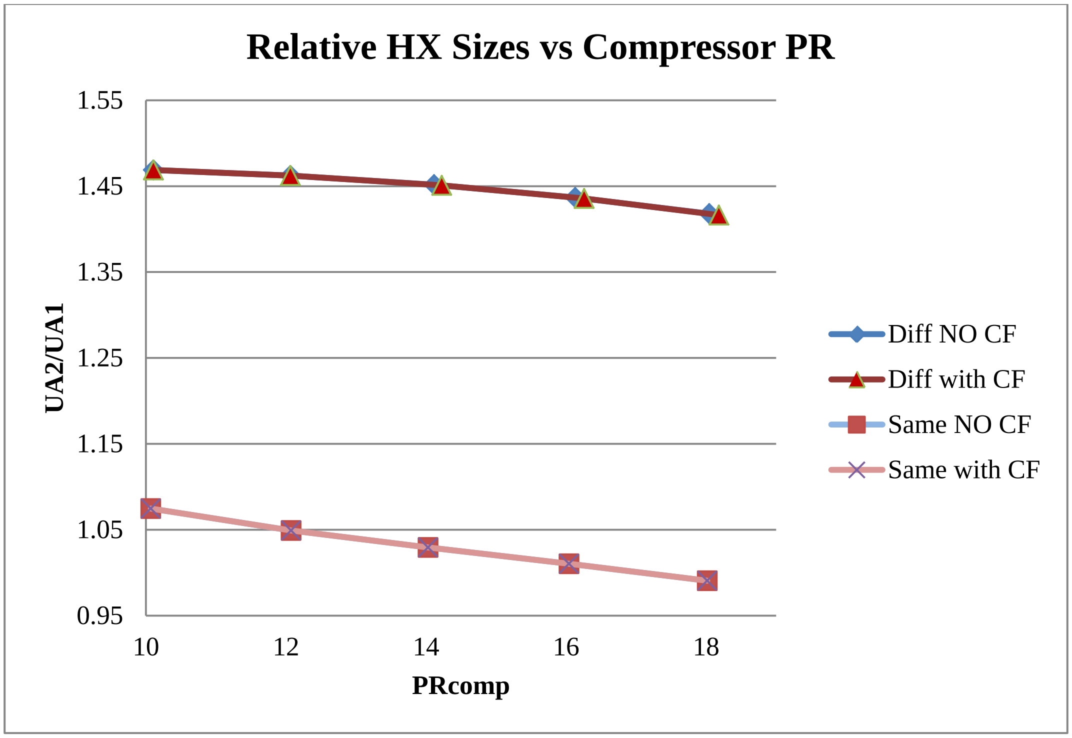

Figure D-7. Relative HX Sizes vs. Compressor PR

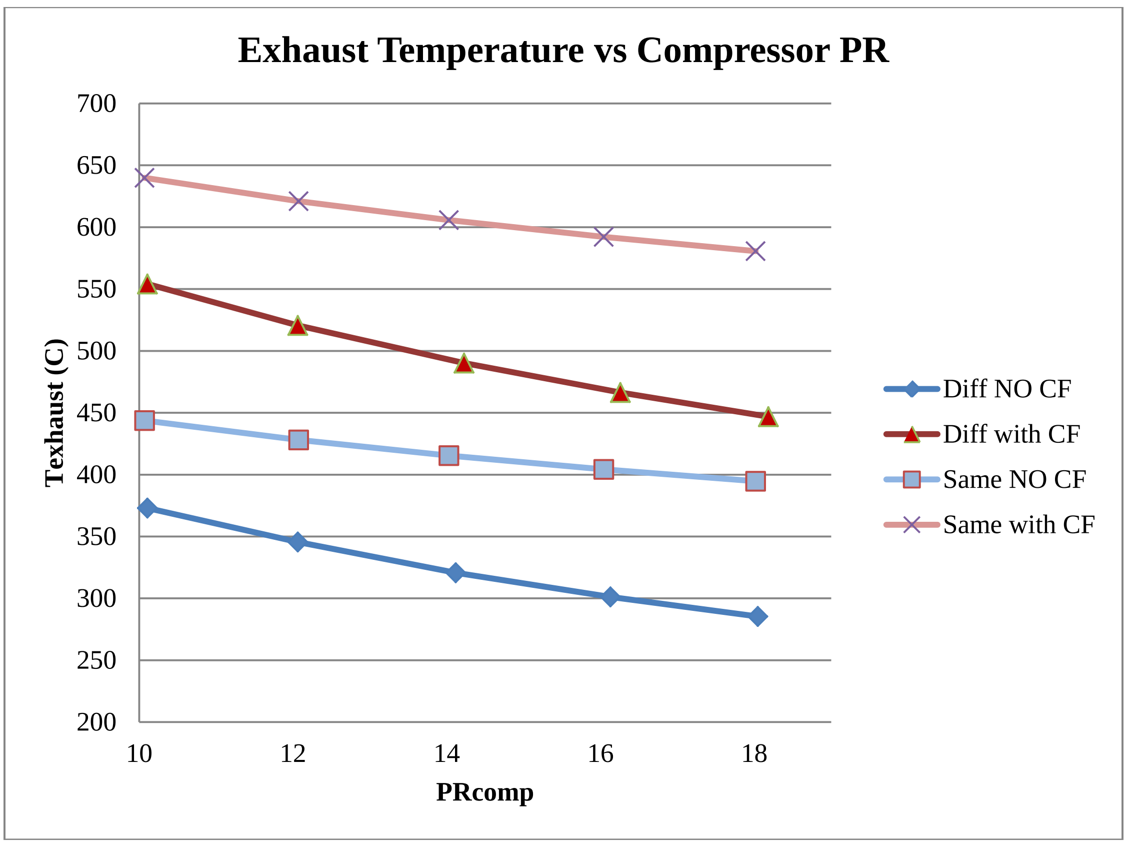

Figure D-8. Turbine Exhaust Temperature vs. Compressor PR

Appendix E ALWR Utility Requirement to FHR Design

E.1 Safety and Investment Protection

Requirement	Applicability to FHRs
Accident Resistance	
Simplification	Applies directly, however this is challenging to quantify.
Fuel design margin of 15 percent over and above regulatory fuel design requirements	Applies directly, however the fuel design metrics are different for the coated-particle high temperature fuel used in FHRs compared to the ceramic oxide fuel used in ALWRs.
Safe Shutdown Earthquake shall be 0.3g	Applies directly.
The reactor shall be designed so that the power reactivity coefficient is negative under all conditions	Applies directly.
For investment protection purposes, the operator shall have adequate time (30 minutes or more after indication of the need for action) to act to prevent damage to equipment or to prevent plant conditions, which could result in significant outages.	Applies directly.
Core Damage Prevention	
The ALWR shall meet applicable NRC requirements with regard to engineered safety system design and analysis of plant and engineered safety system response to regulatory specified transients and accidents	Applies directly.

Requirement	Applicability to FHRs
For investment protection purposes, the ALWR design shall be such that no fuel damage (i.e. the core can be used for further power operation) is predicted to occur for a postulated near instantaneous reactor cooling system break of up to six inches. Consistent with Safety Margin Basis evaluation, this analysis shall use best-estimate methodology to calculate core temperature and resulting effects.	This initiating event does not apply directly to FHRs directly. However, to apply this requirement to FHRs one must consider any initiating events will the similar or more frequent occurrences in FHRs as instantaneous reactor cooling system break of up to six inches. An FHR must be able to resume further power operation after this set of initiating events.
The role of the operator in the ALWR shall be that of an intelligent overseer in the event of off-normal conditions. The plant shall be designed to allow the operator significant time to evaluate the plant condition and decide what, if any, manual action is needed. The plant shall, however, be designed so as to prevent operator override of safety system functions as long as valid system actuation signal exists	Applies directly.
The mean annular core damage frequency for the design shall be evaluated using PRA and it shall be confirmed by the Plant Designer that this frequency is less than $1x10^{-5}$ events per reactor year, including both internal and external events.	Applies directly.
Mitigation	
A large, rugged containment building and associated containment system shall be proved for heat removal and retention of fission products for licensing design basis events. Containment design pressure shall be based on the most limiting loss of coolant or steam line break accident.	
Licensing Design Basis source term analyses shall be more realistic than the TID 14844, Regulatory Guide approach for current LWRs.	Applies directly.
The ALWR design shall allow siting at most sites available in the United States.	Applies directly.

Requirement	Applicability to FHRs
Containment system components for which a change of state is necessary to assure an intact containment (e.g., containment isolation valves, capacity/lower drywell flooder valves) shall be redundant and shall be sufficiently independent form the systems whose failure could lead to core damage so as to avoid significant vulnerability to common cause failure	Applies directly.
Severe accident risk shall be evaluated using PRA and it shall be confirmed by the Plant Design that the whole body dose at the site boundary (approximately 0.5 miles from any individual reactor) is less than 25 rem for releases form sever accidents, the cumulative frequency of which exceeds $1x10^{-6}$ per reactor year.	Applies directly.
Requirements for Passive Plants	
Engineered safety systems necessary for the Licensing Design Basis shall utilize passive means for water injection, cooling and other functions. Passive means are natural forces such as gravity and natural circulation, stored energy such as batteries and compressed fluids, check valves, and non-cycling powered valves. The design shall not rely on features such as multiple acting valves, and ac powered divisions and continuously rotating machinery, other than inverter supplied components, to prevent or mitigate LDB events.	Applies directly. However, it should be noted that the working fluid for the DRACS loop in FHRs is a fluoride salt rather than water.
The passive plant designer shall not require safety-related ac electric power other than inverter supplied ac power for instrumentation and control functions.	Applies directly.

Requirement	Applicability to FHRs
For investment protection, the Passive ALWR shall have a low likelihood of loss of all ac power. In addition to power from the main generator and from the normal tie line to the plant switchyard, the plant shall have at least two non-safety-related ac power sources (not including inverter supplies). At least one of these sources shall be an on-site power generator.	Applies directly.
The Passive ALWR design shall provide a greatly increased time for operator response. For transients and accidents analyzed under the initiating event plus single failure Licensing Design Basiss assumptions (which include loss of all ac power), no credit for manual operator actions shall be necessary to meet core protection regulatory limits for at least 72 hours following initial indication of the need for action (i.e. approximately the time of the initiating event).	Applies directly.
Only simple operator actions (e.g., few in number, unhurried, dependent on straightforward diagnostics, requiring common operator skills) and minimal off-site assistance (e.g., commercial supplies and components which are readily available, easily transported, and easily installed, such as a portable ac generator with its fuel and connection cables) shall be necessary beyond 72 hours to prevent core damage for the transient and accidents noted above.	Applies directly.

Requirement	Applicability to FHRs
Containment performance for the sequences surviving the severe accident selection process shall assure containment leak tightness sufficient to meet off-site dose limits, for at least 72 hours without the need for off-site assistance. Beyond 72 hours, only minimal off-site assistance shall be necessary to maintain required containment leak tightness.	Applies directly. FHRs rely on their coated particle fuel layers and the sorption properties of the liquid fluoride salt primary coolant. The effect of these addition layers of defense in depth need to be accurately assessed in the containment performance evaluation. Furthermore, the chemical state of the primary salt effects its sorption capability, therefore the functionality of the primary salt chemistry control system needs to be taken into consideration in this evaluation.
Permanent features shall be designed into the plant to facilitate connection and use of any portable equipment (e.g. ac generator) required for the off-site assistance referred to above, and to minimize radiation exposure from this connection and use.	Applies directly.
The plant shall be designed to provide a technical basis for simplification of plume exposure pathway-related off-site emergency planning. The intent is to retain an on-site emergency plan and certain elements of the off-site plan, but demonstrate that doses are low enough that early notification, evacuation planning of the public, and provisions for exercising the off-site plan are not necessary.	Applies directly.

E.2 Performance

Requirement	Applicability to FHRs
Plant Characteristics	
Evolutionary ALWR have power ratings of 1200-1300 MWe per unit whereas Passive Plants have power ratings close to 600 MWe	These power limits do not apply to FHRs because the economics of FHRs are inherently different that those in ALWRs.

Requirement	Applicability to FHRs
The plant shall be designed to operate for 60 years. Over this life span, components will need to be replaced, and special attention will need to be paid to material issues such as fatigue, corrosion, thermal aging and radiation embrittlement effects. Therefore, the design shall include features to permit necessary component replacement within the design availability requirements and shall include analyses and data necessary to support the design life of materials.	The plant lifetime applies directly to FHRs. However, there are different materials degradation issues in FHRs compared to ALWRs due to different sets of coolants and materials. The two biggest material issues that limit the life of the plant are the long-term thermal creep in the reactor vessel and the radiation damage to the outer graphite reflector.
The plant should be capable of operating on a fuel cycle, from post-refueling startup to the subsequent post-refueling startup, with a refueling interval of 24 months.	The shutdown frequency in FHRs is not limited by reactivity because the baseline design implements online refueling. Rather, the shutdown period is limited by radiation damage to the internal graphite structure and metallic pebble separator.
BWR fuel mechanical design shall be capable of peak bundle-average burnup of at least 50,000 MWD/MTU. For PWRs, fuel mechanical design shall be capable of assembly-average burnups of at least 60,000 MWD/MTU.	The burnup limits in FHRs will be determined based on the fuel performance, and the trade off between cost of fuel fabrication and enrichment of LEU of the coated-particle high temperature fuel. Based on the current best-estimates economic models for coated-particle fuel fabrication, the fuel design optimizes to fuel designs with maximized average discharge burnup.
The premature fuel failure rate due to manufacturing defects shall be less than one in 50,000 fuel rods.	The premature fuel failure rates shall be calculated to limit the dose at the reactor sight boundary to below the regulatory limit. Fission product transport from failed fuel out of the reactor needs to be studied to determine an appropriate premature fuel failure rate limit.

Requirement	Applicability to FHRs
The radioactive waste and water treatment systems and plant shielding design basis shall use a failed fuel rate consistent with regulatory requirements. For purposes of normal operation performance evaluation, 0.025% failed fuel for PWRs and a noble gas release rate of 15000 μCi/second at 30 minutes BWRs shall be utilized.	This requirement will likely remain, however new FHR design basis source terms need to be developed based around the yet to be determined premature fuel failure rate.
The ALWR shall be designed and constructed so that the amount of radioactive gaseous, liquid and solid waste released from the plant shall be equal to or better than comparable values for the 10% best plants of the same type (i.e. BWR or PWR) currently operating in the U.S. Furthermore, the ALWR shall provide on-site storage capacity for a minimum of six months radioactive solid waste accumulated during a period of maximum generation rate.	Applies directly. Special care should be taken due to the production of tritium during normal operation and its high temperature diffusion through metals.
Wet storage capacity for spent fuel resulting from ten years of operation plus one core off-load of fuel shall be provided. In addition, on-site land shall be reserved to permit the construction of a dry storage system with capacity to store all of the fuel discharged over the plant design life.	Applies directly.
The ALWR shall be designed and constructed so that occupational radiation exposure can be less than 100 person-rem/year averaged over the operation life of the plant.	Applies directly. In addition to the occupational radiation exposure, FHRs must also limit the occupational beryllium exposure within a regulatory limit. Additional experience handling beryllium salt in experiment loops and test reactors will likely inform the establishment of this limit.
Maneuvering and Transient Response Requirements	

Requirement	Applicability to FHRs
The plant shall be designed to be capable of startup from cold shutdown to hot standby at full pressure and temperature in 24 hours. Similarly, it shall be capable of cooling down from reactor critical at full temperature and pressure to start of refueling operations in 24 hours.	The startup and shutdown times will likely translate to FHRs. However, the “cold-shutdown” condition in FHRs is defined as a temperature greater than or equal to the melting point of the primary coolant.
The plant shall be designed for a 24-hour load cycle with the following profile: starting at 100 percent power, power ramps down to 50% in two hours, power remains at two to ten hours, and then ramps up to 100 percent in two hours. Power remains at 100 percent for the remainder of the 24-hour cycle. The plant shall be designed to permit this cyclic load following for 90 percent of the days of each fuel cycle for the life of the plant.	Applies directly – the FHR is likely to feature power peaking from a natural gas heater.
Reliability and Availability	
The plant shall be designed for an annual availability of greater than 87 percent over the life of the plant.	Applies directly.
The plant shall be designed to achieve the following outage durations: • Planned Outages: less than 25 days/year • Forced Outages: less than 5 days/year • Major Outages: less than 180 days/ 10 years	Applies directly.
The plant shall be designed so that a refueling outage free from major problems can be conducted in 17 days or less (break to breaker) assuming 24-hour productive days.	Outage time applies directly. However, the outage will revolve around performing periodic maintenance and replacing components due to high temperature-, chemical- and radiation damage- degradation.
The plant shall be designed to limit the number of unplanned automatic trips to be less than one per year.	Applies directly.

Requirement	Applicability to FHRs
Non-safety related active RCS makeup capability and any other necessary measures shall be provided in the Passive ALWR such that RCS depressurization is not required for RCS breaks up to a size equivalent to 3/8-inch diameter.	This requirement is really for the performance of the ALWR to a 3/8-inch diameter break. The integral design of the baseline PB-FHR reduces the consequences for small breaks because the coolant must remain in the primary circuit. The FHR requirement should relate the effort to recover back to a safe power producing state from any initiating event with frequency similar to 3/8-in break in the RCS in ALWRs.
Recovery from inadvertent RCS depressurization in the Passive ALWR shall be rapid enough that lifetime-average design availability requirements can still be met assuming one inadvertent RCS depressurization during the 60-year plant life. Specifically, design features shall be provided to permit recovery from an inadvertent RCS depressurization within 30 days and this outage shall be included in the lifetime-average availability.	This requirement does not apply to FHRs directly because they have different transient sequences. However, the idea that recovery from an inadvertent transient (LOFC, LOHS, unprotected LOFC or unprotected LOHS) with similar frequency to an inadvertent RCS depressurization shall be fast enough to meet the lifetime availability requirements presented earlier.
Where feasible, Passive ALWR systems and equipment shall be designed to withstand a complete loss of ac power (other than inverter supplied power) for at least two hours without exceeding equipment design limits. Where it is not feasible to provide this protection, the design shall be such as to allow repair or replacement of the damaged equipment within 24 hours after power restoration	Applies directly.
Operability, Maintainability and Testing	
Ease of operation shall be designed into the ALWR through such features as use of modern digital technology for monitoring, control, and protection functions, a forgiving plant response to upset conditions, design margins, and consideration of the environment in which the operator must perform.	Applies directly.

Requirement	**Applicability to FHRs**
The design shall incorporate the results of systemic identification and resolution of operational and maintenance problems which exist in current plants.	Applies directly.
Consistent with overall simplification, the number of different types of equipment which must be specified and maintained, i.e., valves, pumps, instruments, and electrical equipment, shall be minimized by standardization except in those limited applications where diversification is adopted by the designer as an appropriate means to protect against common cause failure.	Applies directly.
Equipment shall be designed to have minimal, simple maintenance needs, and be designed to facilitate maintenance.	Applies directly.
The layout of systems shall consider the maintenance needs for access, pull space, laydown space and heavy lifts.	Applies directly.
The plant shall be designed so that the environment under which the maintenance and testing of equipment must be performed provides satisfactory working conditions, including temperature, dose, ventilation, and illumination.	Applies directly. Special consideration must be considered given that most of the components operate at elevated temperatures and workers need to be protected from tritium and beryllium exposure in addition to standard radiation safety procedures in ALWRs.
The surveillance tests shall be designed to measure simply and directly the system design basis performance parameters, preferably with the plant at power in order to avoid adding tasks to the planned outage time. Mechanical and electrical systems shall be designed to avoid plant trips and plant equipment and layout shall be designed to facilitate and simplify surveillance testing. The allowable interval between tasks should be increased where justified. Where surveillance tasks must be performed during an outage, the design should assure that the tests will not be critical path for the outage.	Applies directly.

Requirement	Applicability to FHRs
The protection system and control systems for the engineered safety system shall be designed so that: (a) the plant can be safely operated indefinitely at full power with one protection channel in test or bypassed (because of failure or other reasons), (b) one subsequent single failure will not cause a plant trip.	Applies directly.
The M-MIS shall be such that testing and maintenance is greatly simplified with respect to current plants. For example, self-testing shall be included and the testing automated to the degree practical.	Applies directly.
Man-Machine Interface System	
The M-MIS shall employ modern digital technology, including multiplexing and fiber optics, for monitoring, control, and protection functions. Multiplexing is to be used for any function, including safety functions, where it is appropriate and reduces the cost and complexity of cable runs throughout the plant.	Applies directly.
Existing regulatory requirements enforce segmentation and separation of safety and protection systems. In addition, for the major plant control and monitoring functions, the M-MIS shall incorporate segmentation of major functions, separation of redundant equipment within a segment, and fault tolerant equipment to achieve high reliability and prevent propagation of a fault between redundant equipment and form one segment to another. The M-MIS shall assure “graceful” failure which allows continued plant operation to the extent practical.	Applies directly.

Requirement	Applicability to FHRs
The M-MIS design process shall be fully integrated with the remainder of the ALWR plant design. The design process shall provide for iteration among the M-MIS and plant designers and shall use mockups, dynamic simulation, and operations and maintenance personnel input in the M-MIS design.	Applies directly.
The main control room shall be designed on the basis of a specified number of operators (two or three) being available for operation of the plant in all modes of operation. Adequate space and layout shall be available for up to 10 occupants on a temporary basis the design is to be such that a single operator can adequately control the plant during normal power operations.	Applies directly.
The main control room shall contain compact, redundant, operator workstations with multiple display and control devices that provide organized, hierarchical access to alarms, displays, and controls. Each workstation shall have the full capability to perform main control room functions as well as support division of operator responsibilities. A supervisor's workstation shall also be located in the main control room.	Applies directly – although, this requirement will be updated based on developments on staffing for multi-module SMRs. Currently, the NRC recommends using exemption requests to address staffing issues as an intermediate solution. Then implement regulatory changes based on experience with SMRs and staffing changes (Johnson 2011).
The main control room shall incorporate modern, computer-driven displays to provide enhanced trending information, validated data, and alarm prioritization and supervision, as well as diagrammatic normal, abnormal, and emergency operating procedures with embedded dynamic indication and alarm information. In addition, extensive use of data management and computer-aided design (CAD) techniques shall be made to display plant information at appropriate levels of detail with updated equipment status indication.	Applies directly.

Requirement	Applicability to FHRs
The main control room shall include large, upright, spatially dedicated panels which provide an integrated plant mimic, indicating equipment status, plant parameters, and high level alarms.	Applies directly
Local and stand-alone control systems shall be designed in the same rigorous way as the main control stations and will use consistent labeling, nomenclature, etc. Particular attention is to be paid to visibility, color coding, use of mimics, access to lighting, and communication.	Applies directly.
An integrated, plant wide communication system shall be provided for construction and operations.	Applies directly.
The Passive ALWR design shall be such that the main control room shall be available for post-accident monitoring for all Licensing Design Basis accident and transients (except for events requiring main control room evacuation, e.g. control room fire), including loss of all AC power, for 72 hours without the need for off-site assistance. Beyond 72 hours, reasonable off-site assistance as defined in Section 3.1.3 may be utilized.	Applies directly.

E.3 Constructability

Requirement	Applicability to FHRs
Construction Duration and Design Completion	
The passive plant (600MWe) shall be designed for construction in 36 months from the first structural concrete placement milestone to fuel load. Allowing 6 months for plant startup and low power testing and 18 months as representative of the duration necessary to prepare the site and complete major excavation work, the planning base is for an overall duration of 60 months from owner commitment to construct to commercial operation.	The PB-FHR will have a more aggressive construction schedule given the reduced power rating, smaller footprint and smaller Brayton cycle power conversion system.
Construction and Design Coordination	
Plant Constructor personnel shall participate in the ALWR design process to assure that constructability requirements are adequately implemented.	Applies directly.
Design provisions to simplify and facilitate construction and startup shall be explicitly considered in the design process. Such provisions include good crane and material handling access, adequate space and access for construction activities, and provision for temporary construction buildings and equipment.	Applies directly.
Standardized component sizes, types, and installation details shall be provided to improve productivity and reduce material inventories.	Applies directly.
Reasonable construction tolerances shall be specified to minimize unnecessary re-work and improve productivity.	Applies directly.
An experience review of previous LWR construction problems shall be performed to assure lessons learned are addressed in ALWR design and construction.	Applies direction. Additionally, previous construction of MSR, HTGR and SFR shall be reviewed.

Requirement	Applicability to FHRs
Advanced Construction Technology	
Extensive use of multiplexing for the instrument and control systems to reduce electrical raceways and cable pulling.	Applies directly. (don't know what this means)
Designs which permit construction craftwork to be performed at "out of hole" locations so that large fabrications of material and equipment are assembled and installed in the final location using heavy load capacity cranes, thereby reducing congestion in the installation locations.	Applies directly.
Modularization of equipment packages and structural elements to take advantage of improved productivity by reducing congestion and reduced costs to field versus shop labor. Modularization shall be accomplished while preserving space needed for maintainability, testing, and other access related requirements. More extensive use of modularization of structural and equipment packages was expected to be necessary in the Passive ALWR in order to achieve the very ambitious 36-month construction schedule.	Applies directly.
Integrated Construction Planning and Scheduling	
A detailed living construction plan shall be jointly developed prior to start of construction by the Plant Designer, Constructor, and Startup Test organizations, utilizing input from principle suppliers and subcontractors. The plan shall establish the overall approach and provide a basis for developing and assessing schedules.	Applies directly.

Requirement	Applicability to FHRs
Detailed schedules shall also be developed prior to start of construction to integrate the design, procurement, construction, and startup testing activities up to Plant Owner acceptance. The startup testing requirements shall establish the logic for system turnover sequence and schedule including requirements necessary for defining system boundaries, establishing system numbering, and assuring timely turnover.	Applies directly.
Monitoring of the construction progress shall be accomplished using quantitative methods appropriate to the particular activity, e.g., number of welds, feet of cable pulls, to make up-to-date assessments of progress and anticipate where deviations from schedules may occur in time to take appropriate action to resolve problems and maintain schedule milestones. The schedules shall be updated as work progresses to realistically reflect the actual work status.	Applies directly.
Inspections, Tests, and Analyses for Assuring Construction Adequacy	
The NRC Standardization Rule, 10CFR52, requires that the tests, inspections, and analyses, performed to provide reasonable assurance that the plant is properly constructed, shall be identified in the combined license. Accordingly, the Plant Designer shall prepare a set of tests, inspections, and analyses and associated acceptance criteria which will demonstrate that the plant has been constructed and will be operated in conformity with Commission regulations, the combined license, and the Atomic Energy Act. The technical basis for the completeness of the set of inspections, tests and analyses and for the specified acceptance criteria shall be provided. The nature and level of detail of acceptance criteria shall be such as to allow third party (i.e., the NRC staff) verification that the acceptance criteria have been met.	Applies directly.

E.4 Design Process

Requirement	Applicability to FHRs
Design Integration	
The design process is to be managed and executed as a single integrated process. Therefore, the requirements have been addressed to the "Plant Designer" even though the effort may involve more than one organization (e.g. an Architect Engineer, an NSSS supplier, and a constructor)	Applies directly.
The Plant Designer shall prepare design basis documents for each plant system or element which describe specific design criteria, the design features, and how these features satisfy the criteria. The documents shall be sufficiently complete that an acceptable design can be developed and that the potential acceptability and conformance to ALWR requirements can be judged.	Applies directly.
Interdisciplinary design reviews shall be conducted throughout the design and construction process. These reviews shall include confirmation that the utility simplification policy is being emphasized in the design and that all simplification requirements are being addressed.	Applies directly.
The Plant Designer shall utilize verified and validated computer models of the plant, and a control room simulator as design tools in studying plant response, defining human-engineering aspects of the plant controls and control room design, and developing plant operating procedures. The verification and validation should be documented.	Applies directly – though validation in an FHR will be more challenging than an LWR given there is no direct experience base with FHRs. The validation base will be developed from experimental program for the Thorium Molten Salt Reactor project lead by the Shanghai Institute of Applied Physics and salt loops at ORNL and UW-Madison.
Configuration Management	

Requirement	Applicability to FHRs
The configuration management program shall include the following features: • Methods for controlling and providing accessibility to design basis documents. • Verification methods to insure compliance of the hardware and software design at all levels with the design basis documents. • Change control methods to assure that all changes from the original designs are approved at the appropriate level of authority in the design and plant owner organization and documented for the life of the plant. • A process to assure verification and auditing of program data gathering, updating, revising, dissemination and security. • Auditing and checking of the configuration management programs and data on a regular basis.	Applies directly.
Information Management System	

Requirement	Applicability to FHRs
The main objectives of the IMS are as follows: • To provide a logical breakdown of the ALWR into a number of systems and system groups and to use standard identification for all systems, components, facilities, and documentation which can be used for design, construction, and operation; • To make effective utilization of computer aided design and engineering during design and construction, and after the plant is turned over to the operator; • To provide for efficient implementation of a project information network which utilizes a methodology such as that described in EPRI NP-5159, Guidelines for Specificying Integrated Computer-Aided Engineering (CAE) Applications for Electric Power Plants; • To provide an effective means to acquire, store, retrieve and manipulate the documents and data necessary to design, construct, startup, operate and maintain the plant; and • To assure that information needed for construction and operations is available when the plant is turned over to the owner.	Applies directly
Engineering Verification of As-built Conditions	
Engineering verification activities shall be identified early in the construction and scheduled so that completed walkdowns and evaluations, as well as any necessary rework, support project completion milestones.	Applies directly
Engineering verification activities shall include seismic walkdown to verify all key seismic PRA assumptions such as equipment anchorages and system interactions	Applies directly

Requirement	Applicability to FHRs
To the extent practical, the design shall include provisions which minimize the complexity and scope of engineering verification walkdowns during construction. Where verification is necessary, the Plant Designer shall develop procedures, including walkdown objectives and scope, process for evaluation, and process for resolution of items which do not meet the design intent. Sampling techniques shall be used in preference to inspections of the total population in question.	Applies directly