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DESIGN OF THE MARK-I PEBBLE-BED, FLUORIDE-SALT-COOLED, HIGH-TEMPERATURE REACTOR COMMERCIAL POWER PLANT

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Abstract *– The University of California, Berkeley (UCB) is developing a pre-conceptual design for a commercial pebble-bed, fluoride-salt-cooled, high-temperature reactor (PB-FHR). The baseline design for this Mark-I PB-FHR plant (Mk1) is a 236-MWth reactor. The Mk1 uses a fluoride salt coolant with solid, coated-particle pebble fuel. The Mk1 design differs from earlier FHR designs because it uses a nuclear air-Brayton combined cycle designed to produce 100 MWe of base-load electricity using a modified General Electric 7FB gas turbine. For peak electricity generation, the Mk1 has the ability to boost power output up to 242 MWe using natural gas co-firing. The Mk1 uses direct heating of the power conversion fluid (air) with the primary coolant salt rather than using an intermediate coolant loop. By combining results from computational neutronics, thermal hydraulics, and pebble dynamics, UCB has developed a detailed design of the annular core and other key functional features. Both an active normal shutdown cooling system and a passive, natural-circulation-driven emergency decay heat removal system are included. Computational models of the FHR – validated using experimental data from the literature and from scaled thermal hydraulic facilities – have led to a set of design criteria and system requirements for the Mk1 to operate safely and reliably. Three-dimensional, computer-aided-design (CAD) models derived from the Mk1 design criteria are presented, and the design process behind the CAD models is discussed.*

I. INTRODUCTION

This paper describes work performed at the University of California, Berkeley (UCB) to develop an initial pre-conceptual design for a small, modular 236-MWth pebble-bed fluoride-salt-cooled, high-temperature reactor (PB-FHR), as an element of a larger Department of Energy Integrated Research Project (IRP) collaboration with Massachusetts Institute of Technology and University of Wisconsin, Madison to establish the technical basis to design, license, and commercially deploy FHRs. The key novel feature of this Mark-I (Mk1) PB-FHR design compared to previous FHR designs is the use of a nuclear air-Brayton combined cycle (NACC) based upon a modified General Electric (GE) 7FB gas turbine (GT). This combination is designed to produce 100 MWe of base-load electricity when operated with only nuclear heat and to increase the power output to 242 MWe by injecting natural gas (co-firing) for peak electricity generation.

The primary purpose of the Mk1 design, with its co-firing capability, is to change the value proposition for nuclear power. The new value proposition arises from additional revenues earned by providing flexible grid support services in addition to base-load electrical power generation.

A summary of work contributing to the design – and the design itself – are provided below.

II. MK1 PB-FHR BASICS

FHRs combine several technologies from other reactor types. Key Mk1 operating parameters and design basics are as follows:

- Graphite pebble fuel compacts with coated-particle fuel
- Flibe ($^7Li_2BeF_4$) molten salt coolant
- Inlet/outlet temperatures of 600/700°C
- Pool-type reactor at near atmospheric pressures

The Mk1 is designed so that all components – including the reactor vessel, GT, and building structural modules – can be transported by rail, enabling modular construction. With these modules being fabricated in factories using computer-aided manufacturing methods, the assembly of a Mk1 at a reactor site will more closely resemble three-dimensional (3D) printing than conventional nuclear construction. The design constraint of rail transport limits the width of all components, including the reactor vessel, to under 3.6 m, which in turn constrains the Mk1's thermal power. However, the constrained value matches well to the largest rail-shippable GTs now commercially available.

One key new characteristic of the Mk1 is that it eliminates the intermediate coolant loop used in all previous FHR designs and in all sodium-cooled fast reactors (SFRs) built to date, as shown in the flow schematic in Fig. 1. SFRs have used intermediate loops because sodium reacts energetically when contacted with water in a steam generator (as well as with air and carbon dioxide). However, the fluoride-salt coolant used in FHRs has high chemical stability [1].

For advanced reactors, the reactor vessel volume provides one metric for primary system cost. The Mk1 PB-FHR reactor vessel has a volumetric power density of 0.87 MWe/m^3. This is lower than typical pressurized water reactors (PWRs) (2.8 MWe/m^3), but is approximately 3 times larger than both the S-PRISM SFR (0.29 MWe/m^3) [2], which uses a low-pressure, pool-type vessel, and the Pebble Bed Modular Reactor (0.24 MWe/m^3) [3], which uses a high-pressure reactor vessel.

II.A. Nuclear Air-Brayton Combined Cycle

In the current fleet of nuclear power plants, designers try to maximize the thermal power of the reactor and subsequently couple an appropriate steam turbine/cycle. In the case of the NACC, the opposite applies; the GT capabilities determine the reactor thermal power. For the Mk1 baseline design of the NACC, a modified version of the GE 7FB GT is used. The GE 7FB is a rail shippable 60Hz machine, already widely deployed in the U.S. In its conventional, natural-gas-only configuration, it supplies 183 MWe in a simple cycle and 280 MWe in a combined cycle. The NACC is also a hybrid power conversion system that allows supplementary firing with fossil fuels (gas/liquid) above the nuclear base-load heat. This enables peak power production, as well as the ability to provide flexible capacity and several ancillary services to the grid. Among these are spinning reserve, black start services, peaking power, and frequency regulation. The performance of the Mk1 NACC design is summarized in TABLE I.

Another ability of the NACC is to decouple power conversion transients from the reactor due to the open cycle configuration. This reduces risks involved with loss of load events for example, as the reactor will not feel this transient.

The Mk1 PB-FHR uses two coiled tube air heaters (CTAHs) to transfer heat from the main salt to pressurized air. Due to the compact size of the Mk1 reactor vessel and main salt system, these CTAHs are located only 12.5 m from the centerline of the reactor vessel. Even though significant thermal expansion occurs when the reactor and main salt system are heated from their installation temperature to their normal operating temperature, the relatively short spacing between the reactor vessel and CTAHs allows this ~0.13-m expansion to be accommodated by placing the CTAHs on horizontal bearings and using bellows in the air ducts. This is similar to the approach taken to manage thermal expansion in the primary loop of conventional PWRs, where the steam generators are supported on vertical bearings and move horizontally in response to thermal expansion of the reactor hot and cold leg pipes.

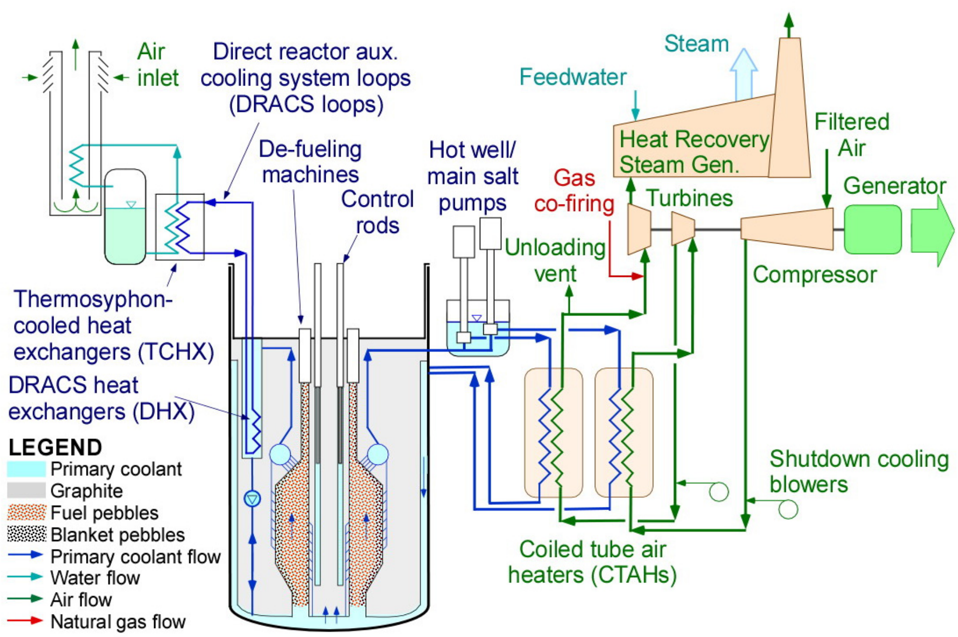

Fig. 1. Mk1 flow schematic.

TABLE I
Mk1 NACC operating parameters at ISO conditions

Parameter	Unit	Value
$P_{base\text{-}load}$ / $P_{co\text{-}fired}$	MWe	100/241
$P_{base\text{-}load}$ / $P_{co\text{-}fired}$	MWth	236/448
$\eta_{baseload}$	%	42.4
$\eta_{co\text{-}fired}$ (Net)	%	53.8
$\eta_{co\text{-}fired}$ (Gas only)	%	66.0

II.B. Normal and Safety Decay Heat Removal

The Mk1 PB-FHR uses the CTAHs for normal shutdown cooling and maintenance heat removal. For shutdown cooling, one or both main salt pumps are operated at low speed to circulate salt. A variable-speed blower system circulates ambient air through one or both of the CTAHs. The air flow rate is controlled to match the CTAH heat removal to the decay heat generation rate, and the salt flow rate is controlled to keep the salt cold leg temperature constant at 600°C to minimize thermal stresses to the reactor vessel and core internals. Because the two CTAHs can be drained independently, for maintenance, a single CTAH can be drained while the other CTAH continues to provide shutdown cooling.

The Direct Reactor Auxiliary Cooling System (DRACS) is a natural-circulation-driven decay heat removal system. It provides a diverse and redundant means to remove decay heat, in the event that the normal shutdown cooling system does not function. The DRACS transfers heat to ambient air, which serves as the ultimate heat sink for decay heat. The DRACS coolant loop uses natural circulation to transfer heat from the DRACS heat exchanger (DHX) to a thermosyphon-cooled heat exchanger (TCHX). Heat is removed by convection and thermal radiation from the tubes of the TCHX to water-filled thermosyphon tubes, where water boils and transports heat to a natural draft, air-cooled condenser. The DRACS coolant is flibe, to reduce the probability of the primary salt becoming contaminated with other salts due to heat exchanger leaks.

For emergency decay heat removal through the DRACS, natural circulation is established in the primary system, with flow upward through the core, then downward through the DHX and downcomer. During normal operation, the primary coolant flows in forced circulation upward through the core, and a small amount of coolant by-passes the core upward through the DHX and other core by-pass paths. A fluidic diode can provide high flow resistance for upward flow through the DHX during forced

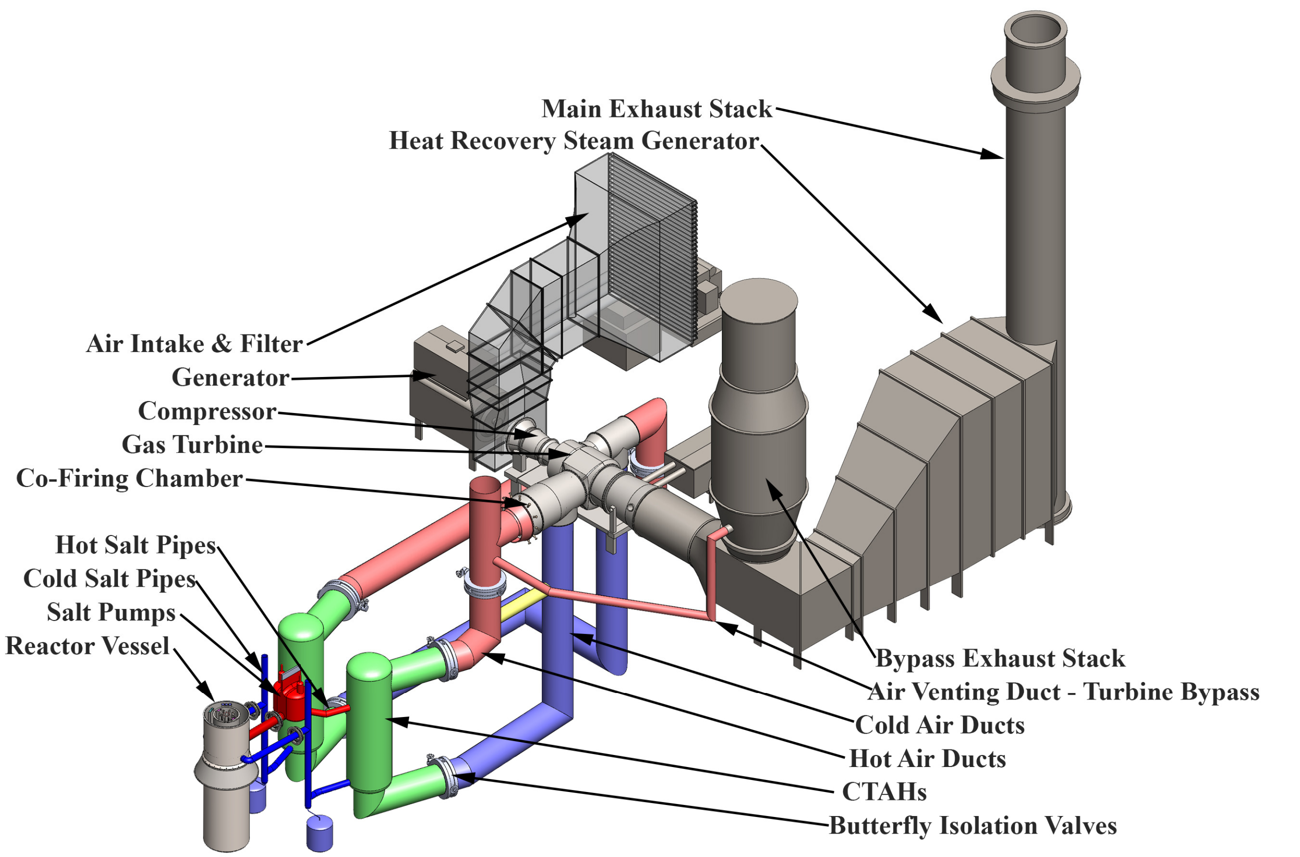

Fig. 2. Isometric view of the Mk1 plant's key components and systems.

convection, to limit parasitic heat losses, and low flow resistance for downward flow through the DHX during natural circulation. This function can also be served by a simple ball-type check valve, with a negatively-buoyant graphite and/or silicon carbide ball. A check valve could provide precise and predictable flow loss coefficients in both flow directions. Fig. 3 shows the coolant flow paths and by-pass flows during forced circulation and natural circulation operation.

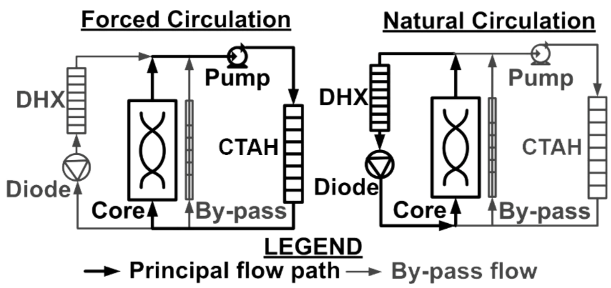

Fig. 3. FHR primary coolant flow paths for forced circulation and natural circulation operational modes.

Each DRACS loop is fabricated and mounted into a frame that can be lifted by crane. A key issue for detailed design of the DRACS is the containment penetration barrier needed for horizontal legs, and how to design hatches above the frame to allow loop installation and removal. The DRACS loop is not accessed under normal operation, so these hatches are designed to act as effective, passive missile barriers.

II.C. Balance of Plant

The Mk1 plant layout facilitates multi-module plant configurations. The configuration of the reactor and power conversion systems allows multiple PB-FHRs to be lined up in a row, and to have a clear boundary between the reactor and vital safety areas, and the balance of plant (BOP). The GT and associated equipment are configured to minimize the length of the air ducts and the associated pressure losses and circulating power, while maintaining a clear boundary between the reactor and the BOP.

Most BOP components are off-the-shelf and no modifications are needed to accommodate the NACC. One potential modification to BOP may be the natural gas supply system, which is a source of stored energy. Conventional natural gas safety standards need to be applied, as well as additional safety measures that will be implemented for the Mk1.

II.D. Plant Site Layout

Fig. 2 provides an isometric view of a notional Mk1 unit arrangement. This 3D computer-aided-design model was generated using input from all of the research described hereafter in this paper as well as expert input collected during four workshops hosted as part of the FHR-IRP. Supporting systems – such as those for fuel handling, control rooms, and other auxiliary services – still need to be added to complete the model.

III. NEUTRONICS

The pebble-filled reactor core is annular, with an inner radius of 0.35 m and an outer radius of 1.25 m. Coolant flows upward and radially outward through the core, injected at the bottom and from the center graphite reflector cylinder. At the outer radius the core is surrounded by graphite reflector blocks. The lowermost and uppermost regions of the annular core are tapered chutes for fueling and defueling, respectively. Fig. 4 presents a cross section of the reactor showing the core geometry.

There are two pebble regions within the pebble bed. The inner region, from radius 0.35 m to 1.05 m, contains fuel pebbles. The outer region, from radius 1.05 m to 1.25 m, contains a reflector of inert graphite pebbles. The primary purpose of the graphite pebble reflector is to attenuate the fast-neutron flux at the outer solid graphite reflector so as to extend the outer reflector lifetime to the full plant lifetime. The resulting design has an active core volume of 10.4 m^3 and a graphite pebble volume of 4.8 m^3.

The annular geometry of the Mk1 pebble design, shown in Fig. 5, reduces the peak and average fuel temperature of the pebbles, thus increasing the safety margin for transient accident behavior. Also, the annular design allows control of pebble buoyancy in the liquid salt coolant by adjusting the density of the central graphite core in the pebble. This design has a 1.5-mm-thick annular shell containing, on average, 4,370 tristructural-isotropic particles. This shell surrounds a 12.5-mm-radius inert graphite core. A 1.0-mm-thick, high-density graphite protective layer encapsulates the entire fuel pebble.

III.A. Fuel Management

In the Mk1 design, fuel pebbles are continuously circulated through the core at a slow pace. The pebbles are introduced into the bottom of the pebble bed and rise up as pebbles are removed from the top of the bed at an approximate rate of 0.2 Hz. Fuel pebbles are introduced through four inner pebble injection channels and blanket pebbles are introduced through four outer channels. Pebbles rely on their positive buoyancy in the coolant salt to move upward through the core and move in plug-flow through the active region. Pebbles are removed at the top of the core though an annular slot that converges into two defueling machines. Pebbles are recirculated through the core approximately eight times before reaching their discharge burnup. With an average resonance time of 2.1 months, each pebble is expected to spend 1.4 years in core.

III.B. Neutronics Modeling and Results

Current analyses of neutronic performance of the FHR have relied on a combination of MCNP5 and ORIGEN modeling. A suite of Python-based tools was developed to manage an iterative search for equilibrium core composition accurately accounting for the complex core and pebble geometries. The suite of tools developed for the FHR core design include Burnup Equilibrium Analysis Utility (BEAU), FHR Input-deck Maker for Parametric Studies (FIMPS), and mocup.py [4].

Using 19.9% enriched uranium fuel, the attainable discharge burnup from the equilibrium core was calculated to be 180 GWd/tU and the corresponding pebble residence time is 1.4 effective full power years (EFPY). The peak power density is 80 W/cm^3 while the bed average power density is 20 W/cm^3. Three out of 8 control rods, located close to the periphery of the center graphite reflector, can keep the reactor subcritical at cold zero power condition. Likewise, 4 out of 8 shutdown blades provide adequate shutdown margin when inserted into the bed of pebbles. Temperature coefficients of reactivity calculated for the Mk1 core are summarized in TABLE II. While the fuel, graphite moderator and coolant have strong negative temperature coefficients of reactivity, both center and outer graphite reflectors have small positive reactivity feedback. As the reflectors temperature will strongly depend on the coolant temperature, the net reactivity effect of uniform coolant temperature increase is expected to be close to zero, when the graphite reflectors temperature will equilibrate with the coolant temperature.

A preliminary estimate of the radiation damage to the center graphite reflector is 2.1 dpa/EFPY, implying approximately 10 EFPY lifetime. The peak radiation damage to the outer solid graphite reflector is 0.03 dpa/EFPY, implying that there will be no need to replace this reflector over the FHR plant lifetime.

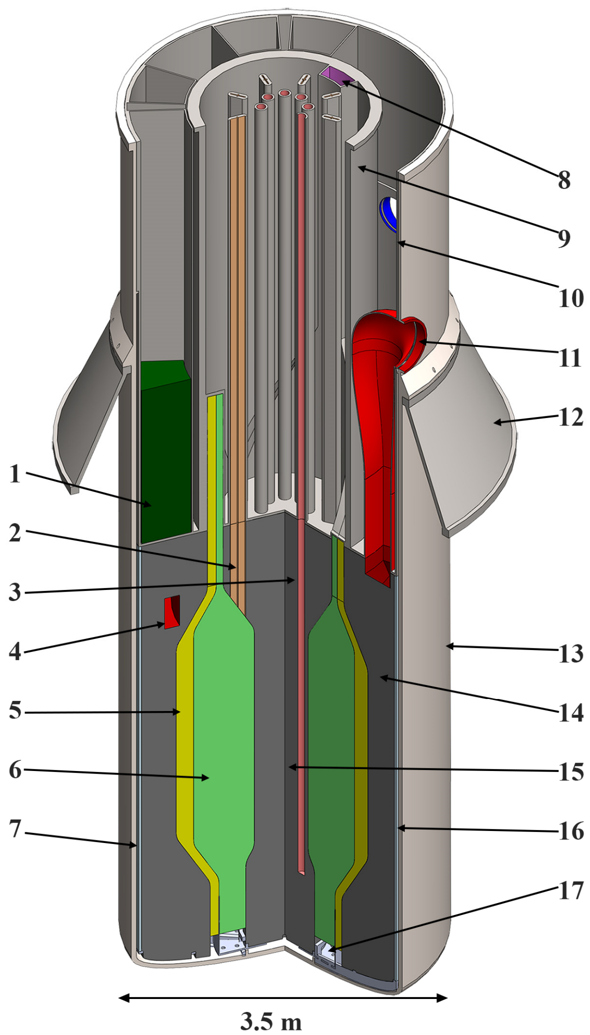

Fig. 4. Mk1 reactor vessel. 1: Fuel Canister. 2: Shutdown Blade Channel. 3: Control Rod Channel. 4: Hot Salt Collection Annulus 5: Graphite Pebbles. 6: LEU Pebbles. 7: Downcomer. 8: Defueling Well. 9: Vessel Inner Lid. 10: Vessel Outer Lid. 11: Hot Salt Extraction. 12: Support Skirt. 13: Reactor Vessel. 14: Outer Reflector. 15: Central Reflector. 16: Core Barrel. 17: Divider Plate.

TABLE II
Mk1 Temperature Coefficients of Reactivity [4]

Component	Temperature
Fuel	-3.8 (pcm/K)
Coolant	-1.8 (pcm/K)
Center Graphite	+0.9 (pcm/K)
Graphite Moderator	-0.7 (pcm/K)
Outer Graphite Reflector	+0.9 (pcm/K)

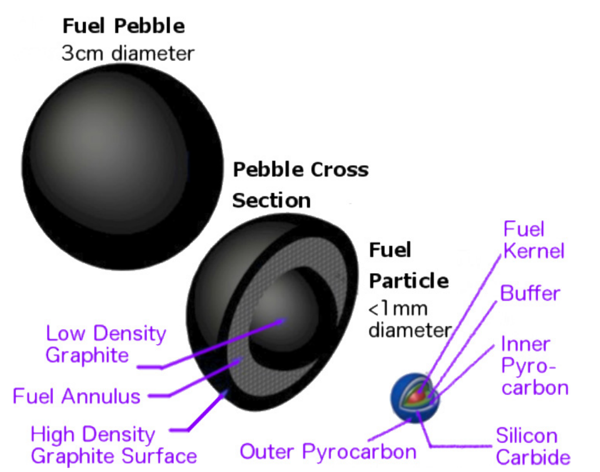

Fig. 5. Detail of Mk1 fuel compact geometry and composite components.

IV. THERMAL HYDRAULICS

This section provides a list of key thermal hydraulic phenomena for FHR technology and presents a high-level overview of codes that have been used for thermal hydraulic steady-state and accidental transient analyses of FHRs. This section then focuses on the experimental basis needed to identify FHR-specific phenomenology and validate FHR modeling codes. A more detailed version of this discussion is presented elsewhere [5].

IV.A. Key FHR Thermal Hydraulic Phenomena

Fluoride salts are low-volatility fluids with high volumetric heat capacities, melting temperatures, and boiling temperatures. The differences in thermal hydraulic phenomena in FHRs emerge from the differences in the thermophysical properties of the fluoride salts and the structural materials used with them, compared to other reactor coolants and their typical structural materials.

Fluoride salts have high volumetric heat capacities. The volumetric heat capacity of the primary coolant flibe exceeds even that of water. Therefore, FHRs operate with lower primary coolant volumetric flow rates, pressure drops, and pumping power than light water reactors (LWRs). These FHR operating parameters are also much lower than those for SFRs and high temperature gas-cooled reactors (HTGRs).

The fact that low volumetric flow rates of fluoride salts can transport large amounts of heat has many implications for the design of FHRs. For example, this characteristic makes fluoride salts particularly effective in passive, buoyancy-driven natural circulation heat transfer [6]. For future FHR reactors to be commercially attractive, it is critical that FHR designers leverage the favorable thermophysical properties of the fluoride salts to the maximum degree possible, while simultaneously mitigating the impacts of the non-favorable properties – primarily the high freezing temperature of the fluoride salts.

The following subsections review key thermal hydraulic phenomena that arise from the unique thermophysical properties of the fluoride salts and FHR structural materials.

IV.A.i. High Prandtl Number Coolant

The thermal conductivity of the baseline FHR primary coolant flibe is greater than water. However, flibe is also a highly viscous fluid, thus making it a high Prandtl number (Pr) fluid (~13). Most previous nuclear experience is with moderate Pr (~1 for water/helium-) or low Pr (~10^{-3} for sodium).

The greater thermal conductivity of flibe creates the potential for achieving heat transfer coefficients comparable to those for water even though the viscosity of flibe is much higher. However, the high volumetric heat capacity of flibe means that FHR convective heat transfer commonly occurs at Reynolds numbers that result in laminar or transition regime flow even under forced circulation, and natural circulation heat transfer is almost always in the transition or laminar regime. For this reason, unlike reactors using other coolants, FHR designs will commonly optimize to use enhanced heat transfer surfaces or small-diameter flow channels, such as those occurring in pebble beds.

IV.A.ii. Potential for Freezing (Overcooling Transients)

Mixtures of fluoride salts have high freezing temperatures, typically between 320°C and 500°C, which makes overcooling transients an important topic for design and safety analysis. The 8-MWth Molten Salt Reactor Experiment (MSRE), which operated from 1965 to 1969, experienced freezing in its air-cooled radiator; the radiator was then thawed without damage [7]. The lack of damage can be attributed in part to the particularly low volume change that the MSRE coolant salt, flibe, experiences upon freezing – about 2.07% [8].

Moreover, buoyancy forces can lead to significant flow re-organization in porous media, such as the pebble bed core and the shell side of twisted tube heat exchangers in FHRs [6]. Buoyancy forces are likely to be significant in FHRs as they operate at relatively low Reynolds numbers compared to water-, liquid-metal-, or gas-cooled reactors. This makes FHRs resilient to high thermal gradients such as cold spots from overcooling or hot spots due to local power peaking.

IV.A.iii. By-pass Flow

The graphite reflector blocks in the FHR can shrink and swell as complex functions of irradiation and temperature. These changes can lead to the formation of gaps between the blocks through which coolant will flow. The nature of this by-pass flow must be carefully studied to assess the impact on temperature profiles within the fuel blocks. By-pass flows can have significant effects on the coolant outlet temperature gradient. For fast transients, especially, detailed temperature profiles of the coolant should be taken into account for thermal stress calculations on metallic structures outside the core.

IV.A.iv. Radiative Heat Transfer

At high operating temperatures, radiative heat transfer to and from the reactor cavity, as well as total heat transfer to and from the reactor vessel, must be calculated. Likewise, wavelength-dependent absorption data are needed for coolant salts to allow their radiative interactions with heat transfer surfaces to be assessed.

IV.B. Thermal Hydraulic Modeling

Numerous computer codes have been written to simulate the thermal hydraulic characteristics of reactor cores and primary loops under steady-state and operational transient conditions, as well as potential accidents. New versions of some of these codes can be expected to be developed, and efforts are now focused on adapting existing codes and developing new ones for the new generation of advanced LWRs as well as HTGRs. A similar capability is needed to properly model steady-state and transient thermal hydraulic phenomena for the FHR, with an initial focus on design codes that will allow for rapid prototyping of the FHR system. The IRP is now focusing on developing models using two existing codes: RELAP5 and Flownex. These efforts are summarized here.

IV.B.i. RELAP5 Modeling

At this point on the development path of FHR technology, most thermal hydraulic analyses have been performed using the RELAP5 systems analysis code. Although RELAP5 was originally developed for thermal hydraulic analysis of LWRs and related experimental systems during loss-of-coolant accidents and operational transients, the code has recently been improved to simulate candidate Generation IV designs cooled by gas, supercritical water, and lead-bismuth. Liquid salt coolants, and more specifically flibe, have also been implemented into RELAP5 [9], which allows it to model thermal hydraulic steady-state and transient phenomena for the Mk1 PB-FHR.

Correlations for heat transfer and friction losses in the pebble bed core can be manually implemented into the code, but a significant validation effort of these correlations is required. Because of its wide use in the nuclear industry for design and licensing of reactors, RELAP5, along with sister codes like TRACE, appear to be good candidates for simulation of FHR steady-state and transient responses. As an example, Galvez used RELAP5 to simulate the transient response of an earlier FHR design to a loss of forced circulation with scram [10] and used this model to optimize dimensions of the DRACS loop for a 900-MWth FHR. However, additional efforts are needed to properly account for all phenomena described in the previous section if RELAP5 and other thermal hydraulics codes are to be used as the main system analysis codes for thermal hydraulic behavior of the FHR, and a significant verification and validation (V&V) will be needed.

IV.B.ii. Flownex Modeling

Flownex is a one-dimensional thermal-fluids analysis software whose main purpose is to model thermal-hydraulic systems [11]. Flownex was used to model several systems in the Pebble Bed Modular Reactor (PBMR), including the main power system and several supporting subsystems. Both transients and steady-state cases were studied. Flownex was partially verified and validated against other codes for the PBMR, but no such effort has been undertaken for FHRs. Benchmarking Flownex against other codes and experimental data will be very valuable in terms of code V&V.

In parallel with RELAP5, Flownex will be used to model the PB-FHR thermal hydraulic systems and subsystems, including the main salt loop and the DRACS. A detailed estimate of salt volumes and salt flow paths is required to do this. Efforts are underway to quantify salt volumes and outline flow paths both within the reactor vessel and outside it. The Flownex model can be enhanced by appending the NACC on the secondary side of the CTAHs.

Single phase natural circulation is an important mechanism that transfers decay heat from the core to the environment, via the DRACS. Therefore, accurate modeling of single phase natural circulation in both RELAP5 and Flownex is crucial. Simple natural circulation loops can be modeled analytically, and using Flownex and RELAP5, and relevant system response metrics, such as mass flow rates and temperatures, can be compared. Such efforts are ongoing with models based on an experimental test loop built at UCB: the Compact Integral Effects Test (CIET) Test Bay, described in the following section. Preliminary results from this V&V study indicate that RELAP5 is an appropriate tool to model the CIET Test Bay, as a first step towards predicting the performance of the passive decay heat removal system of FHRs, with the code showing agreement within 5% with analytical predictions and within 10% with experimental data for natural circulation in the laminar regime, and agreement within 8% with analytical solutions and within 25% with experimental data in the transition regime [12].

IV.C. Integral Effects Tests (IETs) for Thermal Hydraulic Model Validation

Although preliminary thermal hydraulic modeling of the FHR has been performed with systems analysis codes, these codes, in their current state, are not capable of capturing some of the key FHR thermal hydraulic phenomena. Significant V&V efforts are therefore needed to increase the reliability of these codes to properly model thermal hydraulic phenomena for the FHR. Some of these efforts are presented here.

IV.C.i. Scaling and the Use of Simulant Fluids

Thermal hydraulic transient phenomena associated with FHR response to licensing basis events (LBEs) evolve over brief time periods of minutes to days. Therefore, the major constraint on experiments is not duration, but rather scale, because of the impracticality of performing IETs at the full-power level of the reactor. The major importance of geometric and power scaling was recognized in earlier studies of FHRs [13].

Liquid salts are unique reactor coolants because simulant fluids can replicate salt fluid mechanics and heat transfer phenomena at reduced length scales, temperatures, and heater and pumping power, with low scaling distortion. UCB has identified a class of heat transfer oils that, at relatively low temperatures (50-120°C), match the Pr, Reynolds, and Grashof numbers of the major liquid salts simultaneously, at approximately 50% geometric scale and heater power under 2% of prototypical values [14].

Experiments have shown that the Pr numbers of Dowtherm A – a commonly used heat transfer oil – match those of flibe for certain temperature ranges. Specifically, the Pr of flibe throughout the expected Mk1 operating temperatures (600-700°C) can be matched by Dowtherm A with a much lower temperature range (57-117°C) [5]. The availability of such simulant fluids significantly reduces the cost and difficulty of performing IETs required for system modeling code validation for reactor licensing, compared to working at prototypical temperatures and power levels with the actual coolant. Thus, the key IET experimental facilities needed to validate the FHR transient analysis codes can be university-scale facilities built and operated during the pre-conceptual design phase for FHR technology. Two of these facilities, operated at UCB, are described here.

IV.C.ii. CIET Test Bay

The CIET Test Bay is a scaled-height, reduced-flow-area loop, which reproduces the integral thermal hydraulic response of the FHR primary coolant flow circuit using Dowtherm A. In the CIET Test Bay, heat is added to the fluid through an annular, electrically-heated pipe, and removed through water-cooled heat exchangers. Mass flow rates and bulk fluid temperatures along the loop are collected at various levels of heat input. The facility can run in both steady-state and transient modes to model the performance of the primary loop of FHRs under a defined set of LBEs. More details can be found elsewhere [6].

The CIET Test Bay has also provided data for V&V efforts, and was instrumental in providing experience with operation and maintenance of components to be used on the CIET Facility, which is described in the next section.

IV.C.iii. CIET Facility

UCB has designed the CIET Facility to reproduce the integral transient thermal hydraulic response of FHRs under forced and natural circulation operation. CIET provides validation data to confirm the predicted performance of the DRACS under a set of reference LBEs. Using Dowtherm A at reduced geometric and power scales, test loops for CIET are fabricated from stainless steel tubing and welded fittings, allowing rapid construction and design modifications. The simplicity of the construction, compared to the complexity and safety requirements for tests with the prototypical salt, was a key element in enabling the experiments to be performed at lower cost than previous IETs for other types of reactors. Fluid flow paths in CIET replicate those in the FHR shown in Fig. 3.

The research program for CIET has been planned as follows, with specific objectives associated to each step:

i) Isothermal, forced circulation flow around the loop, with pressure data collection to determine friction losses in the system.
ii) Steady-state forced and coupled natural circulation in the primary loop and the DRACS loop.
iii) Thermal transients: startup, shutdown, loss of forced circulation with scram and loss of heat sink with scram.

More details about the design of the CIET facility can be found elsewhere [15].

V. POWER CONVERSION SYSTEM

The Mk1 design uses a NACC for its power conversion system. An overview of the system can be found in Section II.A. This section provides details of specific Mk1 NACC components.

V.A. CTAHs

The Mk1 PB-FHR design has two CTAHs, which transfer heat from the primary salt to compressed air from the GT system. The CTAHs are located below grade in the filtered confinement volume, immediately adjacent to the PB-FHR reactor cavity. The CTAHs use an annular tube bundle formed by coiled tubes with air flowing radially outward over the tubes, as shown in Fig. 6.

The coiled tube assembly of each CTAH is located in a vertical cylindrical steel pressure vessel that is insulated

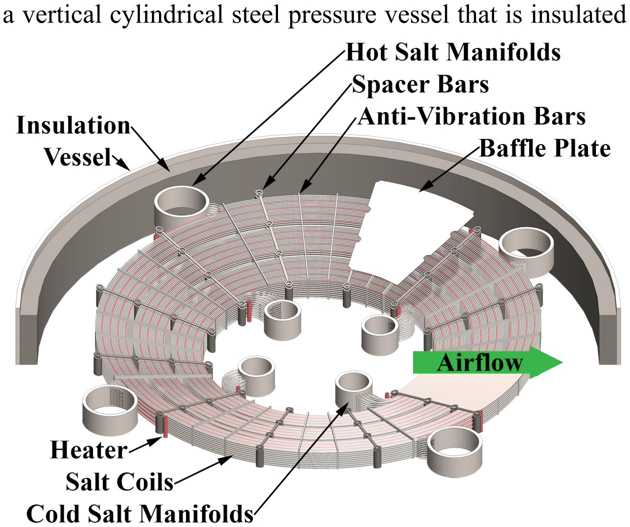

Fig. 6. Isometric view of the CTAH sub-bundle 3D model with major components labeled.

on the inside to allow the vessel to operate at near room temperature. The air temperatures in the CTAHs are comparable to air temperatures inside modern heat recovery steam generators (HRSGs) for natural gas combined cycle plants, so the design of the insulation system can draw upon this experience base.

Each CTAH uses inlet and outlet manifold systems that distribute the liquid flow into and out of the coiled tubes. The inlet manifolds consist of four vertical hot liquid manifold pipes that enter from the top of the vessel and extend downward along the outside of the coiled tubes. The Mk1 hot manifold pipes are 0.320 m in outside diameter with a 0.020-m-thick wall, and the cold manifold pipes are 0.215 m in diameter with a 0.020-m-thick wall. At each tube row elevation in the coiled tube bank, hot liquid is supplied into multiple tubes that then wrap around the coiled bundle, forming a single "lane" of tubes at that elevation that wraps around the tube bank one or more times. Likewise, at the center of the tube bundle, there are three vertical cool liquid manifold pipes that receive the flow from the tubes, and direct it downward and out of the heater vessel.

V.B. Power Conversion and Turbine

To implement nuclear heating, the Mk1 NACC design modifies the GE 7FB GT as shown in Fig. 7. Flow through the NACC system occurs as follows:

Air intake occurs through a filter bank, and the air is compressed to a pressure ratio of 18.5. For a nominal 15°C, 1.01 bar ambient condition, the air exits the compressor at a temperature of 418°C.

After the compressor outlet, the air passes through a high-pressure CTAH and is heated up to a turbine inlet temperature of 670°C. The air is then expanded to approximately the same temperature as the compressor outlet temperature, 418°C. This criterion determines the expansion ratio of the first expansion stage at nominal design conditions.

The air is then reheated back up to 670°C by passing through a second low-pressure (LP) CTAH. It is important to design this LP external heating system to have minimum pressure drop in order to achieve acceptable circulating power loss and cycle efficiency.

After the LP CTAH, the air is above the auto-ignition temperature of natural gas. To provide power peaking, a fuel such as natural gas can be injected and burned to increase the turbine inlet temperature and the power output.

The heated air is then expanded down to nearly atmospheric pressure and 395-700°C, depending on the peak power level, by passing through an additional set of LP turbine blades, before entering the HRSG. The HRSG must be designed to accommodate a relatively wide range of air inlet temperatures due to the large change that occurs between low-carbon base-load operation and peak power operation with natural gas injection.

Reheat and external firing are both proven technologies and are commercially available on large industrial GTs (e.g. Alstom GT11N2 and Alstom GT24).

The modifications needed to accommodate nuclear heat for the GE 7FB GT include an extended shaft to accommodate reheat and a redesigned casing.

VI. CONCLUSIONS

This paper presents a brief summary of past and ongoing research related to FHR-enabling topics at UCB. Various 3D models of components and systems for the Mk1 PB-FHR plant are also presented. A more in-depth look at the technical aspects – including code development,

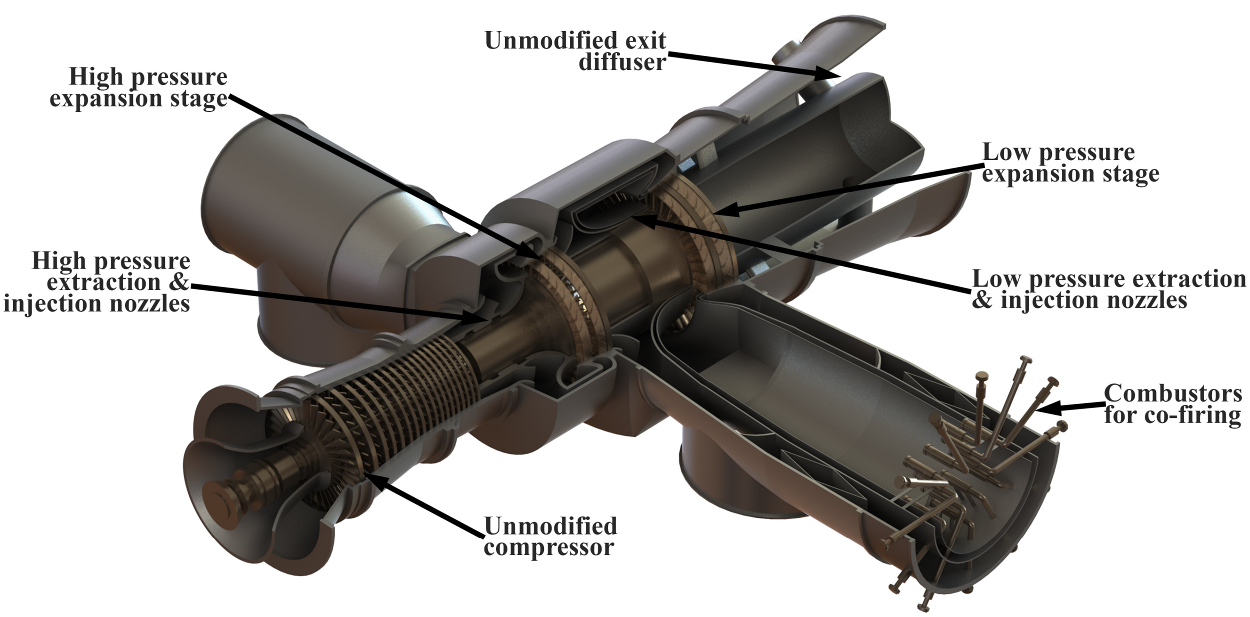

Fig. 7. Detail of a GE F7B GT.

materials issues, licensing strategies, and technology development roadmaps – can be found elsewhere in works produced by the member institutions of the FHR IRP [5], [16]–[19].

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