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(54) Title: PRODUCING POWER WITH MOLTEN SALT REACTORS

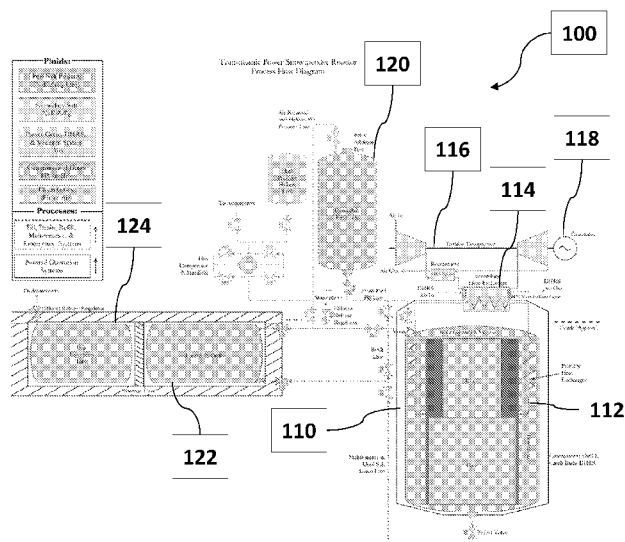


Figure 1

(57) Abstract: A power system including a fuel salt with between 48 and 62 mole percent NaF; between 31 and 40 mole percent ZrF₄; and between 5 and 13.2 mole percent UF₄. The power system also including a natural circulation reactor with a height between 3.25 and 4 meters and a radius between 0.5 and 1.3 meters.

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Producing Power with Molten Salt Reactors

TECHNICAL FIELD

This invention relates to molten salt reactors.

BACKGROUND

In the 1950's and 60's, two liquid fueled molten salt reactors, the Aircraft Reactor
5 Experiment (ARE) and the Molten Salt Reactor Experiment (MSRE) were constructed
and operated. These systems showed, among other things, that the emergency cooling
requirements for a liquid fuel could be greatly simplified as a result of the convective
nature of the fluid. Although promising in terms of the safety potential, molten salt
reactors can require costly, complex systems and components in other areas of the design,
10 most notably in regard to chemical processing, material compatibility, and special nuclear
material accounting.

SUMMARY

This disclosure describes a nuclear reactor that builds on the inherent safety
advantage of liquid fuel while employing a cost-effective simplified design that is built to
15 succeed in today's energy market. This reactor is an integral, Very Small Modular
Reactor (VSMR), that runs entirely on natural circulation and has a variable power output
(5 - 50 MW_{th}). The reactor is passively safe, and has a high capacity factor as a result of
its long refueling intervals (1000-3000 GW_{th}h) and minimal maintenance requirements. It
requires no commercially unavailable enrichment processes (⁷Li, ³⁷Cl, ²³⁵U > 5%), and
20 produces next to no tritium. Based on the small size and low cost, this reactor is
anticipated to be useful in the alternative energy market for unique applications such as
data centers, mining, microgrids, and critical infrastructure.

The details of one or more embodiments of the invention are set forth in the
accompanying drawings and the description below. Other features, objects, and
25 advantages of the invention will be apparent from the description and drawings, and from
the claims.

DESCRIPTION OF DRAWINGS

Figure 1 is a process flow diagram overview of the system integration and layout.

Figure 2 is an axial cross-section of an integral reactor vessel.

Figure 3 is the NaF-ZrF₄-UF₄ phase diagram, with the compositional range
5 selected for the reactor shaded.

Figure 4 shows vapor pressure at 900°C as a function of the mole % of ZrF₄ in the NaF-ZrF₄ system.

Figure 5 is a radial cross-section of the core layout of the reactor vessel of Figure
1.

Figure 6 is the zirconium hydride phase diagram overlaid with the partial pressure
10 of hydrogen ($1 \cdot 10^{-4}$ atm), with α indicating an hcp solid solution, β a bcc solid solution,
 δ an fcc, and ϵ an fct.

Figure 7 plots the infinite multiplication factor (k_{inf}) for a square pitch unit cell as
a function of the salt volume fraction (SVF), with the legend indicating the pitch of the
15 cell in cm.

Figure 8 plots the effective multiplication factor (k_{eff}) and corresponding non-
leakage probability (PNL) as a function of core diameter.

Figure 9 plots energy per unit volume of fuel salt as a function of the neutron
leakage in the core. The dotted line represents the expected leakage of the geometry
20 displayed in Figure 5 (1.5m core diameter).

Figure 10 shows the infinite multiplication factor as a function of isothermal unit
cell temperature. The dotted line is defined by a slope of -4.7 pcm K^{-1} .

Figure 11 is an overview of the PFD for the stationary plant layout.

Figure 12 shows optional mobile configuration components. The top truck
25 includes the refuel/maintenance module built in and the bottom truck carries a new empty
storage cask.

Figure 13 shows the life cycle for mobile refueling configuration. The fuel system
truck and components are reused, while the plant and cask components are delivered and
not reused.

Figure 14 shows example options for multi-unit site configuration.

Like reference symbols in the various drawings indicate like elements.

DETAILED DESCRIPTION

This disclosure describes a power system that is designed for simplicity and ease of operation. Both the primary and secondary salt loops can function on natural
 5 circulation, eliminating the need for any molten salt pump components, increasing passive safety, and dampening any operational or unplanned thermal transients. The secondary loop could also include a non-nuclear molten salt pump to increase thermal efficiency. The secondary loop is connected by radiator to a tertiary power cycle. This power cycle is an open-air Brayton cycle with regeneration. Operators can control core
 10 power and temperature change through the primary loop by regulating turbine load.

Power cycle and regulation

Figure 1 is a process flow diagram of a power system 100. The power system 100 includes a reactor vessel 110, a primary heat exchanger 112, a secondary heat exchanger 114, a turbine-compressor 116, a generator 118, a clean salt fill tank 120, a used salt tank
 15 122, and a gas expansion tank 124. The thermal hydraulic operation of the power system 100 is summarized in Table 1.

Table 1 Summary of the TPX heat loops and power cycle

Primary	
Operational Principle	Natural Circulation
Working Fluid	NaF-ZrF ₄ -UF ₄
Power Range (MW _{th})	5 – 50
Operational Temperature Range (°C)	600 – 700
Secondary	
Operational Principle	Natural Circulation (optional Forced Circulation)
Working Fluid	NaF-ZrF ₄
Operational Temperature Range (°C)	690 – 550
Tertiary	
Operational Principle	Open Brayton Cycle (optional Combined Cycle)
Working Fluid	Air
Turbine Intake Max. Temp. (°C)	680

The specific reactor vessel design and fuel salt composition described below combine to provide long periods of continuous power operation without maintenance, with brief intervals for refueling, and completely passive emergency safety. The simplicity and long lifetime of operations opens up the possibility of limited equipment onsite during operation, eliminating material accountability risk and capital cost as much of the equipment could be reused across multiple sites.

Primary loop circulation

The range of acceptable core power levels are limited by the ability to remove heat through natural circulation. In general, the onset of natural circulation occurs when the Rayleigh number is sufficiently high that buoyancy-driven flow (free convection) can occur. Circulation of this flow is maintained in the primary loop if the flow generated by natural circulation forces can overcome the frictional losses the flow endures around the loop.

Through conservation of energy, conservation of momentum, and a few reasonable assumptions, the relationship between primary loop power, natural convection, and geometry can be simplified to a single equation describing achievable fluid velocity as

$$u = \left(\frac{\beta Q L_{th} g}{\rho A_{core} c_p \Pi} \right)^{\frac{1}{5}} \quad \text{Eq. 1}$$

where u is achievable core velocity; β is thermal expansion coefficient, Q is core power, L_{th} is the characteristic thermal length of natural convection, g is gravitational acceleration, ρ is density, A_{core} is core flow area, c_p is specific heat capacity, and Π is the dimensionless flow loss term.

The derivation of Eq. 1 assumes

- 1-D flow
- Boussinesq approximation
- Incompressible working fluid
- Core inlet temperature (T_C) is constant (600 °C)
- Power generated in the core (Q) is equal to that removed by the heat exchanger

In determining the operation power/temperature range under natural circulation, the achievable core velocity can be compared to the minimum required core velocity, v , (determined from conservation of energy alone). This is the required velocity to remove some amount of core power at a given temperature change across the core. The
 5 minimum required core velocity (v) can be described as:

$$v = \frac{Q}{\Delta T_{\text{core}} c_p \rho A_{\text{core}}} \quad \text{Eq. 2}$$

The core power and temperature change (ΔT) are functions of the heat transfer design and controlled by the tertiary cycle. Characteristic length of convection (distance from midplane of core to midplane of heat exchanger), and loop resistance, are geometry
 10 dependent parameters, determined by the vessel and internal design. The vessel and heat transfer designs are interrelated through the heat exchanger geometry. All other parameters are thermophysical properties of the salt or determined by the reactor physics.

The dimensionless loop resistance, Π , accounts for flow resistance which inhibits natural circulation due to friction, area change, and other K-loss contributors. It is represented as

$$\Pi = \sum_{i=1}^N \left[\frac{1}{2} \left(\frac{f_i L_i}{D_h} + K_i \right) \left(\frac{A_{\text{core}}}{A_i} \right)^2 \right] \quad \text{Eq. 3}$$

15 where f_i is the friction factor, L_i is the flow length, D_h is the hydraulic diameter, K_i is the additional K-loss term, and A_i is the flow area.

Geometry

Figure 2 is an axial cross-section of the reactor vessel 110 showing a reactor core 126 and primary heat exchanger 128 inside a containment shell 130. The geometry of the
 20 reactor was chosen based the following required conditions:

$$(1) \quad u \geq v$$

Achievable core velocity (u) exceeds minimum required core velocity (v). This ensures natural circulation results in sufficiently high flow rates for all heat removal during operation.

$$(2) \quad Q = \tau \cdot V_{\text{salt}}$$

5 Sufficient salt (V_{salt}) for 10 years of operation at highest core power (Q). The lifetime constant (τ) is a function of the achievable burnup.

$$(3) \quad H_{\text{upper vessel}} > 1.5 \text{ m}$$

The upper vessel interior height (riser, upper plenum, and gas gap) must be taller than the core height to accommodate control rod withdrawal and drive shaft connectors in
10 order to reduce vessel penetrations.

$$(4) \quad Q_{\text{HX}} \geq Q$$

The power removed by the heat exchanger must meet or exceed the maximum core power. This must also take into account heat exchanger fouling.

$$(5) \quad D_{\text{vessel}} \leq 2.6 \text{ m}$$

15 The overall vessel diameter was limited to 2.6m (including the vessel itself) to fit within standard road-shipping limits. It should be noted that a wider vessel will allow for more salt and thus a higher power. Wider vessels increase manufacturing challenges and initial cost.

The base of the heat exchanger was also set to be above the core to minimize
20 activation and material damage.

In selecting the geometry that meets these conditions, higher powers allow for higher peak electrical output and larger operational ranges. Minimal salt volume is also preferred, because vessel materials, salt fabrication, and total actinide content are directly proportional to salt volume and it is assumed they dominate overall plant costs. The
25 reactor vessel 110 has a riser radius (R_R) of 0.75 m, a riser height, (R_H) of 0.8 m, a heat exchanger channel thickness, T_{HX} (the annulus between the vessel and inner barrel where

the heat exchanger is located) of 0.4 m, a heat exchanger channel primary side flow area, (A_{HX}) of 1.4 m, and a thicknesses of the upper and lower gaps, (T_{UG} & T_{LG}) of 0.25 m. Some reactor vessels have different geometries. The riser radius (R_R) can be between 0.5 m and 1.05 m. The riser height (R_H) can be between 0.2 m and 1.4 m. The heat exchanger channel thickness (T_{HX}) can be between 0.1 m and 0.65 m. The heat exchanger channel primary side flow area (A_{HX}) can be between 0.25 m and 2.45 m. The thicknesses of the upper gap (T_{UG}) can be between 0.05 m and 0.6 m. The thicknesses of the lower gap (T_{LG}) can be between 0.05 m and 0.45 m.

Salt Selection

Table 2 presents a potential composition for the reactor fuel salt along with its thermal physical properties.

Table 2 A summary a potential fuel salt's composition and properties.

Composition (Mole Fraction, X_i)	
NaF	0.504
ZrF ₄	0.364
UF ₄	0.132
Thermal Physical Properties	
Melting Temperature (°C)	550
Density* (kg m ⁻³)	3772
Thermal Expansion Coefficient* (°C ⁻¹)	0.000273
Heat Capacity (J kg ⁻¹ °C ⁻¹)	946
Thermal Conductivity (W m ⁻¹ °C ⁻¹)	.810
Viscosity* (Pa s)	0.0098

* Values have been given for the expected minimum operational temperature of 600 °C

Absorption cross section

In 1970, the director of the reactor chemistry division at Oak Ridge, Warren Grimes, proposed selection criteria for the use of molten salts in reactor applications. At the time, an emphasis was placed on the breeding performance of the fuel salt, causing many viable options to be passed over. Specifically, isotopes that had higher absorption cross-sections than ⁷Li were deemed unfit to serve as a major constituent of the fuel salt. If this criterion is removed, it can be seen that a wider range of material combinations are available to meet the demands of the current industry (Table 3).

Table 3 Elements or isotopes which may be tolerable in high-temperature reactor fuels.

Material	Thermal Neutron Absorption Cross Section (barns)
¹⁵ N	0.000024
O	0.0002
² H	0.00057
C	0.0033
F	0.009
Be	0.01
Bi	0.032
⁷ Li	0.033
¹¹ B	0.05
Mg	0.063
Si	0.13
Pb	0.17
Zr	0.18
P	0.21
Al	0.23
¹ H	0.33
Ca	0.43
S	0.49
Na	0.53
³⁷ Cl	0.56
Sn	0.6
Ce	0.7
Rb	0.7

Melting point

By introducing more parasitic constituents, feasible compositions in terms of achieving and maintaining criticality become restricted to mixtures that can accommodate higher fractions of fissile material. As higher concentrations of uranium tend to lead to higher melting points, certain salt combinations are eliminated based on the practicality of their operational temperature range. For this reactor, it was decided that in order to limit the strain on the moderator and structural materials, only salts that melt at or below 550 °C would be considered. Figure 3 is the NaF-ZrF₄-UF₄ phase diagram, with the compositional range 132 showing those that melt below 550 °C.

Acid-base chemistry

As we move away from the realm of aqueous solutions, the role of acid-base chemistry is something that can often be forgotten in the early stages of the design

process. However, despite the limited experience with molten salts, the acid-base nature of these systems has been used to explain the trends observed in several key design parameters such as corrosion performance and vapor pressure.

Figure 4 shows vapor pressure at 900 °C as a function of the mole % of ZrF_4 in the NaF- ZrF_4 system. This figure has been adapted from K. Sense, C. Alexander, R. Bowman and R. Filbert, "Vapor pressure and derived information of the sodium fluoride-zirconium fluoride system. Description of a method for the determination of molecular complexes present in the vapor phase.," Journal of Physical Chemistry, 1957. As UF_4 is expected to exhibit the effect of a weak base (due to the stronger Gibbs free energy of formation of UF_3) the trend observed is assumed to be roughly applicable for the NaF- ZrF_4 - UF_4 system.

In molten salts, basic species (electron pair donors) such as the alkali halides, (e.g. NaF) can complex with the more acidic constituents (electron pair acceptors, e.g. ZrF_4) present in solution. This complexation has been used to describe the non-linear trend in vapor pressure (see Figure 4) that has been observed as a function of composition, by indicating that the volatile acidic constituents are stabilized.

The acid-base ratio of these components can also be shown to affect several secondary equilibria within these systems, specifically, the stability of corrosion products such as CrF_2 (Table 4). Extremely basic solutions are thought to drive corrosion through the formation of CrF_3 , while extremely acidic solutions are also considered unfavorable.

Table 4 Equilibrium concentrations of dissolved chromium upon contact of the pure metal with several fuel salt compositions.

Salt Composition (mol fraction)			[Cr] (ppm)	
NaF	ZrF_4	UF_4	600°C	800°C
0.469	0.500	0.041	2300	2550
0.490	0.470	0.040	1700	2100
0.553	0.410	0.037	975	1050

To reduce the complexity of the reactor as a whole, priority has been given to fuel salts that both have a low vapor pressure and exhibit good inherent corrosion performance. If the acidity of the solution is appropriately selected, it is possible that external redox control may not be required for the regulation of corrosion and UF_3

solubility. Some fuel salt compositions include between 48 and 62 mole percent NaF; between 31 and 40 mole percent ZrF₄; and between 5 and 13.2 mole percent UF₄.

Additional considerations

Although NaF is probably one of the only constituents that can be considered a commodity chemical, emphasis was placed on de-risking the supply chain by not becoming reliant on enrichment processes that are not commercially available in the United States (⁷Li, ³⁷Cl, ²³⁵U > 5%). By eliminating the option of ⁷Li, tritium production within these systems can be greatly reduced, and thus it was decided to also eliminate the possibility of using Be in order to devise a reactor that could operate with almost no tritium migration considerations.

Core Design

Figure 5 shows the radial cross-section of one of TPX's potential core layouts, with Table 5 summarizing the materials and dimensions used. The design process that has gone into this selection is discussed below.

Table 5 A summary of the moderator and control rod materials and dimensions.

	Moderator	Cladding	Control Rod
Material	ZrH (δ – phase)	SiC	MHC*
Radius or Thickness (m)	0.009	0.001	0.010

*Molybdenum Hafnium Carbide

Moderator

Metal hydrides are intriguing moderators because of their high hydrogen density and wide range of allowable operational temperatures. Of the metal hydrides, δ – phase zirconium hydride is one of the most promising candidates, as it possesses good neutronic properties (low absorption cross-section, high hydrogen density) and shows adequate hydrogen stability when subject to a temperature gradient (Figure 6).

However, in order to be compatible in a fluoride salt environment, zirconium hydride must be clad so that it does not come in direct contact with the salt. Potential claddings must be able to withstand high temperature and radiation environments, be compatible with fluoride salts themselves, have low absorption cross-sections, and act as

a barrier for hydrogen diffusion in order to mitigate outgassing. With these criteria in mind, as well as the recent advancements in ceramic composite manufacturing, silicon carbide (SiC) composites were chosen. A SiC cladding thickness of 1 mm was selected.

Figure 7 shows the infinite multiplication factor (k_{inf}) for a square pitch unit cell as a function of the salt volume fraction (SVF), with the legend indicating the pitch of the cell in cm. As seen in Figure 7, although 4 and 5 cm pitch cells produced higher multiplication factors, the peak 3 cm geometry (0.66 SVF) was selected for the analysis moving forward. This choice was based around concerns over the manufacturing of large ZrH rods, as well as limits in the rods' peak centerline temperature.

Core Geometry

The core size was selected as a balance between neutronic/material performance and cost. Larger cores have lower neutron leakage and reduced flux levels in the vessel and moderator, leading to longer maintenance and refueling intervals. However, the increased vessel radius, salt volume, and total number of moderator rods contributes to the overall plant cost.

Figure 7 shows the effective multiplication factor (k_{eff}) and corresponding non-leakage probability (PNL) as a function of core diameter. The core diameter does not take into account the thickness or presence of a vessel material. Figure 8 shows the energy per unit volume of fuel salt as a function of the neutron leakage in the core. The dotted line represents the expected leakage of the geometry displayed in Figure 5 with a 1.5m core diameter.

The core size was chosen by selecting the smallest geometry capable of achieving and maintaining criticality (Figure 8 and Figure 9). Four-by-four rod assemblies (Figure 5) were defined for the full core analysis. However, other geometries may be used with this reactor.

Safety Performance

Although the emergency cooling requirements for liquid fueled systems are simplified as a result of the convective nature of the fluid, the reactor must still be able to be controlled and shutdown effectively. Figure 10 shows the infinite multiplication

factor as a function of isothermal unit cell temperature. The dotted line is defined by a slope of -4.7 pcm K^{-1} . As this calculation was performed at the unit cell level, it represents a conservative estimate of the slope that will exist within the full-scale reactor as no considerations of leakage have been made. Figure 10 illustrates the strong negative temperature coefficient that the system possesses ($-4.7 \text{ pcm } ^\circ\text{C}^{-1}$), allowing for further inherent stability and safety.

Molybdenum hafnium carbide is a high density, high strength material that is compatible in the molten fluoride environment. The control rod structure shown in Figure 5 has been designed so that the highest worth cluster can be removed and the reactor will remain sub-critical as per NRC regulations.

Operations

The specific reactor vessel 110 design and fuel salt composition described below combine to provide long periods of continuous power operation without maintenance, with brief intervals for refueling, and completely passive emergency safety. The simplicity and long lifetime of operations opens up the possibility of limited equipment onsite during operation, eliminating material accountability risk and capital cost as much of the equipment could be reused across multiple sites. The operational modes include: Fuel Loading; Startup; Power Generation (Normal Operation); Shutdown; Maintenance & Refill; Refuel; and Emergency Operation.

Mobile Plant Component Options

The components common to all modes of operation include:

- Reactor vessel and internals
- Intermediate loop
- Power cycle
- Storage cask including used salt and contaminated gas tanks

Other components are used only during initial loading, maintenance, or refueling, therefore they do not need to remain on site and can be removed and used at other sites during the 10+ year period of Power Operation for a unit. These potentially mobile components include:

- Clean salt fill tank
- Helium supply/compressor system

It is anticipated that the entire initial plant equipment could be transported to site in 5 semi-trailer loads, with subsequent refuels conducted with components housed on a single trailer. The functionality of plant systems with these mobile components will be detailed in sections for each operational mode. A representative diagram for the stationary and mobile components of the mobile refueling option, including stationary plant layout and a pair of mobile component trucks representative of the refuel truck and empty cask transport truck are shown in Figure 11 and Figure 12. Figure 11 is an overview of the PFD for the TPX stationary plant layout. Figure 12 shows optional TPX mobile configuration components including a truck with the refuel/maintenance module built in and a truck carrying a new empty storage cask. Figure 13 shows the life cycle for mobile refueling configuration. The fuel system truck and components are reused, while the plant and cask components are delivered and not reused.

Fuel Loading

Frozen fuel bricks are loaded into the clean salt fuel tank via the brick addition port. This can occur at the plant or at a central facility if the mobile refueling option is selected. To remove volatiles and oxygen impurities, a vacuum is applied to the tank (via lines along valves 110-113 – the valves are shown on Figure 1). The bricks are heated to a temperature slightly below the melting point of the salt via heaters wrapped around the fill tank. The vacuum pump is then shut off and the atmosphere is replaced with helium (via lines along valves 114-116 & 110). Salt is heated further to melt it. Vessel heaters are turned on to prepare it to receive salt. Valves between the core fuel inlet and fill tank (100 & 101) are opened, allowing molten salt to flow into the core along the fill line, driven by gravity and/or helium gas backpressure. Once all fuel has been transferred to the vessel, the valves are closed and the reactor is ready for startup.

Startup

During startup, control rods are withdrawn, and the reactor begins to generate heat. The heat removal is ramped up via compressor-turbine actuation, until a steady-state

minimum operational power has been reached. At 5 MWth, the reactor is considered to be in Power Generation mode.

Power Generation (Normal Operation)

Once the reactor has reached 5 MWth, power regulation up to the peak thermal power is controlled primarily by turbine demand. Over the course of depletion, control rods are adjusted to maintain criticality.

During operation, gaseous fission products accumulate in the upper vessel space above the upper plenum, traveling through a penetration and open valves (107 & 108) into the used salt storage and gas expansion tanks. These large capacity tanks act as a hold-up volume for gases to decay to acceptable equilibrium activities such that effluence regulation limits would not be exceeded by a flow rate of release through the effluence regulator valve (110).

Anticipated operational transients of the largest severity are induced by power changes and accommodated by design, so other transients are lumped either in emergency operations or assumed to be covered by the ranges of normal systems operation.

Shutdown

The reactor is shut down by full insertion of all control rods, thereby ceasing the fission process. Decay heat is removed during normal shutdown operation by continued compressor-turbine actuation. Once a low temperature is achieved, the compressor-turbine system is decoupled by regulated opening of the secondary heat exchanger ventilation gates. Hot shutdown is maintained indefinitely in this mode until salt is moved into the storage cask tank or the reactor is restarted.

Maintenance & Refill

In the unlikely and undesirable event maintenance must be performed on-site, the reactor can be prematurely drained before refueling. From shutdown, the used salt drain line is heated and all valves to the used salt tank except for the refuel valve are opened (103 & 106). The refuel valve (102) is then opened and fuel is forced into the used salt

tank by helium backpressure. Valves along drain line (102, 103, & 106) are then closed. The salt is maintained above 200 °C in order to inhibit the effects of radiolysis during maintenance. Once maintenance is complete, the salt is heated to operational temperatures, downstream valves open (101, 104, 105, & 106), and the salt is reloaded through the refill line and fuel fill line via helium backpressure.

It should be noted, the highest turbine reliabilities are around 99.4%, meaning some maintenance will need to be performed on them during the operational life of the reactor (~3 weeks of downtime for a 10-year cycle). This maintenance may or may not affect the reactor itself if the configuration of the plant allows for the reactor to be maintained at or above 5 MWth during turbine maintenance.

Refuel

Before the refueling process can begin, refueling supplies are brought to the site. This includes a new storage cask containing used salt and contaminated gases tanks. If the plant includes the stationary refuel equipment configuration, new fuel bricks are also transported to site and loaded into the fill tank. If the fueling equipment is mobile, the new fuel bricks are loaded at the fuel facility and transported inside the fill tank.

Refueling begins in the same manner as maintenance. The salt is drained into the used salt tank and allowed to cool to via natural convection. The used salt tank is disconnected (via valves 106 & 108) and sits cooling until ready for transport to processing/disposal sites. The new used salt tank is then connected using the same valve connections. The system is flushed via helium and fresh fuel is loaded from the fill tank according to the same procedure as initial fueling.

Emergency Operations

As there are no pumps in the salt loops, design basis events/emergency scenarios involve a loss of flow in the tertiary, break within the fuel or coolant salt systems, or rupture of used salt tank. All emergency operations are expected to follow a reactor trip (shutdown configuration insertion of the control rods). The decay heat is expected to decline exponentially starting at ~6.5% of peak operational power.

Because the compressor is driven by the turbine, a loss of offsite power (LOOP) or other disconnection from grid does not inhibit standard operation heat removal via the power cycle, and the generator shaft could continue to provide power to auxiliary systems as necessary as the reactor coasts down. For this reason, a LOOP is treated as analogous to normal shutdown.

Loss of tertiary cooling (e.g., turbine malfunction), is accounted for by the secondary heat exchanger ventilation gates. These gates are on electromagnetic actuators, powered by the turbine-generator. On loss of turbine functionality, these gates drop open, allowing the system to cool via natural convection through the secondary heat exchanger.

Secondary loop rupture is accounted for through the design of the containment shell. Normally the shell is filled with inert gas and acting as an insulator for the primary system. When the secondary loop breaks, the salt within drains into the shell space, allowing increased heat transfer by both conduction and radiation from the reactor vessel to the shell, allowing the earth to serve as the ultimate heat sink.

Primary vessel breaks are held within the containment shell. The increase in surface area and increased heat transfer to earth allows for more effective passive to-earth cooling.

Rupture of the used salt tank involves no driving pressure, and the casks containing the tanks themselves will be lined concrete limiting material release. This scenario would be quickly detected by tank monitoring and relatively easy to handle.

Additional Procedures and Plant Designs

Secondary Salt Loading

Secondary salt is prepared and loaded much the same way fuel salt is prepared: heated in brick form, volatiles removed by vacuum, melted, and loaded into the loop via gravity and/or helium backpressure. The salt is loaded through a valve and pipe connecting to the salt at the top of the loop within the secondary heat exchanger radiator compartment. It is not anticipated that the secondary salt loop will be drained or refilled, however, an additional pipe-valve connection could be provided near a low elevation or the salt could be suctioned out through the same inlet.

Multiple Unit Configuration

While a mobile refuel option is quite attractive to customers only needing a relatively small amount of power, the same system could be scaled up using multiple units. In this case, a plant would take the stationary configuration, but a single set of maintenance/refuel components (compressor, helium, and clean salt fill tank) could still
5 be used as for the multi-unit plant. Configurations of the actual site would be dependent on user needs. The maintenance/refuel components could be located centrally or along a track system as shown below. Figure 14 illustrates an example multi-unit site configuration.

Multiple Turbine Configuration

A single turbine may not be able to operate at the full range of viable power levels. Therefore, a combination of turbines could be coupled with a single unit to provide ranges of power accommodating end-user needs. For example, a small variable load turbine could be coupled with a single turbine closely approximating the daily
15 average electricity use of the user, with the variable turbine meeting peak demand loads. A similar arrangement could apply to co-generation.

Decommissioning

Decommissioning is not addressed in depth as the federal policy and procedure for nuclear used fuel disposal may differ from the current status-quo when the reactors
20 are deployed. The salt cask serves the same function as a dry cask storage container at a LWR site, and used fuel could be transported inside the cask to a reprocessing or disposal site in the same manner. After sitting for an acceptable amount of time to reduce decay heat to negligible levels and fission product gases to background levels, the fission product gases are vented to remove any pressure in the vessel during transport. The plant
25 itself would likely be decommissioned using SAFESTOR, where the reactor vessel is allowed to sit until it qualifies as low-level waste, similar to LWR components.

A number of embodiments of the invention have been described. Nevertheless, it will be understood that various modifications may be made without departing from the

spirit and scope of the invention. Accordingly, other embodiments are within the scope of the following claims.

WHAT IS CLAIMED IS:

1. A power system comprising:

a fuel salt comprising:

between 48 and 62 mole percent NaF;

between 31 and 40 mole percent ZrF₄; and

5 between 5 and 13.2 mole percent UF₄;

a natural circulation reactor with a height between 3.25 and 4 meters and a radius between 0.5 and 1.3 meters.

2. The power system of claim 1, comprising zirconium hydride moderator rods with an H. The power system of claim Zr ratio between 1.5 and 1.7

3. The power system of claim 2, wherein the zirconium hydride moderator rods comprise cladding.

4. The power system of claim 3, wherein the cladding comprises a composite cladding

5. The power system of claim 4, wherein the composite cladding comprises a SiC-SiC composite cladding.

6. The power system of claim 1, wherein an initial actinide fuel of the fuel salt comprises of low enriched uranium with less than < 20 wt% U-235.

7. The power system of claim 6, wherein an initial actinide fuel of the fuel salt comprises of low enriched uranium with less than 10 wt% U-235

8. The power system of claim 7, wherein an initial actinide fuel of the fuel salt comprises of low enriched uranium with less than 5 wt% U-235

9. The power system of claim 1, wherein the power system that has a refueling interval of 10 years or longer.

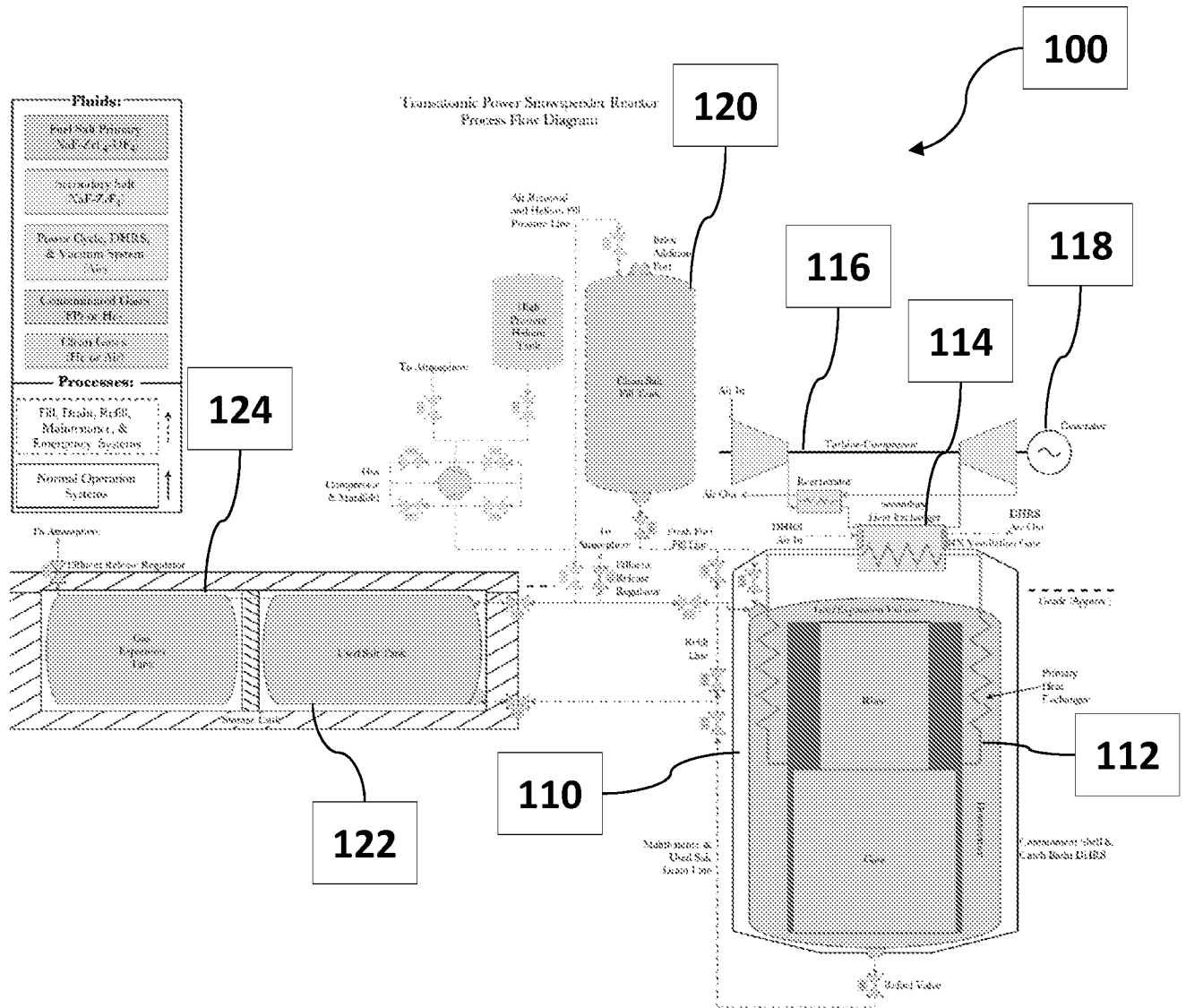


Figure 1

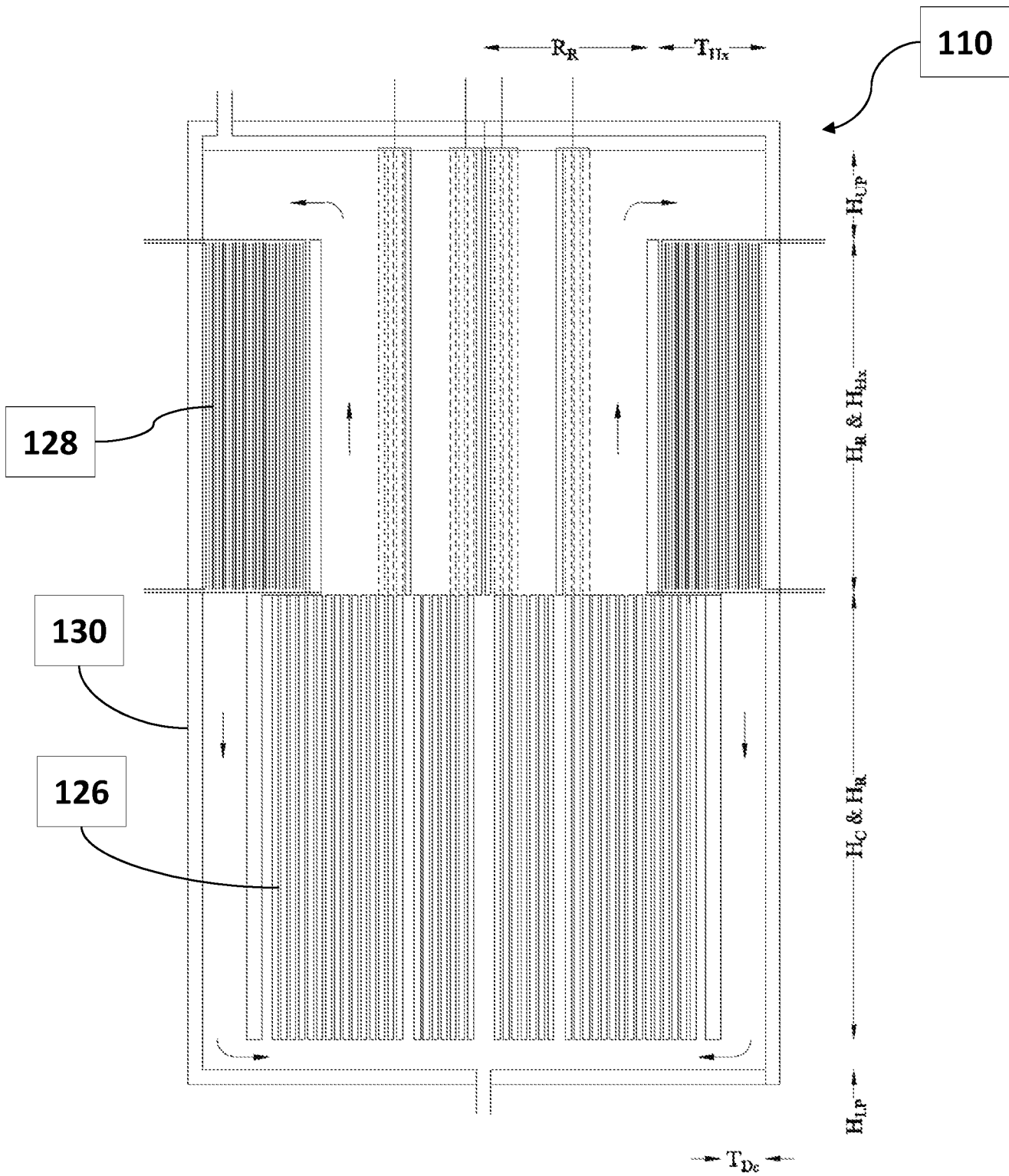


Figure 2

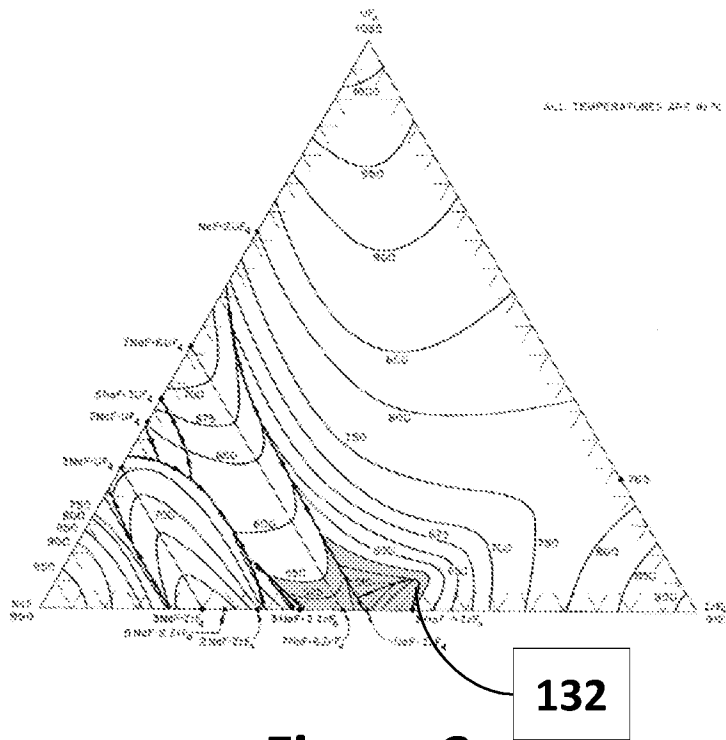


Figure 3

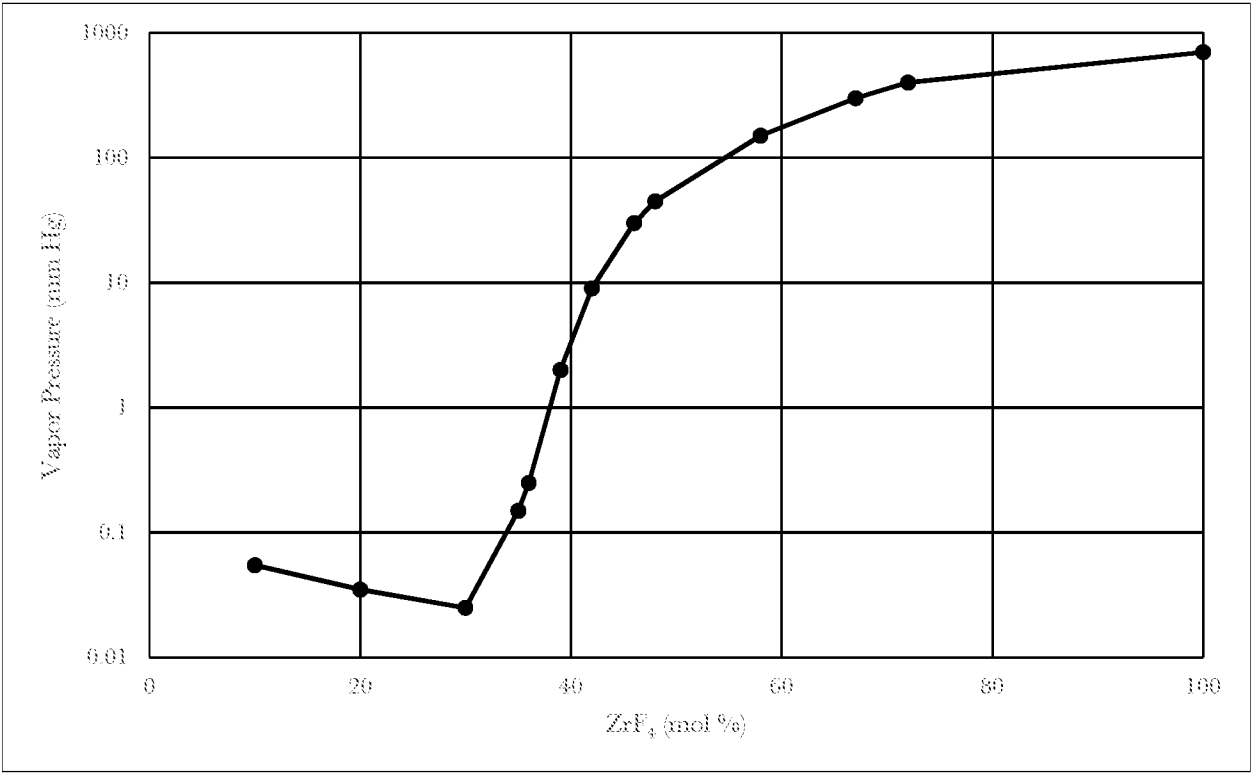
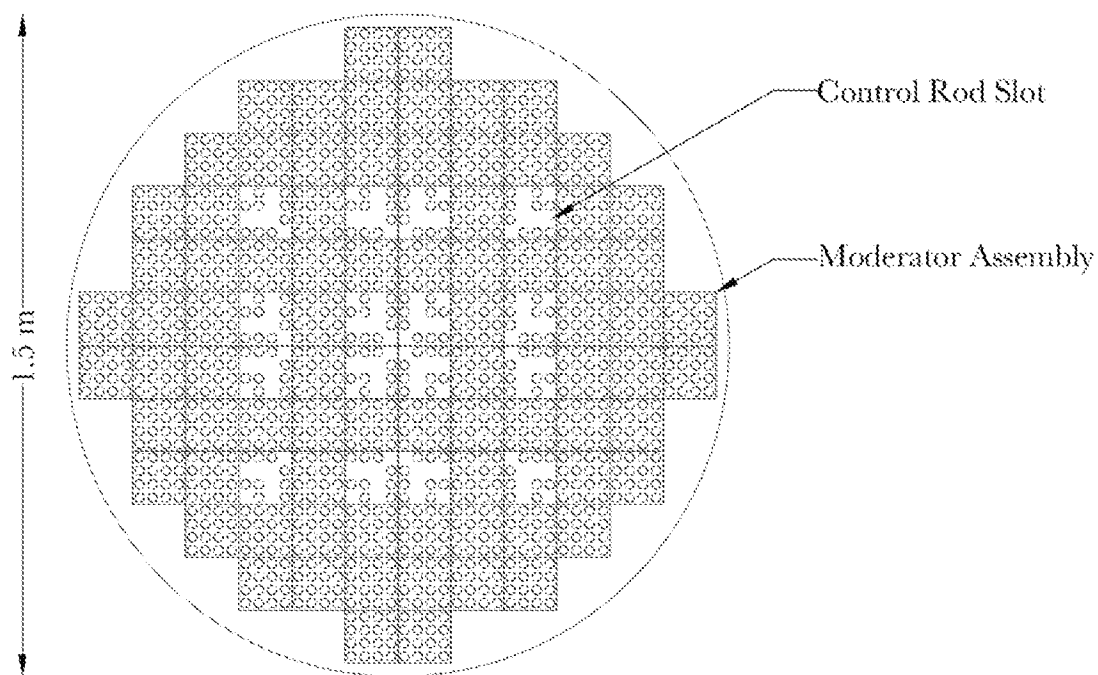
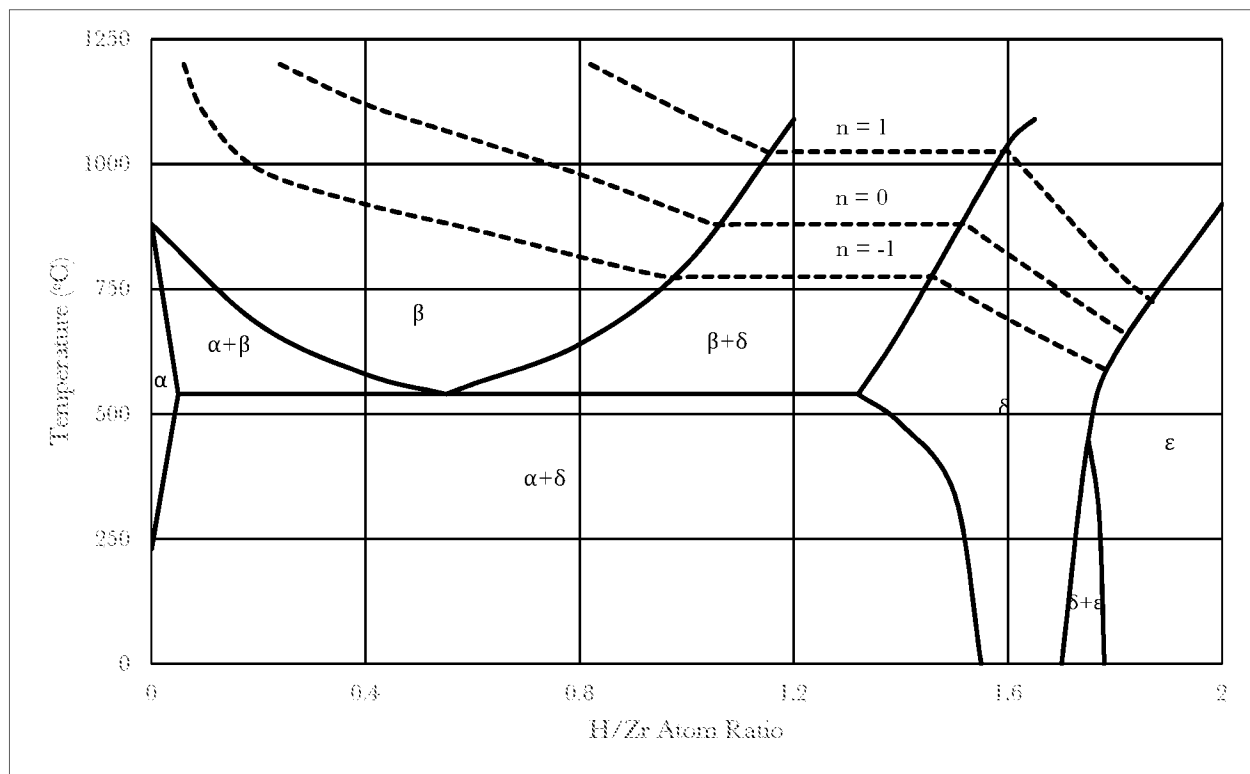
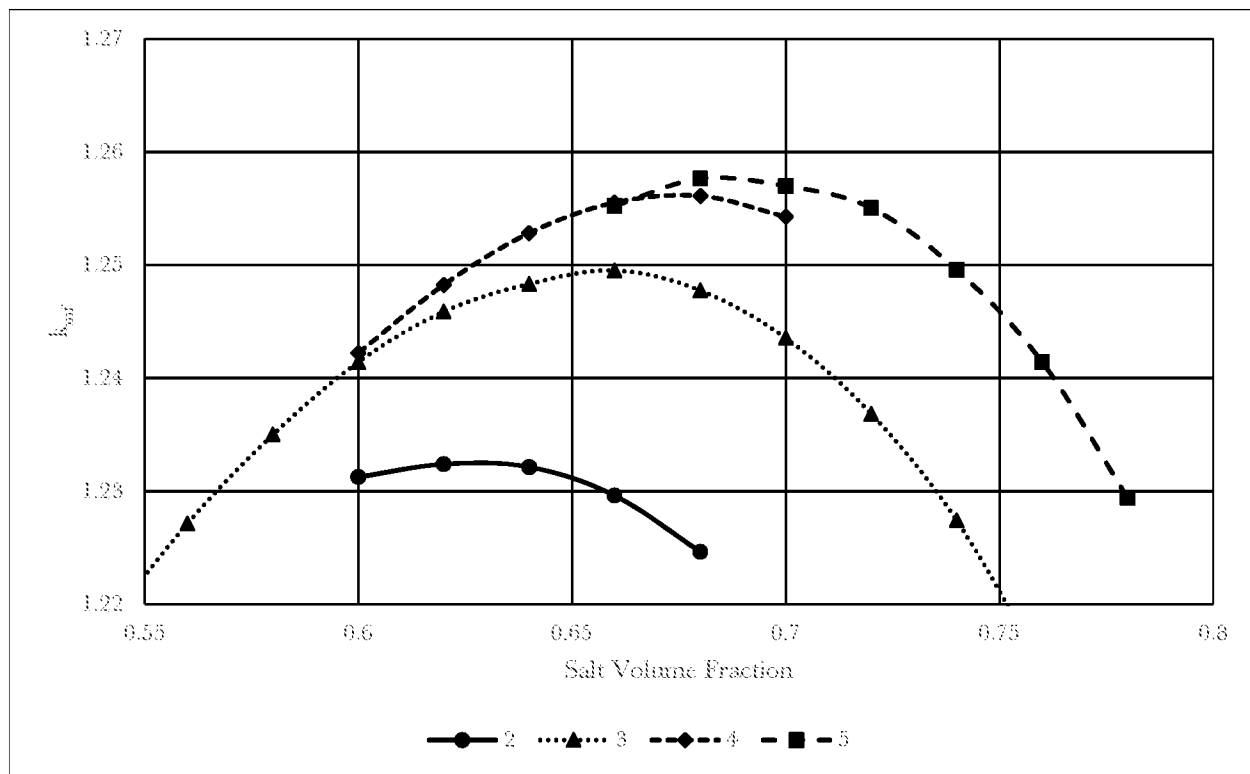
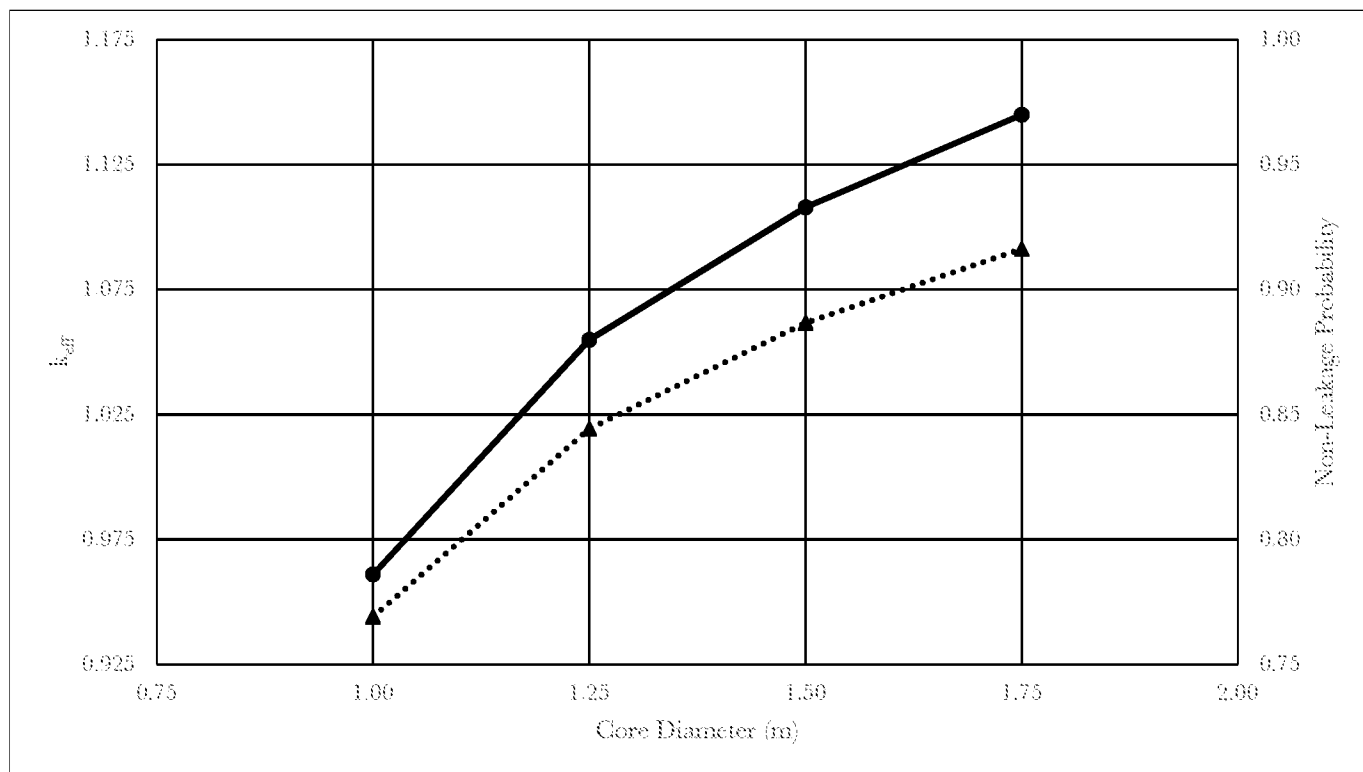
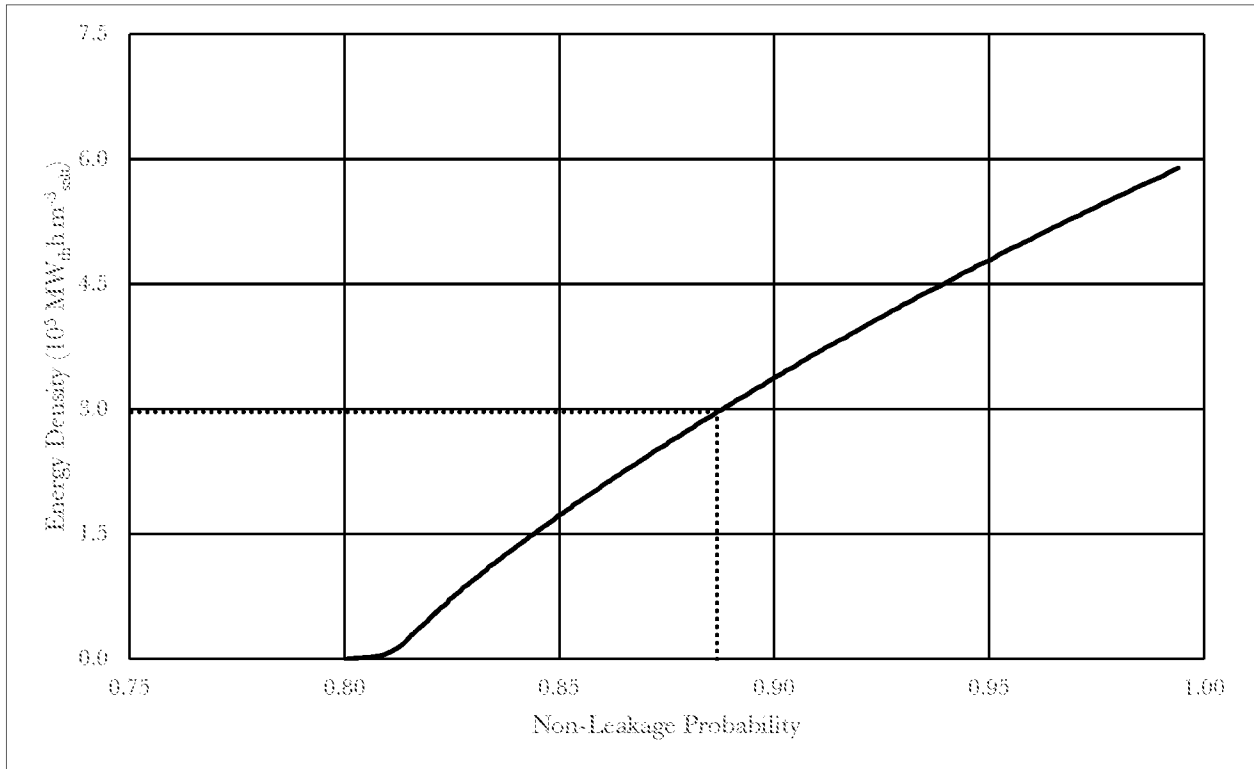
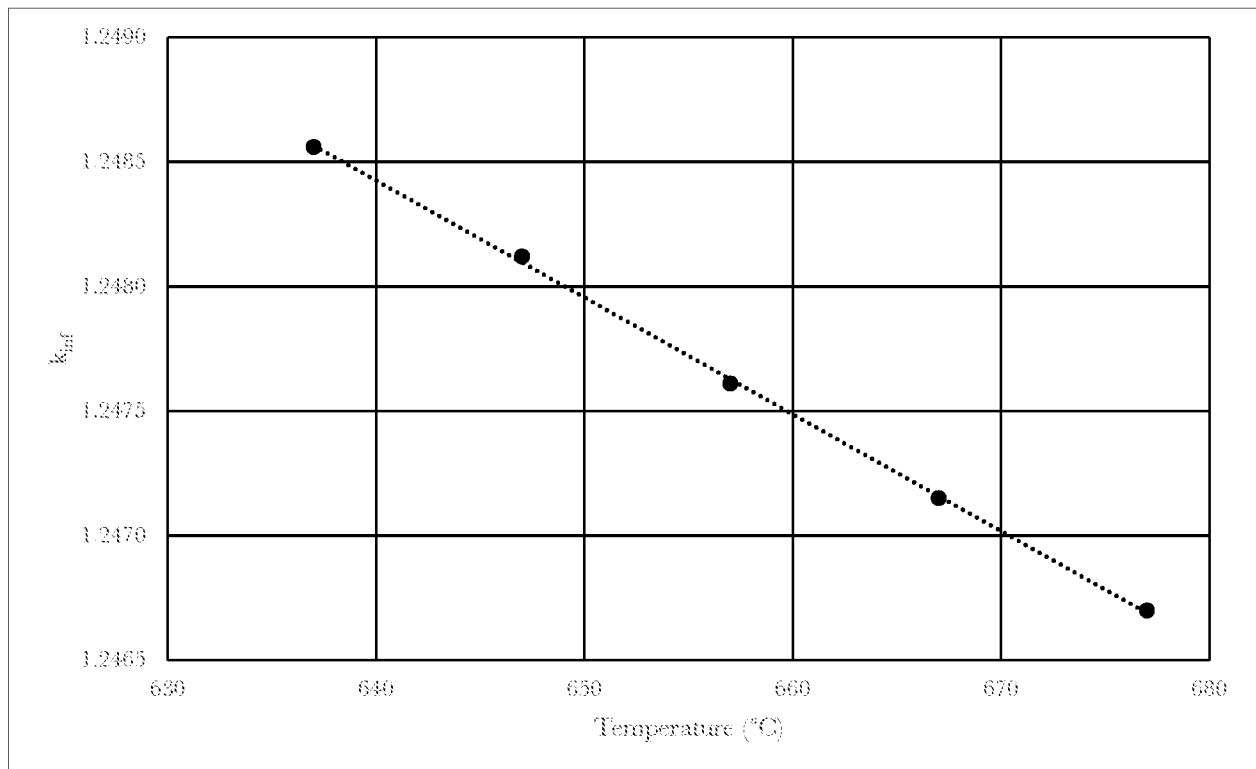


Figure 4

**Figure 5****Figure 6**

**Figure 7****Figure 8**

**Figure 9****Figure 10**

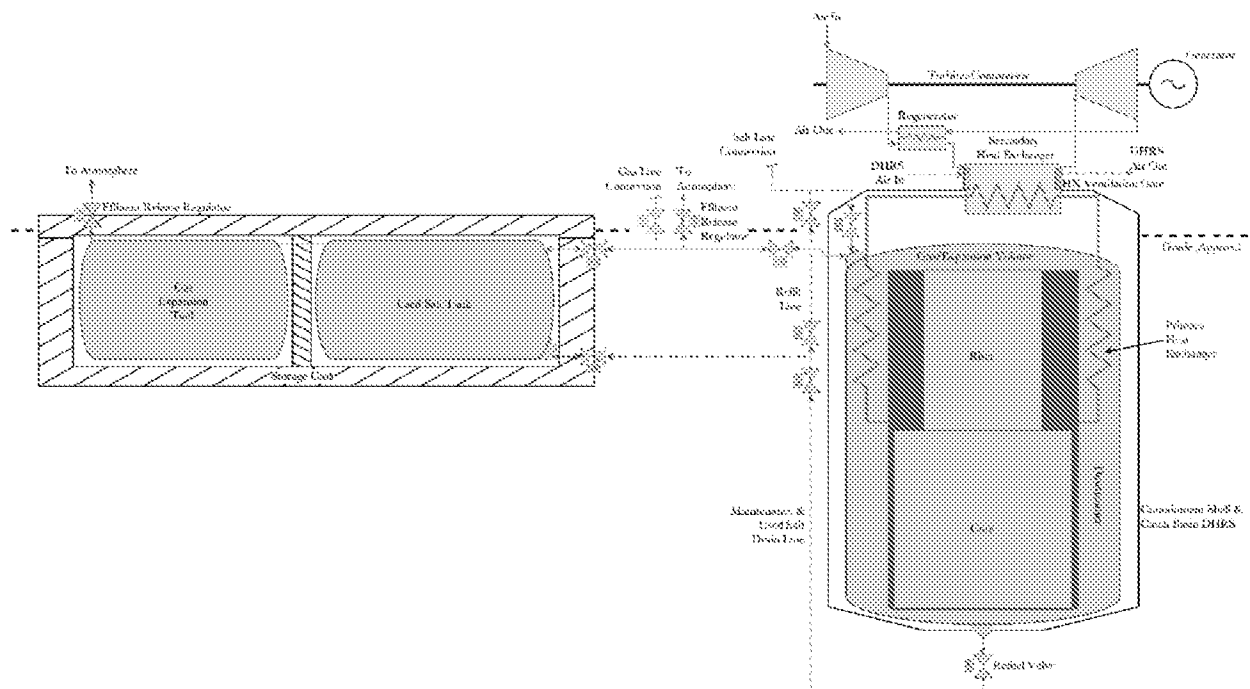


Figure 11

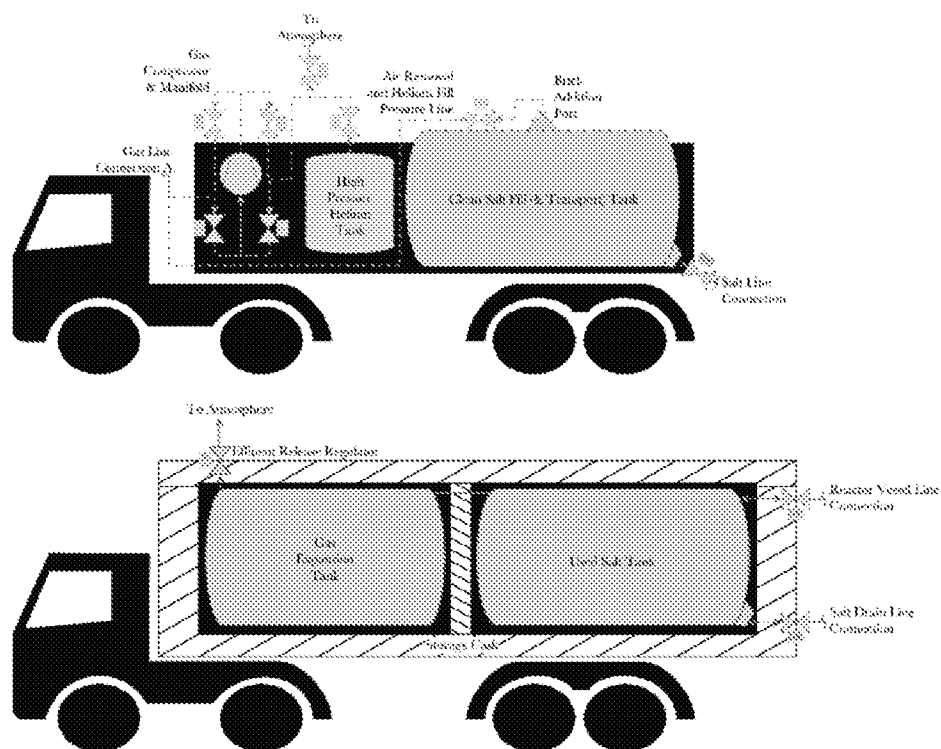
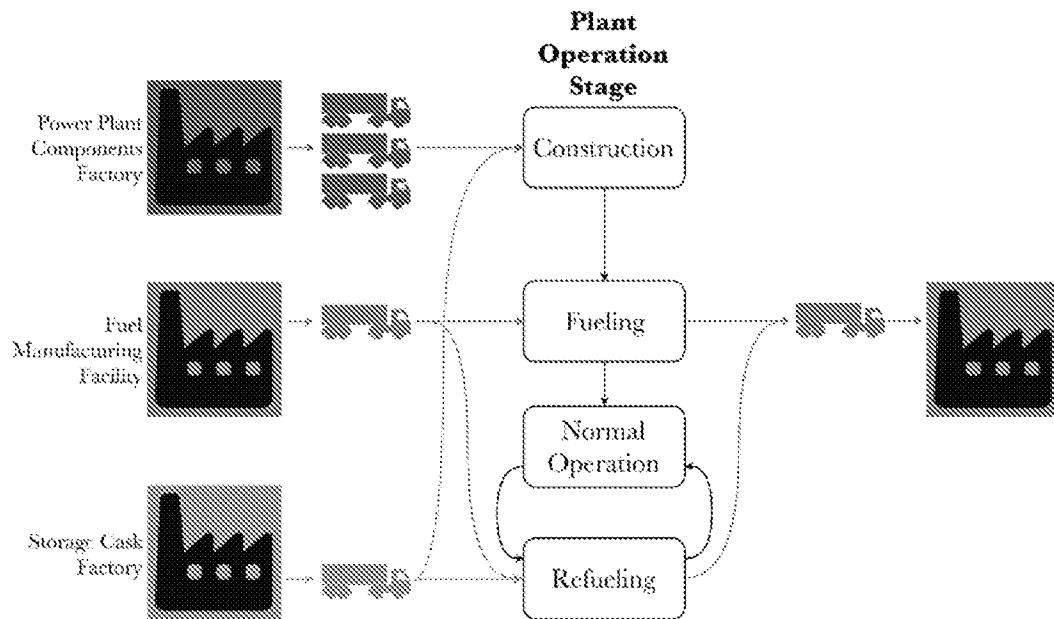
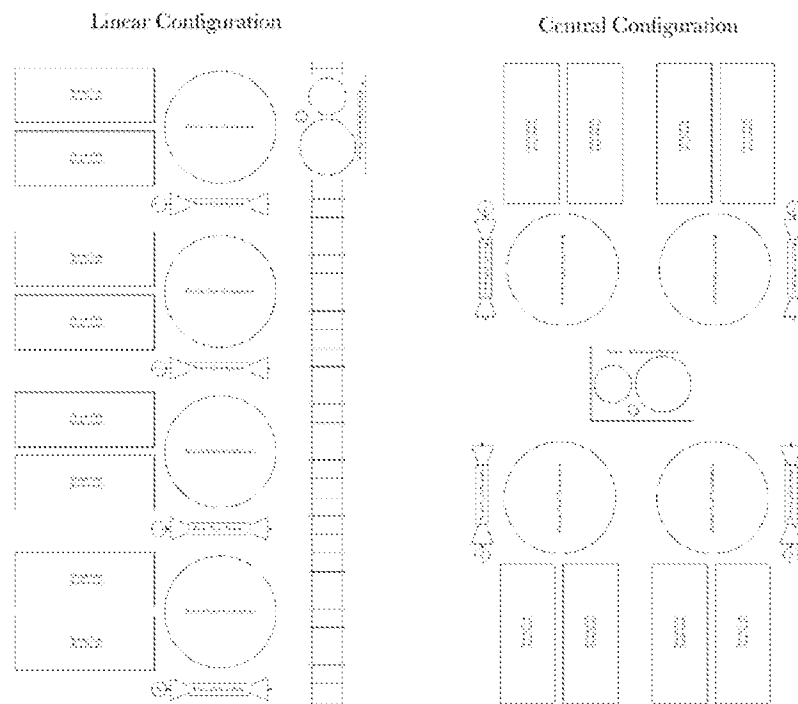


Figure 12

**Figure 13****Figure 14**