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Title: DESIGN OF MEGAWATT POWER LEVEL HEAT PIPE REACTORS

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# DESIGN OF MEGAWATT POWER LEVEL HEAT PIPE REACTORS

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## Abstract

*An important niche for nuclear energy is the need for power at remote locations removed from a reliable electrical grid. Nuclear energy has potential applications at strategic defense locations, theaters of battle, remote communities, and emergency locations. With proper safeguards, a 1 to 10-MWe (megawatt electric) mobile reactor system could provide robust, self-contained, and long-term power in any environment.*

*Heat pipe-cooled fast-spectrum nuclear reactors have been identified as a candidate for these applications. Heat pipe reactors, using alkali metal heat pipes, are perfectly suited for mobile applications because their nature is inherently simpler, smaller, and more reliable than "traditional" reactors.*

*The goal of this project was to develop a scalable conceptual design for a compact reactor and to identify scaling issues for compact heat pipe cooled reactors in general. Toward this goal two detailed concepts were developed, the first concept with more conventional materials and a power of about 2 MWe and a the second concept with less conventional materials and a power level of about 5 MWe. A series of more qualitative advanced designs were developed (with less detail) that show power levels can be pushed to approximately 30 MWe.*

## Introduction

Reactors come in a range of sizes. The size fits a variety of applications as shown in Figure 1. Los Alamos National Laboratory (LANL) has traditionally designed reactors for applications in the 1 to 200 kilowatt electric (kWe) range as shown in first two columns in Figure 1. Most of LANL's designs have been for space applications for the National Aeronautics and Space Administration (NASA.) Almost all of these reactor designs are based on a small highly reflected fast reactor concept that use heat pipes as the means of heat removal from the reactor core. This is an ideal technology for space where reliability and simplicity are key requirements.

LANL performed a study to examine the issues of scaling heat pipe reactor technology to the low megawatt electric (MWe) range (shown in third column of Figure 1.) The low MWe range is an area that was examined in the 1950s through 1970s by the U.S. Army for power at remote locations such as the Arctic, Antarctica and the Panama Canal. Power at remote locations removed from a reliable electrical grid is a potential future niche for nuclear energy. Remote locations include strategic defense locations (such pacific island bases), theaters of battle, remote communities (such as northern Alaska), and emergency locations (e.g., earthquake relief). This was, in part, the goal of the Army Nuclear power Program that ran from 1954 through 1977.

Settlements and installations at remote locations have historically relied on diesel power. An issue for diesel power is the significant logistic implications of delivering the fuel. The logistics can be expensive or involve significant risk. With proper safeguards, a 1 to 10-MWe mobile reactor system could provide robust, self-contained, and long-term power.

Heat pipe reactors are ideally suited for applications in remote areas, because they have characteristics such as self-regulation and highly reliability. A key technology for heat pipe reactors is the heat pipes used to cool the reactor core. A heat pipe transfers heat between two bodies with no temperature change from end to end (isothermally.) A schematic of a heat pipe is shown in Figure 2. The heat pipe makes use of the phase change of the fluid as it moves from boiling at one end to condensation at the other end. This ability makes heat pipes an ideal means to extract thermal power from a nuclear reactor.

Heat pipe reactors, using alkali metal heat pipes, are perfectly suited for mobile applications because their nature is inherently simpler, smaller, and more reliable than "traditional" reactors that rely on pumped coolant through the core. An example of a heat pipe reactor core is shown in Figure 3 with heat pipes protruding from the reactor core. Instead of the single point failure of a pumped loop, hundreds of heat pipes passively remove heat (including decay heat) from the core using simple and well-characterized physics. These reliability and safety advantages are especially important for remote sites. The robust, solid-state characteristics of the core are also advantageous for potentially damaging transport conditions or perhaps hostile operating environments.

The concepts of fast-spectrum reactors and heat-pipe-cooled reactors were pioneered at LANL during the 1950s and 1960s respectively. Heat-pipe-cooled reactor designs to date have had limited power output, reflecting the early intended applications for space missions. Now that compact reactors are under consideration for numerous remote applications, demand exists for safe, reliable, affordable reactors at much higher power.

This paper presents the key issues that must be addressed and the necessary engineering advances needed to scale heat piper reactors to the low MWe range.

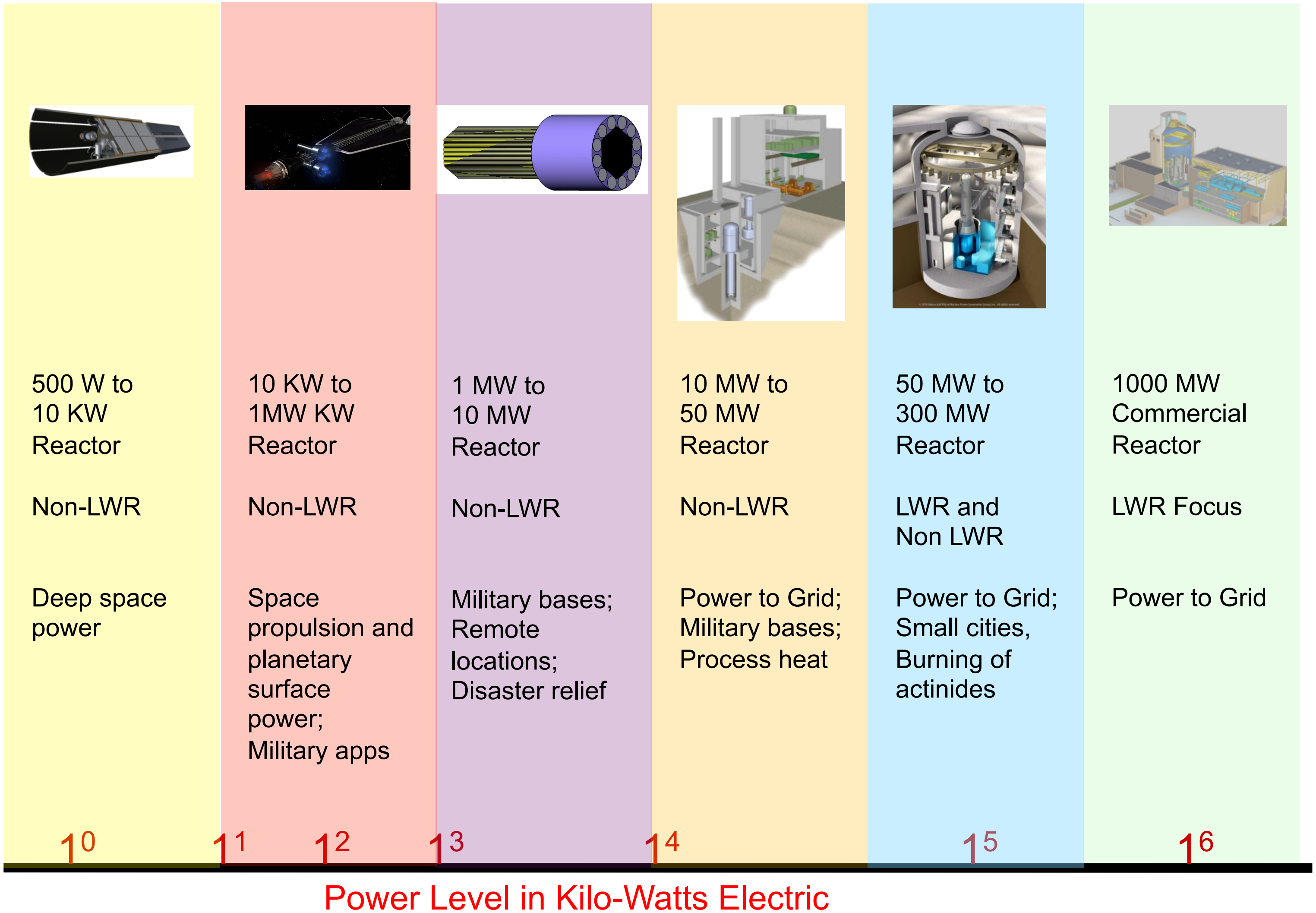


**Figure 1  Sizes Range of Reactor Applications**

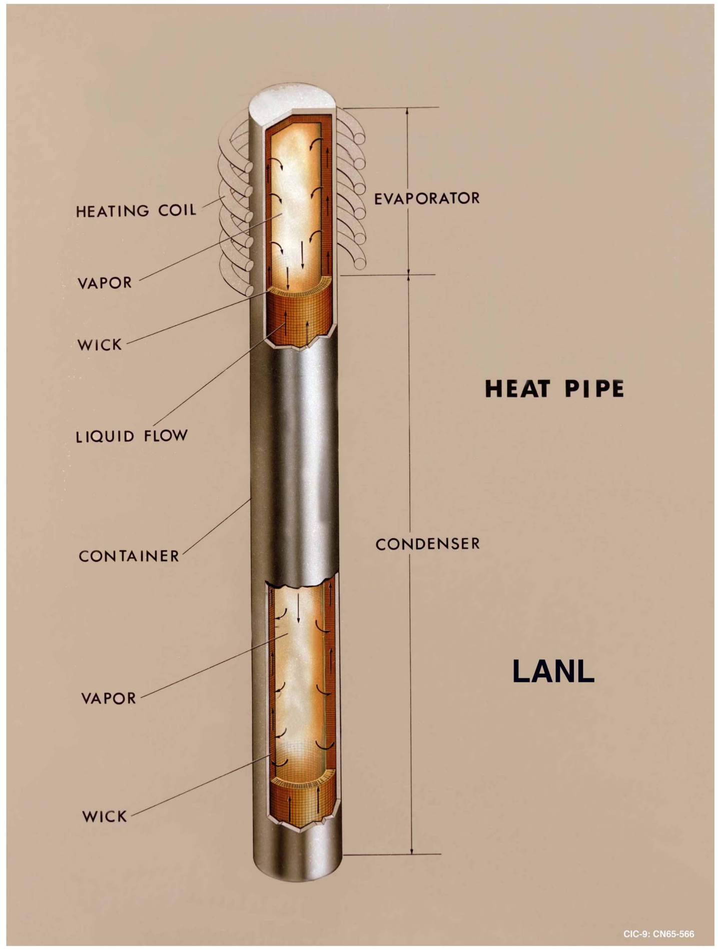


**Figure 2  A schematic of a generic heat pipe showing heated region on top and transfe region on bottom.**

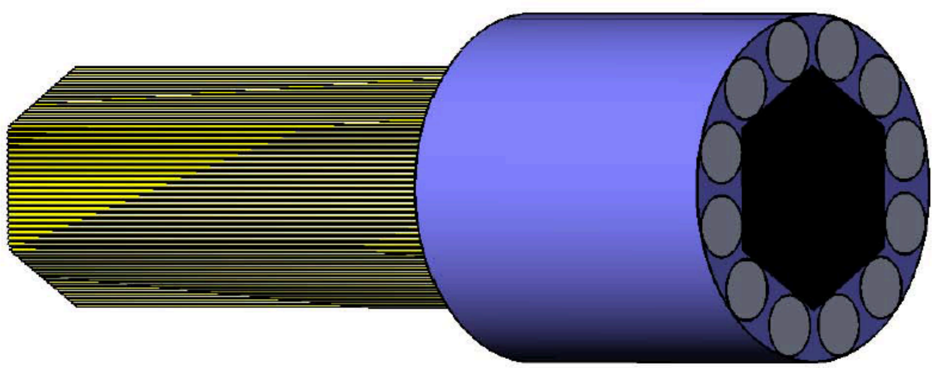

**Figure 3  A heat-pipe reactor core showing the fueled region with the control drums an heat-pipe array projecting to the left.**

## Potential Advantages

A goal of reactor design has been to develop simpler, safer, and more reliable reactors. Heat-pipe-cooled reactors have the potential to advance reactor technology toward this goal. Heat-pipe-cooled reactors are nearly "solid-state" and avoid many of the complexities and issues arising from the traditional reactor concepts that rely on a coolant pumped through the reactor core.

A heat-pipe reactor is typically a solid-block core with the fuel in holes inside the solid block. Heat pipes are built into the block in a lattice configuration (depicted in Figure 4 and Figure 5). The heat pipes remove the heat from the block as the liquid in the heat pipe is vaporized. The heat is deposited in the condenser region of the heat pipe. The condenser region can be sized to accommodate multiple heat exchangers, such as one for power conversion and two for redundant decay heat removal.

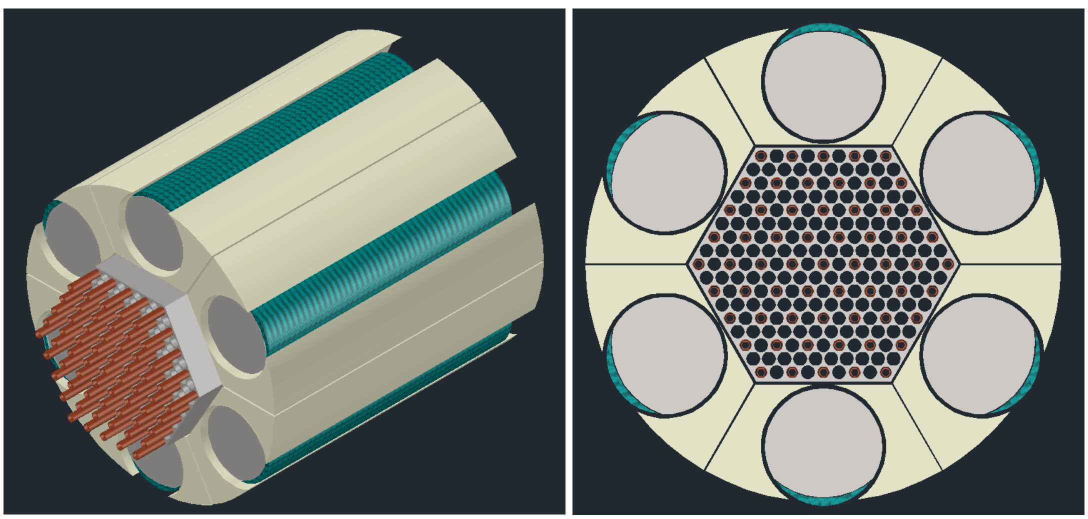

**Figure 4 Heat pipe reactor core side view and end view showing monolithic block, fuel, heat pipes, reflector and control drums**

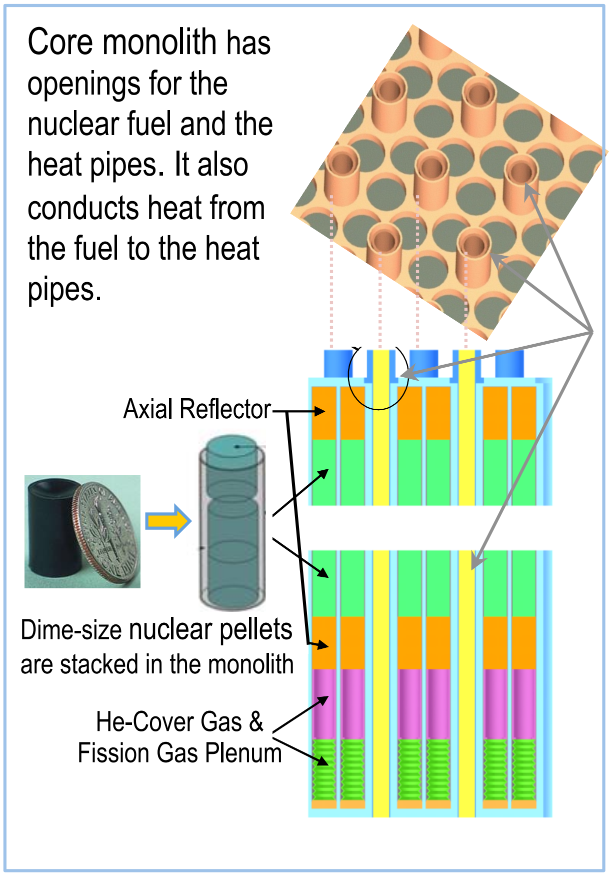


**Figure 5 View of heat pipes and fuel in a solid core monolithic block**

The potential advantages of heat pipe reactors are:

- Size
- Orientation
- Safety
- Self-regulation
- Solid state
- Surface area outside core
- Choice of fluids
- High temperatures

Each is described in detail in the following paragraphs.

## Size

Heat pipe reactors can be physically smaller than other advanced reactor concepts. Enrichment of the fuel to near 20%, the use of a fast neutron spectrum and the use of a highly reflected core allow for a very small reactor core size and weight. Size and weight are important considerations when deciding on transportable or mobile technologies.

## Orientation

If a reactor is to be mobile and transportable, issues with reactor orientation must be kept to a minimum. Heat pipe reactors have less of an issue with orientation than reactors with a liquid that is pumped and must maintain a level in the reactor core to remain cool. Heat pipes will remove heat in any orientation, (although peak performance may be impacted.) This issue is also tied to the type of decay heat removal system designed for the system. However, having the ability of the heat pipes to effectively remove heat in any orientation is a must for safe transportation.

## Safety

In most (if not all) existing reactor concepts, a single reactor coolant is the only the means of getting heat out of the core. Safety is achieved by preventing the set of failures that could lose the fluid, cause it to not circulate, or lose its heat transfer capabilities (e.g. transition from nucleate to film boiling in water cooled reactors). Preventing failure is typically achieved using redundant equipment (pumps, electrical, etc.) or passive components.

In a heat pipe reactor, an array of heat pipes are used to remove heat from the core using simple, reliable and well-characterized physics (capillarity, boiling & condensation.) In a heat pipe reactor, unless common cause failures dominate, the failure of multiple heat pipes will be much lower that the failure rate associated with a coolant system. The decay heat removal can be made passive (much like other advanced reactor designs.) This implies that traditional measures of safety (like core damage frequency) could be much better for a heat pipe reactor. It must be stressed that any final measure of safety would be dependent on the specific reactor system analyzed.

Another safety feature of heat pipe reactors is the solid monolithic core. This feature prevents voids in the core (as might occur in a core with liquids if the liquid is boiled.) This eliminates issues with positive reactivity being introduced by voids in the core.

In addition heat pipe reactors are at ambient pressure inside the core. This again eliminates issues with high pressure as might occur in a high-pressure system such as a gas-cooled reactor design. Depressurization accidents are a major concern for high-pressure systems.

Finally, given that that there are no pumps or valves in vessel/core area (as one would find in a water or liquid metal cooled reactor) general overall reliability and safety are improved. Particularly if the emergency decay heat removal in the heat pipe reactor is passive.

## Self Regulation (Load Following)

One of the key features of small highly reflected fast reactors is the simple and predictable reactivity feedback mechanism that allows for the reactor to be load following. Fast reactors in this size range are controlled by thermal expansion and subsequent negative reactivity feedback.   Thermal feedback will lower the reactor power if less heat is extracted by the power conversion system.  This makes the system more tolerant of power conversion failures.  It also allows for above-rated power extraction if needed (within thermal limits of fuel system.)

## Solid state

Heat pipe reactors are near solid state (i.e. no pumps, no valves, etc.)  Other than control rods (or drums), moving parts can be limited to power conversion system. The implication is that a near-solid state reactor could potentially be more reliable than reactor designs with many moving parts.

## Heat Transfer Surface Area Moved Outside Core

The goal of any reactor concept is to get the heat out of the reactor, transfer it to a working fluid and then use the fluid to do work and do it SAFELY.  Most power-producing reactors pass a fluid (gas, liquid metal, water) through the reactor core. Most (except a gas cooled reactor) then transfer the heat to a working fluid, (gas cooled reactor transfer heat directly to the working fluid.)

A heat pipe reactor uses heat pipes (evaporation and condensation) to remove heat from the core.  Essentially heat pipes are a means of expanding the area available for heat transfer and moving that area outside of the reactor core region.  This makes it possible to use multiple separate means of removing heat.  The concept is shown in Figure 6.  In this configuration one heat exchanger is used for the working fluid that produces energy and a second heat exchanger is used to remove decay heat in emergency conditions.  This second heat exchanger can use a different fluid than the heat exchanger used to do work.  This configuration moves the heat transfer to the working fluid outside of the core.  This means choice of working fluids does not impact the radiation transport inside the core.  This is a great simplification for reactor design and control.

In addition, a cycle such as an open-air Brayton system is available as an attractive option for power conversion.  Since the air would pass through the heat pipe heat exchanger and not the reactor core, the issue of the air becoming activated is removed.  An open-air Brayton cycle would not need a second system for removing residual heat in the thermodynamic cycle.  This greatly simplifies the reactor system.

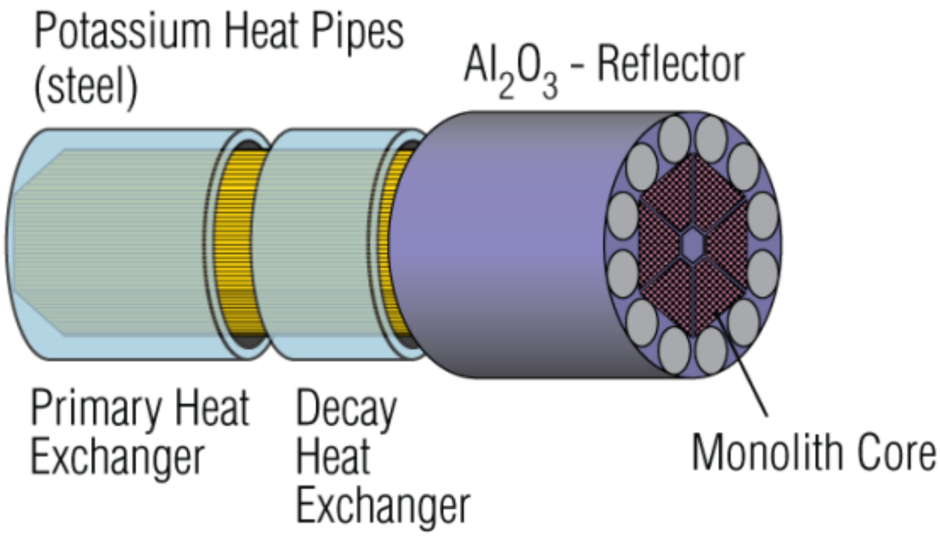


**Figure 6 Heat pipe reactor with primary (working fluid) heat exchanger and decay heat exchanger for safety**

### More Choice of Fluids And Configurations

As talked about in the previous topic, an advantage of a heat pipe reactor is that no fluid flows through core. This allows for more choices of working fluid because the impact to neutron transport (absorbing neutrons, etc.) does not occur. In addition, because there is not large amount of circulating fluid, corrosion is less of an issue than in some reactor configurations. All of this allows for more options in configuring the reactor.

### High Temperatures

Heat pipe reactors can be made with temperatures that range from 650 C to over 1000 C depending on the choice of alkali metal used in the heat pipes (the boiling point determines the temperature of the reactor). This feature allows the reactor to be designed for delivering a working fluid at a high temperature and thereby extend the range of applications and improve on power conversion efficiency.

### Summary of Advantages

In summary, heat pipe reactors have many advantages that make them an ideal choice for a megawatt power mobile reactor. These advantages are why scaling reactors is important. In the next section, issues with scaling are addressed.

## Issues on scaling

To date, most proposed heat-pipe reactors were designed to far less than 1 MW electric in power. This is, in some part, a result of historical applications for space nuclear power, where typical power needs were in the 10's to 100's of kilowatts electric range. The goal of this study was to examine the issues to achieve power levels that are much greater than 1 MW electric. Several issues must be overcome to scale heat-pipe reactors to this size. The key issues that must be addressed and the necessary engineering advances to overcome these issues are presented in this section.

### Limitations On Scaling

To scale a heat pipe-reactor to megawatt power levels, several limitations must be addressed.

- Limits on the number of heat pipes in a single block,
- Limits on design imposed by the design-basis accident conditions,
- Limits on heat-pipe performance (thermal throughput),
- Limits on thermal and mechanical performance imposed by selected materials,
- Other limits imposed by material used to make core (including fuel choice and choice of material for monolithic block).

Each limitation is discussed in more detail below.

### Limits on Number of Heat Pipes

The number of heat pipes in a reactor block can be a potential impediment to scaling a heat pipe reactor. Depending on the type of working fluid in the heat-pipe design, most heat pipes have a limit on heat throughput per heat pipe. Given this, the larger the reactor, the more heat pipes required. However, there is a limit on the practical number of heat pipes that could be realistically manufactured into a solid-block core. (Note, later an alternative core configuration will be postulated that could address this issue.)

Based on experience in space reactor applications this number is set at several hundred (~200) heat pipes per block. This is a soft number and is really a function of the manufacturing technique used. Advances in the 3-D printing of metal represents an example that may easily overcome this issue. This limit is based on current manufacturing technologies and an estimate of the probability of error per heat pipe that would require the block to be discarded.

### Limits Based on Accident Conditions

A heat-pipe reactor must be designed to survive accident conditions. Although no comprehensive safety assessment of a heat-pipe reactor has been performed to date, sufficient work has been done to assess the likely mode of failure of a heat-pipe-cooled reactor. From this work, the accident of greatest concern is the failure of one or more heat pipes leading to cascade failure. "Cascade failure" is the loss of a single or multiple heat pipes, causing the failure of surrounding heat pipes in a cascade (like dominoes falling in a line). If a large enough set of heat pipes fail, the situation could lead to inadequate cooling of the reactor core. Cascade failure is assumed to be the principal design-basis accident for a heat-pipe reactor. Cascade failure is mitigated by designing the reactor for the failure of one or more adjacent heat pipes for the limiting (highest power) fuel pin. LANL designs consider two adjacent heat-pipe failures to occur and not impact operations. Three adjacent heat pipe failures (a delta configuration) are designed not to cause accident conditions but would require the shutdown of the reactor. More than three adjacent heat pipe failures surrounding a single fuel pin is considered a beyond design basis event. This condition is based on the calculated failure probability of a heat pipe being less than 0.5%. The thermal and mechanical conditions produced by the accident are used to determine limits on performance.

## Limits on Heat-Pipe Performance

A heat pipe is a closed solid pipe with an internal shape (a wick) that imposes order on a wetting saturated liquid. This saturated liquid forms vapor at the heated end of the pipe (the evaporator.) This vapor moves toward a cooled zone (the condenser) and condenses, depositing its heat of vaporization with a small attendant temperature change. Capillary action between the liquid and the wick draws the condensate back toward the evaporator, completing the circuit. The heat-pipe wick shape imposes order on a saturated liquid by (1) forming menisci between the condensate and the vapor and (2) allowing condensate to flow toward the evaporator.

Heat pipes have a limit on the heat transfer rate per unit area on a per-heat-pipe basis. These limits are shown in Figure 7 and are based on the following physical issues in a heat pipe.

- The sonic limit is the power level where vapor approaches the sonic velocity at the evaporator exit.
- The capillary limit is the power level that produces mass flow rates sufficient for liquid and vapor pressure drops to exceed the maximum capillary head potential of the wick.
- The entrainment limit is the power level at which counter-flowing vapor sweeps liquid out of the wick, depriving the evaporator of returning liquid.
- The boiling limit is the power level at which incipient boiling of liquid occurs at the superheated wall. The wick may impede radial movement of bubbles from the heated wall leading to local drying and loss of heat transfer capability.

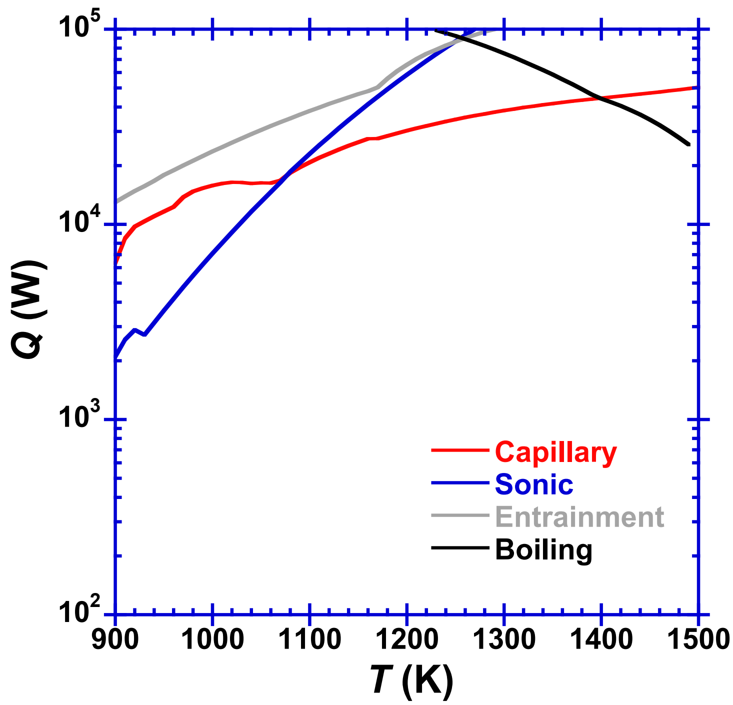


**Figure 7 Typical heat-pipe axial performance limits versus evaporator exit temperature for a 1.5-cm-i.d. x 4-m-long sodium heat pipe.**

The heat-pipe performance limits determine the type of working fluid and heat-pipe geometry that are available for the reactor design. The selection of the working fluid in the heat pipe sets the working temperature of the reactor core. For example, the selection of sodium versus potassium as the working fluid sets the core temperature at ~1200 K and ~950 K respectively. The choice of working fluid of the heat pipe and its size also determine the heat throughput per heat pipe, which, in combination with the number of heat pipes, helps determine the maximum output of the reactor.

### Limits on Thermal and Mechanical Performance

For the design-basis case of two heat-pipe failures adjacent to a fuel pin, the solid-core block is designed to not exceed the ASME limits for thermal stress for the given material in the reactor core block. An example of the thermal analysis (the increased temperature in a fuel pin and block) that is used to determine the stress in the solid-block core is shown in Figure 8. In this figure a maximum temperature criterion is set for the block. In the case of stainless steel this might be 1200 K. The resulting strain in the block is shown in Figure 9. In this figure a total elongation of less than 1% is an example of the criterion used to determine an acceptable design. These calculations inform the designer of the ability of the block to meet ASME mechanical limits and other design requirements.

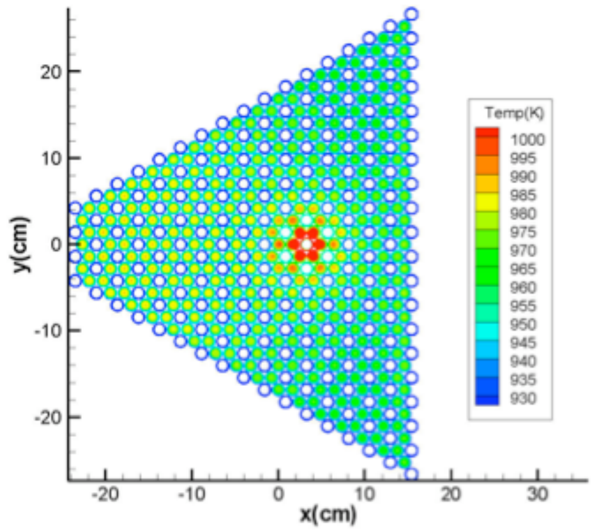


**Figure 8 Temperature distribution of the 5 MWt heat pipe reactor for a failed heat pipe.**

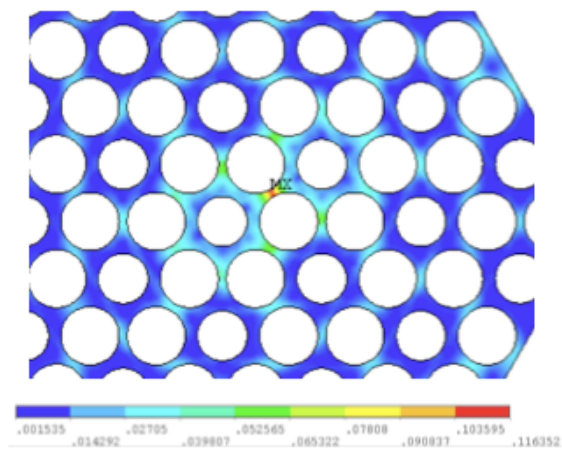

**Figure 9 Strain analysis of a failed heat pipe in the monolithic block.**

## Other Material Limitations

The reactor working temperature helps determine the range of available materials. As was shown in the previous section, after a material is selected, design basis accident calculations are used to analyze the thermal stress on the reactor block during normal and accident conditions to ensure the design is feasible.

The preferred option then to deal with issues of thermal stress and heat-pipe performance is to select materials that optimize solutions to these issues. Then the new materials are examined to see if new problems are introduced.  For example, many materials have great properties but may not be compatible with oxygen or high-radiation environments (i.e. new materials need to be resistant to radiation damage.) The new materials are also examined for an impact on neutron transport. Materials have to perform well over a range of conditions in order to be considered as a solution to scaling heat pipe reactors.

## Overcoming Limits on Scaling

This section presents potential solutions to the issues previously presented. Since these issues are connected any potential solution could be iterative. Meaning, that the solution may solve one issue but fail to solve other issues. Each potential solution set is presented in the same order as before.

### Core segmentation

The main issue to be solved is the ability manufacture the solid core block. A simple solution to this issue is to simply breaking the core into smaller segments. A heat-pipe reactor can be broken into segments that are mechanically and thermally isolated but are neutronically connected. An example of this configuration is shown in Figure 10.

By limiting the segments to a size corresponding the perceived level of manufacturability (assumed to be approximately 200 heat pipes), the reactor core can be scaled to the necessary size.

A second solution would be to not have a solid monolithic block. A stagnate liquid core is another approach to solving this problem. This of course introduces new issues (now a “Loss of Coolant” like accident must be considered.) But this eliminates manufacturing issues and solves other issues such as temperature-induced stress in the block.

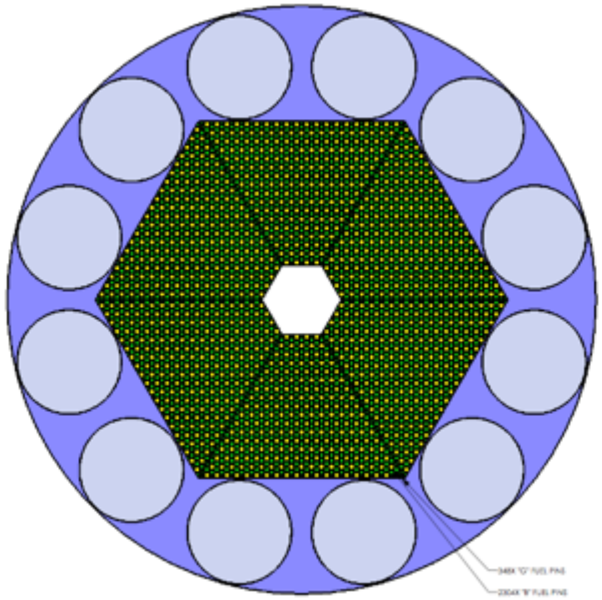

**Figure 10 A heat-pipe reactor core showing fueled region in six pie-shaped segments, surrounded by reflector and control drums.**

## Heat Pipe Performance – The Use of a Double-Ended Heat Pipe

One potential way to increase the performance limits of heat pipes is to introduce a double-ended heat pipe into the core. An example of a double-ended heat pipe reactor is shown in Figure 11

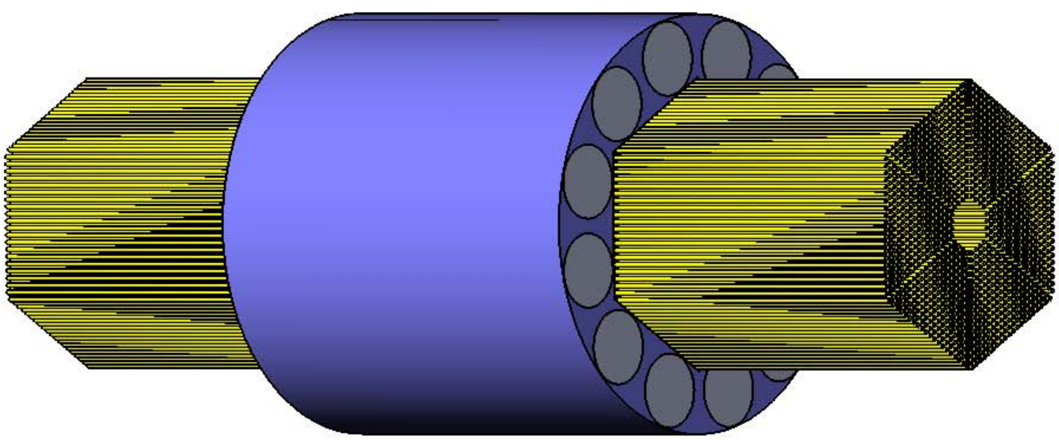

**Figure 11 Reactor concept with double ended heat pipes**

A double-ended heat pipe has the ability to improve the performance limits because it introduces two opposing condenser configurations (cooled zones on either side of a single heated zone) into the reactor system.

The dual opposing condenser configuration permits a heat pipe of given length to operate at higher viscous, sonic, capillary, and entrainment limits than simpler evaporator-condenser arrangement. Heat pipes with dual opposing condensers currently cool computer server circuit boards. The dual heat pipe configuration with uniform heating and cooling enhances capillary and viscous limits by up to a factor of four. This effect is shown in Figure 12. In the analysis it will be assumed that use of the double-ended heat pipe can increase the throughput on energy by a factor of two.

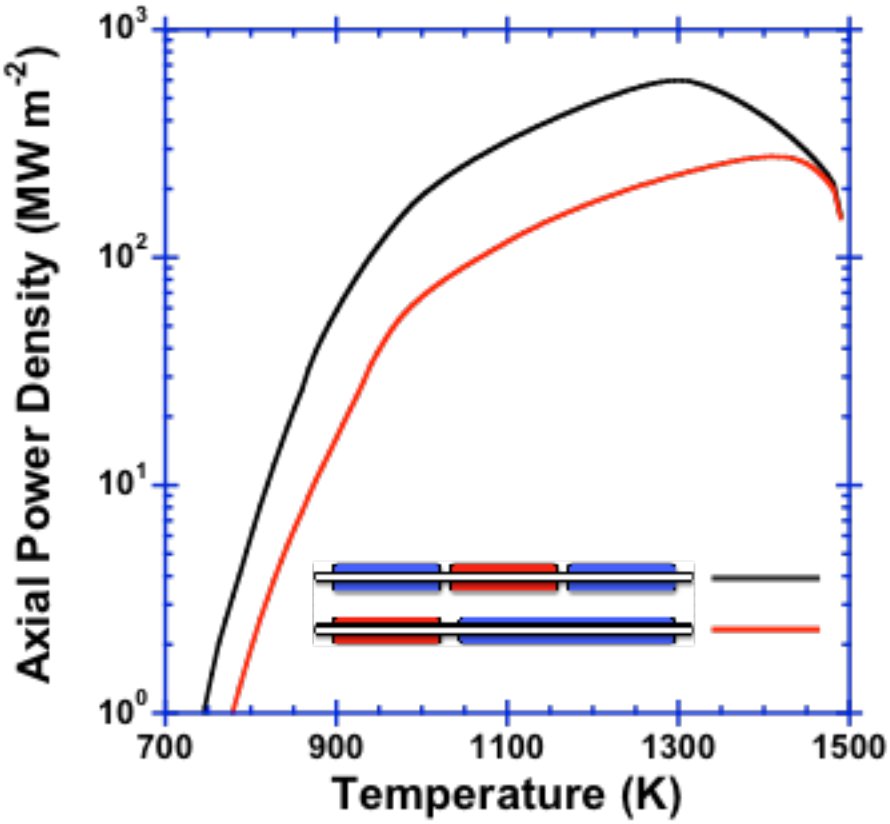


**Figure 12 Change in power density for a single end heat pipe (red) and a double ended heat pipe (black)**

## Overcoming thermal/mechanical/neutronic issues for normal and accident conditions

The response of the reactor design to normal and accident conditions (i.e. the cascade failure of the heat pipes) involves trade-offs in the thermal/mechanical/neutronic behavior of the combinations of choices for the block material and fuel.

The trade-offs for fuel involve neutronic impacts such as the amount of fuel, neutron spectrum, neutron economy, lifetime of core and thermal performance. The choice of block materials involves trade-offs in thermal conductivity of the block and its mechanical properties for stress and strain. The manufacturing issues with the choice of a block material are also of concern.

Overcoming the limits involves trade-offs between neutronic performance, thermal performance and mechanical performance. For this study, several combinations of fuel and monolithic block were considered.

To begin the study two materials were chosen for very detailed study. These two configurations involve using a metal for the monolithic block with fuel in traditional pin configurations, where the cladding is the monolithic block wall. The goal behind these choices was a simple reactor configuration with known materials (Stainless with UO2) and a more advanced configuration with a less conventional fuel and materials, however each with more favorable properties. The two configurations were (abbreviation of concept shown in bold):

- Stainless steel with Uranium oxide fuel pellets (**SS_UO2**)
- Molybdenum alloy (TZM) with Uranium Nitride fuel pellets (**Moly_UN**).

A qualitative study was the next step in the process. The qualitative study allowed for more materials, fuel and geometry combinations to be examined. In the next sections, the detailed study is presented first, followed by the qualitative study.

## Detailed Analysis of Block Designs (SS_UO2 and Moly_UN)

The starting point of the detailed analysis was the two reactor point designs. Both designs have been generated that used the same general core geometry. The first concept, deemed the "SS_UO2" concept, utilizes 19.75% enriched uranium-oxide ($UO_2$) fuel and stainless-steel (SS-316) structure, with potassium (K) heat pipes operating at 677°C. The second concept, deemed the "Moly_UN" concept, utilizes 19.75% enriches uranium-nitride (UN) fuel and molybdenum (Mo) structure, with sodium (Na) heat pipes operating at 977°C. Each reactor uses an alumina ($Al_2O_3$) reflector, with 12 embedded control drums that contain an arc of boron-carbide ($B_4C$) poison. The only significant geometric differences between the cores are that the Moly_UN concept has a longer fueled length, a larger fission gas plenum, and a thicker radial reflector. The method used to perform this analysis is presented in Appendix A.

A mechanical design was not developed in detail for this reactor system, so the design is discussed here in very broad terms. In the proposed concept the core is fabricated in six identical segments, as shown in Figure 13. Each segment contains

204 heat pipes and 352 fuel pins, all housed in a solid metal block. Heat pipes are placed along the entire periphery of the block to reduce the impact of failed heat pipes on the boundary. In the reference approach, the holes in the core block serve as the evaporator wall of the heat pipes, and the rest of the heat pipe is welded to the top of the block (alternatively, a full heat pipe could be brazed or hipped to the block). The six triangular sections are truncated at the center to create a hex-shaped void in the core, which is used for a $B_4C$ shutdown/safety region.

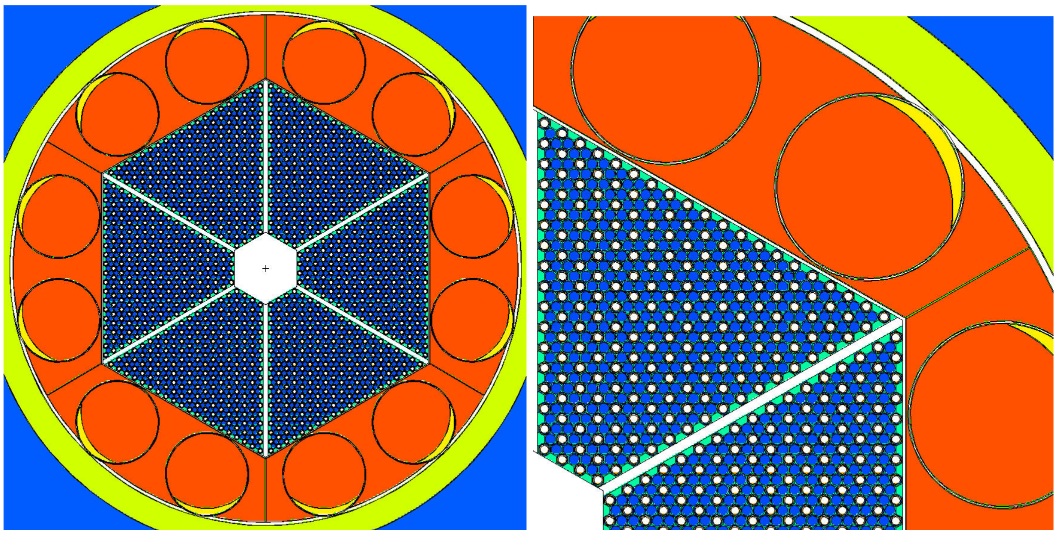

**Figure 13 Core radiation transport schematic, Core (blue), reflector (orange), green is core block, blue is fuel, white circles are heat pipes, orange is alumina, yellow is boron carbide.**

As currently envisioned, each of the reactor segments has its own heat exchanger. Each segment with its heat exchanger, reflector and shield is fabricated, assembled and shipped separately to the reactor site. They are then assembled into the reactor system, and the secondary sides of the heat exchangers are connected to piping linking them to the power conversion system. A simple schematic of the reactor, shield and heat exchanger is shown in Figure 14.

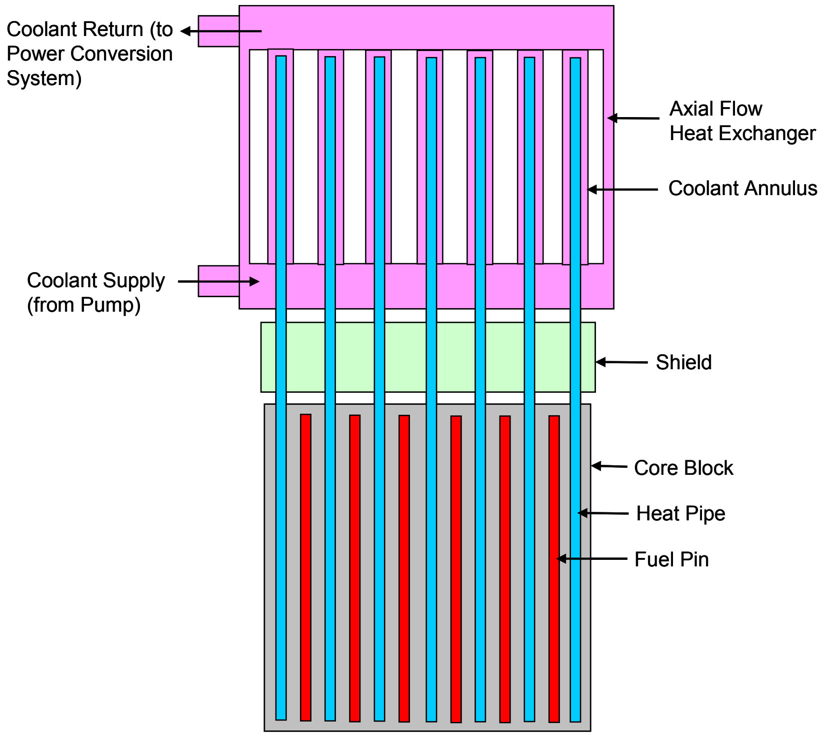


**Figure 14 Core to heat exchanger schematic**

As shown in the figure, the heat pipes extend from the core, through the upper shield and into the heat exchanger. The heat exchanger has an axial flow configuration. The coolant enters into a plenum at the bottom, flows up through annuli formed by the heat pipes and the walls of the heat exchanger, into an upper plenum and exits from a single pipe there. The advantages of the axial flow heat exchanger are that the length and annulus area can be selected to produce a desired pressure drop and a temperature difference between the coolant exit and heat pipe as small as desired. Also, the core can be easily orificed by adjusting the flow areas for the individual flow annuli. This is especially important for the heat pipes located at the edges of the blocks, which receive significantly reduced power.

Each of the concepts was designed to meet the operational and safety criticality requirements. At end-of-life (EOL), the warm reactor has a $k_{eff}$ of 1.01 with the drums turned to their most reactive position. The central cavity $B_4C$ poison brings cold, beginning-of-life (BOL) $k_{eff}$ to <0.95 (the cold/EOL value is even lower). The control drums have the capability to bring the cold/BOL $k_{eff}$ to <0.98, with no poison in the central cavity. Initial reactivity coefficient calculations indicate the all core materials (e.g. fuel, structure, reflector) have a consistent, negative reactivity temperature coefficient (RTC). Nearly all of the reactivity feedback is rooted in material expansion, with only a slight impact of Doppler broadening of cross sections.

Three-dimensional power and fluence distributions were calculated and then passed to the TMSS analysis tool described in Appendix A.  TMSS tool processes the MCNP calculated cell powers to produce pin-average power time histories as well as axial power distributions for each fuel pin.  An example of the core power distribution, for the SS_UO2 design option at BOL is shown in Figure 15 (pin average peaking factors) and Figure 16 (time-average core average axial power distribution).

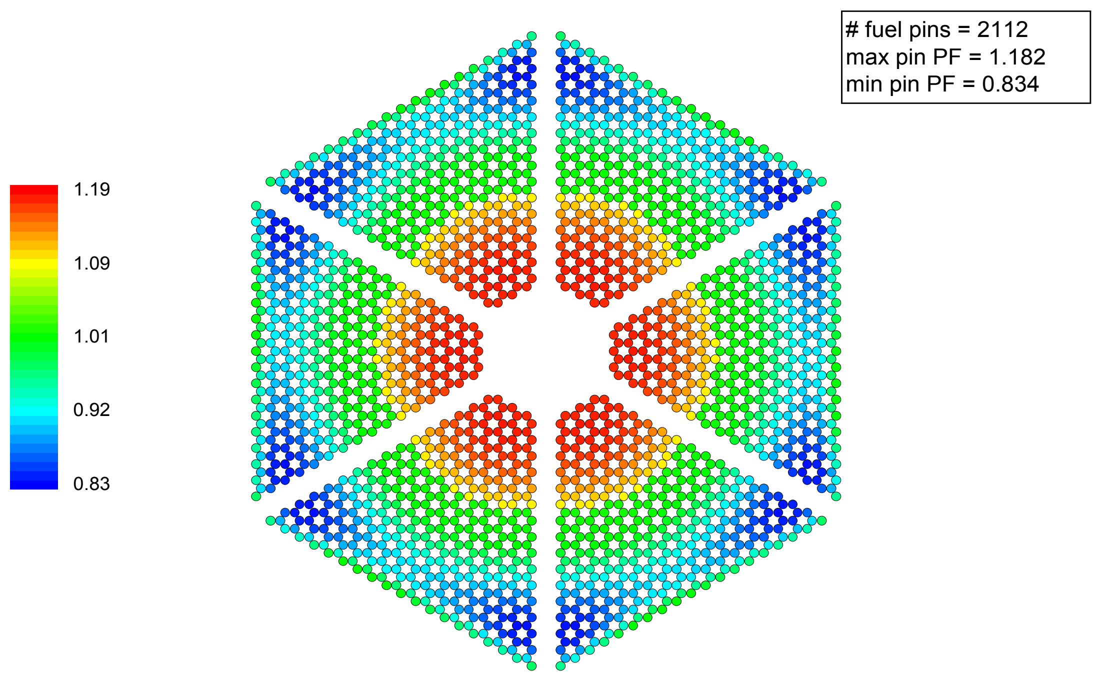


**Figure 15  Fuel pin peaking factors at beginning of life.**

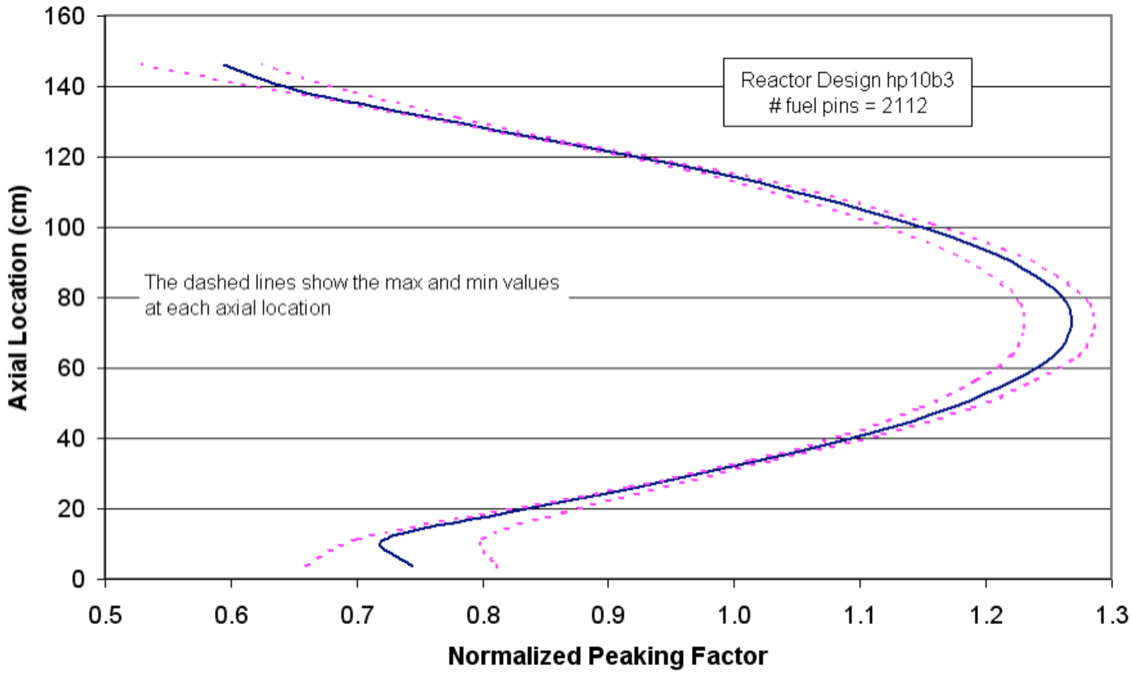


**Figure 16. Time and core average axial power distribution.**

As can be seen in these figures the power radial power distribution is relatively flat, and it will tend to flatten with burnup. There is a somewhat larger axial distribution, and the axial distribution varies very little from pin to pin. This is not surprising for a small, fast reactor that does not use in-core control rods.

A peak-power pin core axial temperature distribution for the SS_UO2 design is shown in Figure 17. The heat pipe vapor temperature is very-nearly isothermal. The largest temperature rise takes place in the fuel-clad gap and in the fuel itself. Further, there is an increase in the peak fuel temperature from BOL to EOL of about 30°C, because the pin gas thermal conductivity more than offsets the reduction in gap width due to fuel pin swelling (the fuel pin is designed to have a positive gap at EOL). The temperature rise from the heat pipe vapor to the inner wall of the fuel clad is relatively small, about 20°C total. For this reason, the block thermal stresses and creep strains are very small for normal operation.

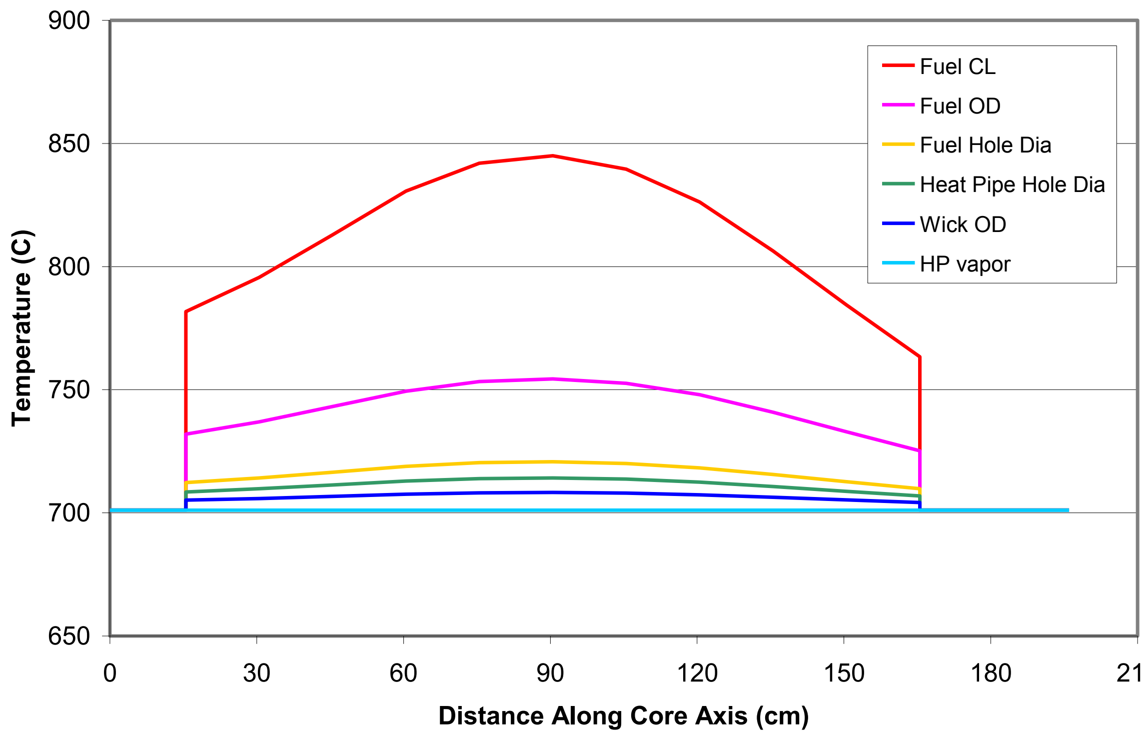


**Figure 17 Peak power pin location temperatures.**

Temperatures in the heat exchanger at the peak power pin location are given in Figure 17. Here also the vapor temperature is near-constant. Most of the temperature rise takes place in the coolant film. In an axial flow heat exchanger, the coolant temperature approaches the heat pipe temperature at the exit end. This heat exchanger was designed to produce a 25$^{o}$C difference between the coolant exit and heat pipe vapor temperatures for a nominal power heat pipe. For the high-power heat pipe, the temperature difference is about 35$^{o}$C.

A 3-D heat transfer analysis was performed to determine the temperatures within one of the six core blocks, using the power depositions from MCNP. Figure 18 shows the nominal core temperatures in one core segment and Figure 19 shows the temperatures for a central failed heat pipe.

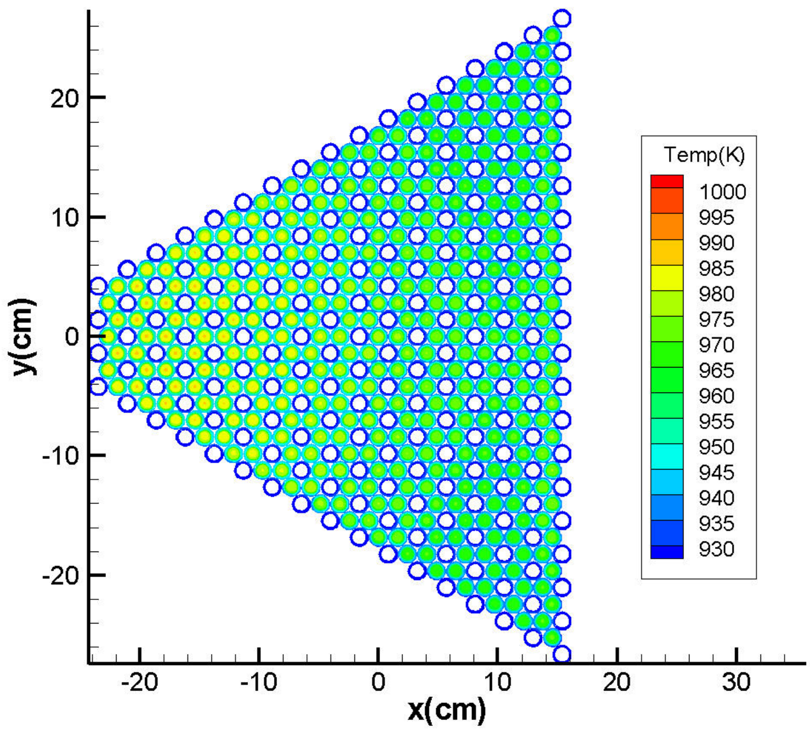


**Figure 18 Temperatures at nominal operating condition.**

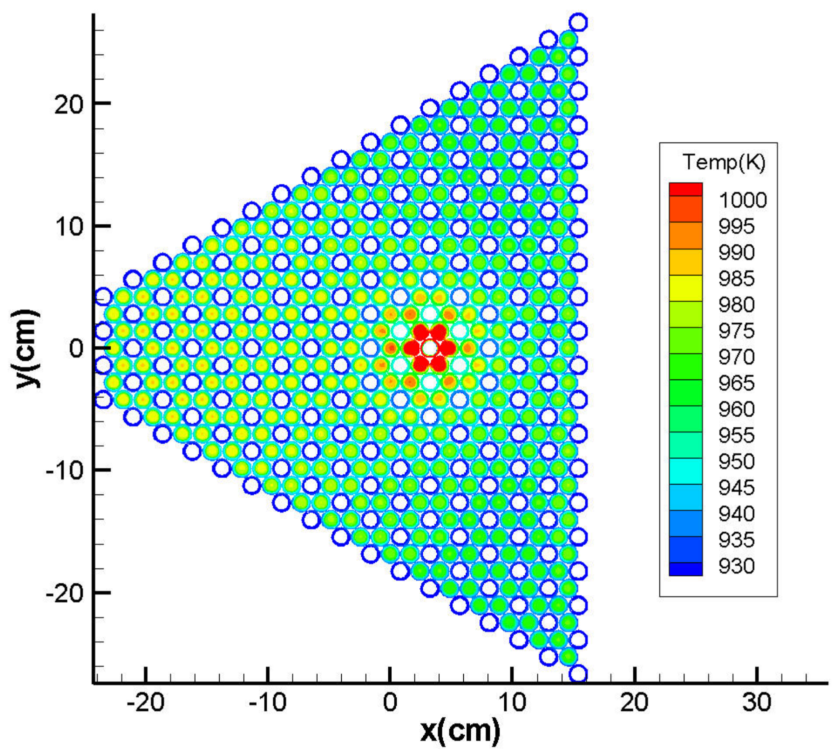


**Figure 19 Temperatures with centralized failed heat pipe.**

Note in Figure 19 that the failed heat pipe provides a very local temperature peak, but there is some conduction beyond the immediate vicinity. In general, the throughput of a heat pipe nest to a failed heat pipe goes up about 12%. In the case of 2 adjacent heat pipe failures, the throughput of an adjacent working heat pipe can increase from approximately 25% to 35% depending on the location of the pin in the core. A central failed heat pipe is more limiting that one near the edge, because of the outer row of heat pipes surrounding the entire block.

Table 1 provides a list of comparative performance parameters for the SS_UO2 and Moly_UN design concepts. The two designs have the same number of fuel pins and heat pipes. The fueled height for the Moly_UN concept is somewhat longer. Because the core power in the Moly_UN concept is three times greater, the fuel pin linear heat rates and heat pipe powers are much larger for the Moly_UN Concept design. For the Moly_UN concept design heat exchanger, the coolant pressure was increased from 200 psia to 300 psia, the pressure drop was increased from 5 psi to 12 psi, and the temperature difference between the coolant and heat pipe was increased from 25$^{o}$C to 50$^{o}$C. Nevertheless the heat exchanger length increased from 76 cm to 101 cm.

$CO_2$ was assumed as the heat exchanger coolant for both concepts. It is likely that $CO_2$ would not be acceptable for the Moly_UN Concept design, because of corrosion concerns at the proposed operating temperatures. An inert gas mix, such as He-Xe may be required, with some degradation in performance.

**Table 1 SS_UO2 and Moly_UN Concept Performance**

| Performance Parameter | Near-Term Design | Advanced Design |
|---|---|---|
| Reactor power (MW[t]) | 5.0 | 15.0 |
| Block | SS316L | TZM |
| Fuel | UO2 | UN |
| Core lifetime (EFPY) | 5 | 5 |
| # Fuel pins | 2112 | 2112 |
| # Heat pipes pins | 1224 | 1224 |
| Fuel height (cm) | 150 | 175 |
| Uranium mass (mT) | 2.61 | 3.92 |
| Max cell linear heat (kW/m) | 4.03 | 10.37 |
| Max cell fuel burnup (%) | 0.56 | 2.14 |
| Max cell fuel swelling (%) | 0.38 | 6.60 |
| Max pin fuel FGR (%) | 6.29 | 6.02 |
| Max T-fuel CL ($^{o}$C) | 876 | 1264 |
| Max T-core block (oC) | 721 | 1027 |
| Heat pipe working fluid | K | Na |
| Heat pipe working fluid temp ($^{o}$C) | 677 | 977 |
| Max heat pipe power (kW) | 5.59 | 16.78 |
| Max heat pipe vapor press (psia) | 7.94 | 40.43 |
| Heat exchanger coolant | CO2 | CO2 |
| Heat exchanger coolant T-in ($^{o}$C) | 327 | 602 |
| Heat exchanger coolant T-out ($^{o}$C) | 652 | 927 |
| Heat exchanger coolant P-in (psia) | 200 | 300 |
| Heat exchanger coolant DP (psi) | 5.0 | 12.0 |
| Heat exchanger condenser length (cm) | 76.1 | 101.2 |

A finite element calculation is required to calculate the temperatures, stresses and creep in the core blocks. A simplified method is included in TMSS tool to do these calculations, which has proved to be reasonably accurate for the case where there are no failed heat pipes. These calculations showed that for both concepts the stresses and lifetime creep strains are low for a condition where there are no failed heat pipes. As discussed earlier, failed heat pipe calculations performed for earlier heat pipe reactor concepts showed that acceptable stresses and creep strains were obtained for a maximum of two adjacent failed heat pipes. However, these reactor concepts were lower power than those proposed in this study. Similar calculations have not been performed for this design.

Finally, there are many factors that allow the Moly_UN concept to produce 15 $MW_{th}$ while the SS_UO2 concept only produces 5 $MW_{th}$.

1. The higher uranium loading of UN fuel versus $UO_2$ fuel allows more excess reactivity, thus the ability to handle 3-times the burnup reactivity loss.

2. The higher thermal conductivity of TZM versus SS-316 increases heat transfer through the block, lowering temperatures and thermal gradients, especially in failed heat pipe scenarios.

3. The higher conductivity of UN fuel versus $UO_2$ fuel keeps the fuel temperature gradients lower, but more importantly aids in heat transfer through the core in failed heat pipe scenarios (the heat can use the wider path through the fuel, instead of the thin path though the web of the block)

4. The lower coefficient of thermal expansion of TZM versus SS-316 creates less thermal stress and resulting strain.

5. The higher temperature of the Moly_UN concept allows the use a Na, a much more effective heat pipe working fluid than K, plus the vapor pressure of Na at 977°C allows very high throughput.

## Qualitative trade offs of Alternative Designs

For the remaining design alternatives, the analysis done next was not to the level of detail presented earlier. Instead, a quantitative analysis has been done to examine the performance of each design alternative. This was done partly because the results of neutronic analysis versus thermal and mechanical performance cannot be compared directly to one another. So, a simple qualitative approach has been develop where each design is ranked ordered relative to one another in terms of performance in each of the analytical categories in an attempt to define how each alternative performs overall.

There were four geometries used in the design of the alternatives reactor concepts. These four were:

1. Monolith Design – Heat pipe wall and fuel cladding are provided by the block
2. HP and Block Design – Core block serves as fuel clad, by heat pipe wall is separate material
3. Tube and Block Design – The fuel cladding and heat pipe wall are different material than the block. This also works for a pool type design were the block is liquid.

4. HP and Composite Design – The fuel is dispersed in the block (so no cladding is used) and the heat pipe wall is a different material than the block.

Each of these designed is shown in the same order in Figure 20, Figure 21, and Figure 22.

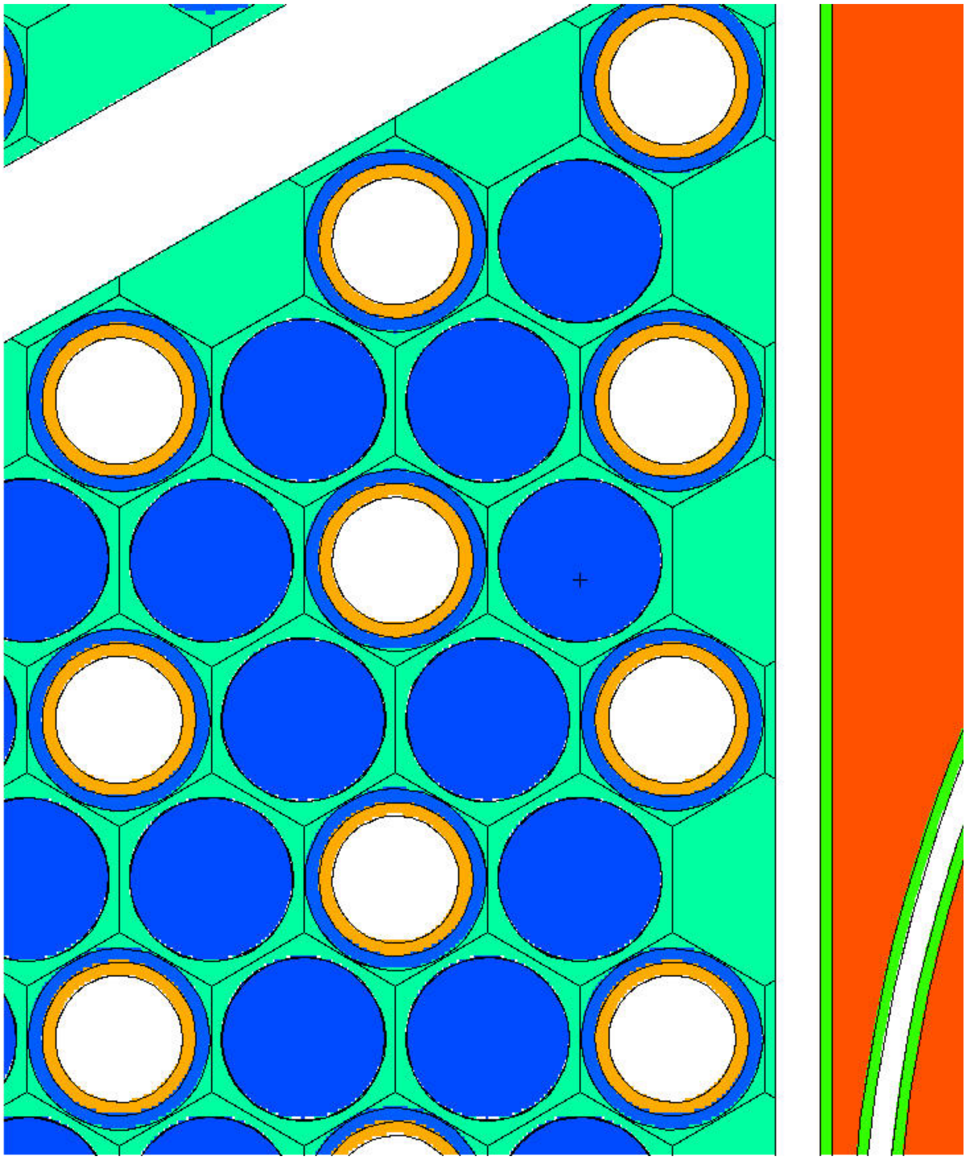

**Figure 20 Monolithic or HP and block configuration**

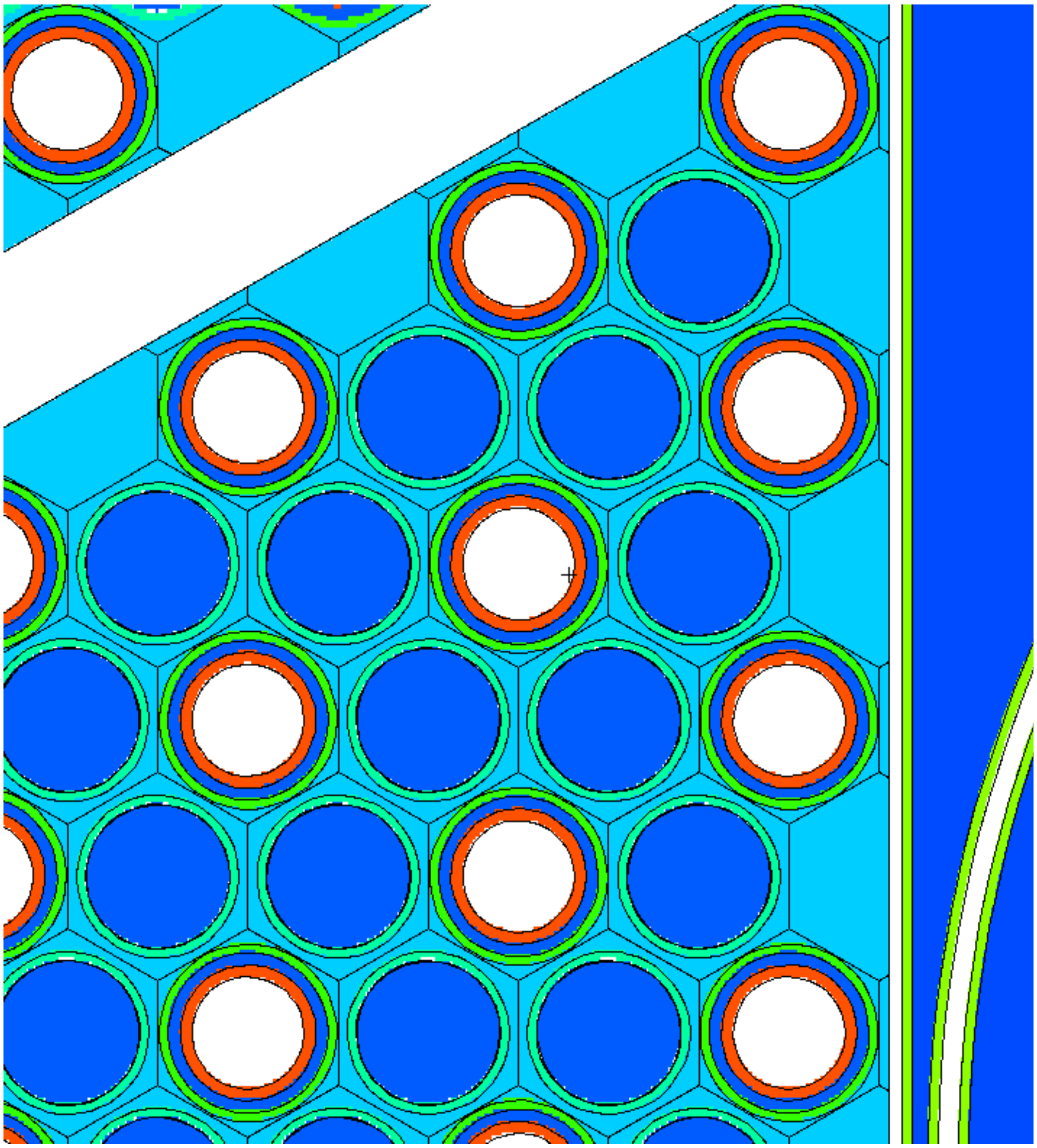

**Figure 21 Tube and block (also pool) configuration**

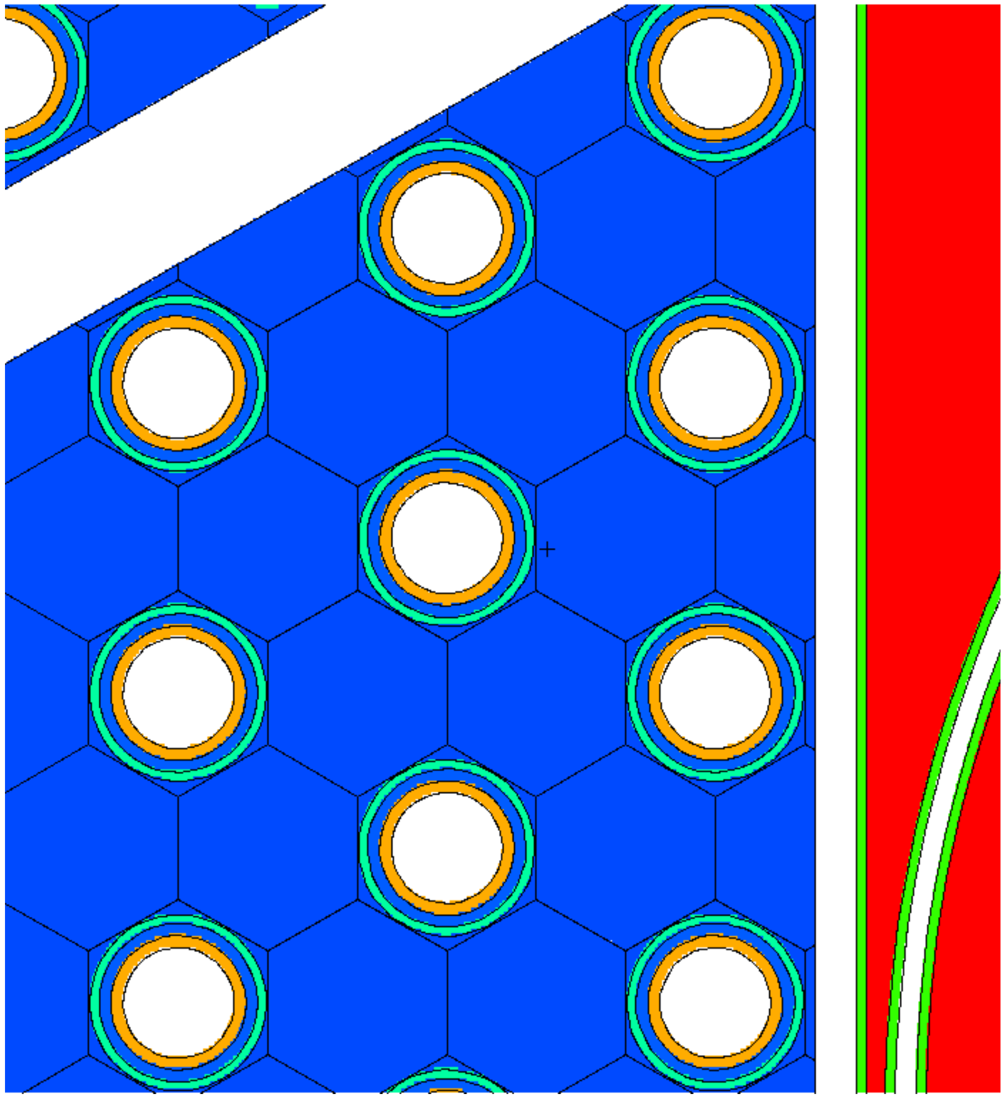

**Figure 22 Heat pipe and composite**

The previous analyzed configurations involve using a metal for the monolithic block with fuel in traditional pin configurations, where the cladding is the monolithic block wall were included in the qualitative alternatives study. One more configuration, stainless steel with UN fuel pellets, was also included as shown below.

- Stainless steel with Uranium oxide fuel pellets (**SS_UO2**)
- Molybdenum alloy (TZM) with Uranium Nitride fuel pellets (**Moly_UN**)
- Stainless steel with Uranium Nitride fuel pellets (**SS_UN**)

The second configuration was the HP and Block design with Silicon Carbide or graphite as the block that also serves as the cladding for the $UO_2$ fuel. The heat pipe wall is metal (stainless steel). This configuration is then:

- Graphite with Uranium oxide fuel pellets (**Graph_UO2_a**)
- Silicon Carbide with Uranium oxide fuel pellets (**SiC_UO2_a**)

The third configuration is using a ceramic, metal or liquid metal for the monolithic block with a cladding for the fuel in a traditional pin configuration, where the cladding is a metal and the heat pipe wall is also a metal. The two configurations were:

- Graphite with Uranium oxide fuel pellets (**Graph_UO2_b**)
- Silicon carbide with Uranium oxide fuel pellets (**SiC_UO2**)

This third variation included replacing the monolithic block by a pool of metal with fuel in a cladded pin configuration. The pool does not circulate and is not intended as a means of heat removal the goal was to use it as a means of conduction between the fuel and the heat pipe, but with the mechanical flexibility of a liquid hence eliminating the stress issue. This variation had three configurations with varying thickness of fuel cladding are shown as:

- LBE metal pool with steel clad (0.040" thick) uranium oxide pellets (**LBE_UO2_a**)
- LBE metal pool with steel clad (0.020" thick) uranium oxide pellets (**LBE_UO2_b**)
- LBE metal pool with steel clad (0.005" thick) uranium oxide pellets (**LBE_UO2_c**)

The final configuration was to use a composite of fuel and monolithic block with the fuel mono-dispersed into the block. The goal was to improve attributes of the heat transfer during accident conditions by eliminating the extreme temperature that occur from the transition from block to fuel-cladding gap to fuel centerline. Two types of fuel in combination with three substrate materials were examined and included:

- Uranium oxide fuel dispersed in stainless steel a small amount of gadolinium as a chemical stabilizer (**Disp_UO2_a**),
- Uranium oxide fuel dispersed in graphite a small amount of gadolinium as a chemical stabilizer (**Disp_UO2_b**),
- Uranium oxide fuel dispersed in molybdenum a small amount of gadolinium as a chemical stabilizer (**Disp_UO2_c**),
- Uranium oxide fuel dispersed in stainless steel no gadolinium (**Disp_UO2_d**),
- Uranium oxide fuel dispersed in graphite no gadolinium (**Disp_UO2_e**),
- Uranium oxide fuel dispersed in molybdenum no gadolinium (**Disp_UO2_f**),
- Uranium nitride in dispersed in stainless steel, (**Disp_UN_a**)
- Uranium nitride in dispersed in graphite, (**Disp_UN_b**)
- Uranium nitride in dispersed in molybdenum, (**Disp_UN_c**)

The analysis was performed in a fashion similar to the methodology presented in Appendix A, except with less detail. As an example, transient thermal response to accident conditions was not done for each proposed alternative reactor design.

Comparing the performance for each design was done in a qualitative fashion. Four categories were defined and each concept was given a qualitative score of 0 to 10 based upon relative performance of each design in that category. The four categories are as follows:

- Neutronic performance – This category encompasses fuel mass needed to go critical, neutron economy and safety.
- Thermal performance – This category examines steady state heat transfer in the core.
- Developmental Surety – This category looks at material issues, material performance, material uncertainties and ease of manufacture.
- Performance Surety – This category looks at evaluates response to accident conditions and the induced stress and corrosion issues.

The results of the qualitative alternatives evaluation is provided in Table 2.

**Table 2 Results of Qualitative Analysis**

| Abbreviation | Core Configuration | Fuel Material | Block Material | Fuel Mass (kg) | Reactor Mass (kg) | Neutronic Performance | Thermal Performance | Development Surety | Performance Surety | Total |
|---|---|---|---|---|---|---|---|---|---|---|
| SS_UO2 | Block | UO2 | SS | 5161 | 22116 | 5 | 5 | 8 | 10 | 28 |
| SS_UN | Block | UN | SS | 6736 | 23849 | 8 | 7 | 5 | 8 | 28 |
| Moly_UN | Block | UN | Mo | 6711 | 25648 | 5 | 12 | 4 | 8 | 29 |
| Graph_UO2_a | HP&Block | UO2 | graphite | 5161 | 18311 | 7 | 8 | 1 | 3 | 19 |
| SiC_UO2_a | HP&Block | UO2 | SiC | 5161 | 22396 | 6 | 6 | 5 | 5 | 22 |
| SS_U02_a | Tube&Block | UO2 | SS | 5161 | 27167 | 2 | 2 | 9 | 9 | 22 |
| Graph_UO2_b | Tube&Block | UO2 | graphite | 5161 | 23379 | 5 | 4 | 9 | 8 | 26 |
| SiC_UO2_b | Tube&Block | UO2 | SiC | 5161 | 23378 | 5 | 4 | 6 | 6 | 21 |
| LBE_UO2_a | Tube&Block | UO2 | LBE-.040" | 5161 | 28751 | 4 | 9 | 7 | 4 | 24 |
| LBE_UO2_b | Tube&Block | UO2 | LBE-.020" | 5161 | 27184 | 5 | 10 | 7 | 4 | 26 |
| LBE_UO2_c | Tube&Block | UO2 | LBE-.005" | 5161 | 25901 | 6 | 10 | 6 | 3 | 25 |
| Disp_UO2_a | Composite | UO2Gd | SS | 9690 | 26163 | 2 | 4 | 8 | 9 | 23 |
| Disp_UO2_b | Composite | UO2Gd | graphite | 7368 | 23851 | 2 | 6 | 7 | 6 | 21 |
| Disp_UO2_c | Composite | UO2Gd | Mo | 10627 | 27110 | 0 | 8 | 6 | 7 | 21 |
| Disp_UO2_d | Composite | UO2 | SS | 9906 | 26389 | 6 | 3 | 7 | 7 | 23 |
| Disp_UO2_e | Composite | UO2 | graphite | 7585 | 24067 | 8 | 6 | 7 | 6 | 27 |
| Disp_UO2_f | Composite | UO2 | Mo | 10843 | 27325 | 2 | 8 | 5 | 6 | 21 |
| Disp_UN_a | Composite | UN | SS | 11935 | 28417 | 8 | 4 | 5 | 8 | 25 |
| Disp_UN_b | Composite | UN | graphite | 9613 | 26095 | 10 | 7 | 4 | 5 | 26 |
| Disp_UN_c | Composite | UN | Mo | 12872 | 29354 | 4 | 9 | 5 | 7 | 25 |

## Results of Alternatives Study

The results of the alternative study are presented by fuel choice, block material choice, geometry choice and finally by overall performance.

### Fuel Material

Uranium Nitride performed better than Uranium Oxide fuel both neutronically and thermally. This was expected given that UN has a higher uranium density and better thermal conductivity than Uranium Oxide. However, the UN fuel does not score as well on development given that its experimental data set is lacking compared to $UO_2$. So a first heat pipe reactor concept would be $UO_2$ but a larger better performing system would be UN or other fuel types.

### Block Material

The choice of block material is a tough choice. Graphite and Stainless Steel perform better neutronically than the other choices. Graphite slows neutrons and make the reactor slightly epithermal in its neutron spectrum. Stainless Steel had the least neutron absorption of the remaining materials. LBE with a thin walled cladding did almost as well.

Thermal performance was best for two materials, the Molybdenum and the liquid LBE core. The Molybdenum has superior thermal performance as was demonstrated in the detailed analysis of the first two concepts. The LBE is unique in that the issue of stress in the core from accident conditions is eliminated. It also has good thermal performance, given that it is assumed that some amount of natural circulation (Rayleigh cell) would occur and improve temperature distributions. Graphite was not far behind either LBE or Moly for thermal performance and ranked well enough to be considered a good choice.

Developmental surety was best for stainless steel and graphite. These materials are well known and have excellent data for resistance to radiation damage and thermal/mechanical performance. LBE is not far behind given its use in Russian reactors. Silicon carbide and Molybdenum would need a greater degree of study.

Finally for performance surety the best scores were for stainless steel and graphite. These two materials have well characterized performance in reactors. Other materials are less well known in terms of their performance for radiation damage. LBE is also well characterized in terms of performance in reactor technologies. However, a liquid pool introduces new issues common to other reactor designs, namely the loss of coolant during an accident. This accident could be made to be low frequency by design, so this may not be a big issue. Finally LBE would have corrosion issues that would have to be designed for.

### Ranking of alternatives

The alternatives were ranked by summing the ranking in each attribute category. The results show that the most promising designs are a combination of proven and unproven technology.

Proven technology would the use of known materials such as stainless steel as the block material and UO2 as the fuel.

Unproven technology would advocate looking to alternate materials for the block such as moly, graphite or a liquid metal. The recommended second alternative choice for unproven technology would be the Moly block with UN fuel. For more detailed analysis, (since the Moly block and UN fuel combination have been examined in detail), the two most intriguing alternatives would be 1) a composite graphite or moly block with either UO2 or UN imbedded in the matrix, with a lean toward UN in graphite and 2) a UN or UO2 fueled system in LBE. These analyses are recommended for future studies.

## Overall lessons learned and recommendation going forward

The lesson learned from this study was that heat pipe reactors could be scaled into the low to mid 10's of MW electric with the right choice of fuel and reactor materials. A recommended power level of about 30 MWe was seen as the upper end of what could be accomplished with current technology. Other reactor technologies would make more sense above power levels approaching 100 MWe or higher.

Going forward it makes the most sense to build the first heat pipe reactors with conventional fuel and materials. The easiest path forward would be to stay with a uranium oxide fuel and stainless steel.

### Possible Reactor Concept for Remote Locations

A possible reactor concept for remote locations is shown in Figure 23. The reactor concept is a heat pipe reactor that would be coupled to an open air Brayton power conversion system. The goal of the concept would be to have a reactor that could be easily deployed by truck or air transport. A concept for transport is shown in Figure 24. In this concept the reactor is flown to a military base and then transported to site via a semi-truck. Dirt berms are used as partial shielding and the entire truck is moved into location. The reactor is integrated into the existing power structure by replacing natural gas driven Brayton power conversion with reactor driven power conversion.

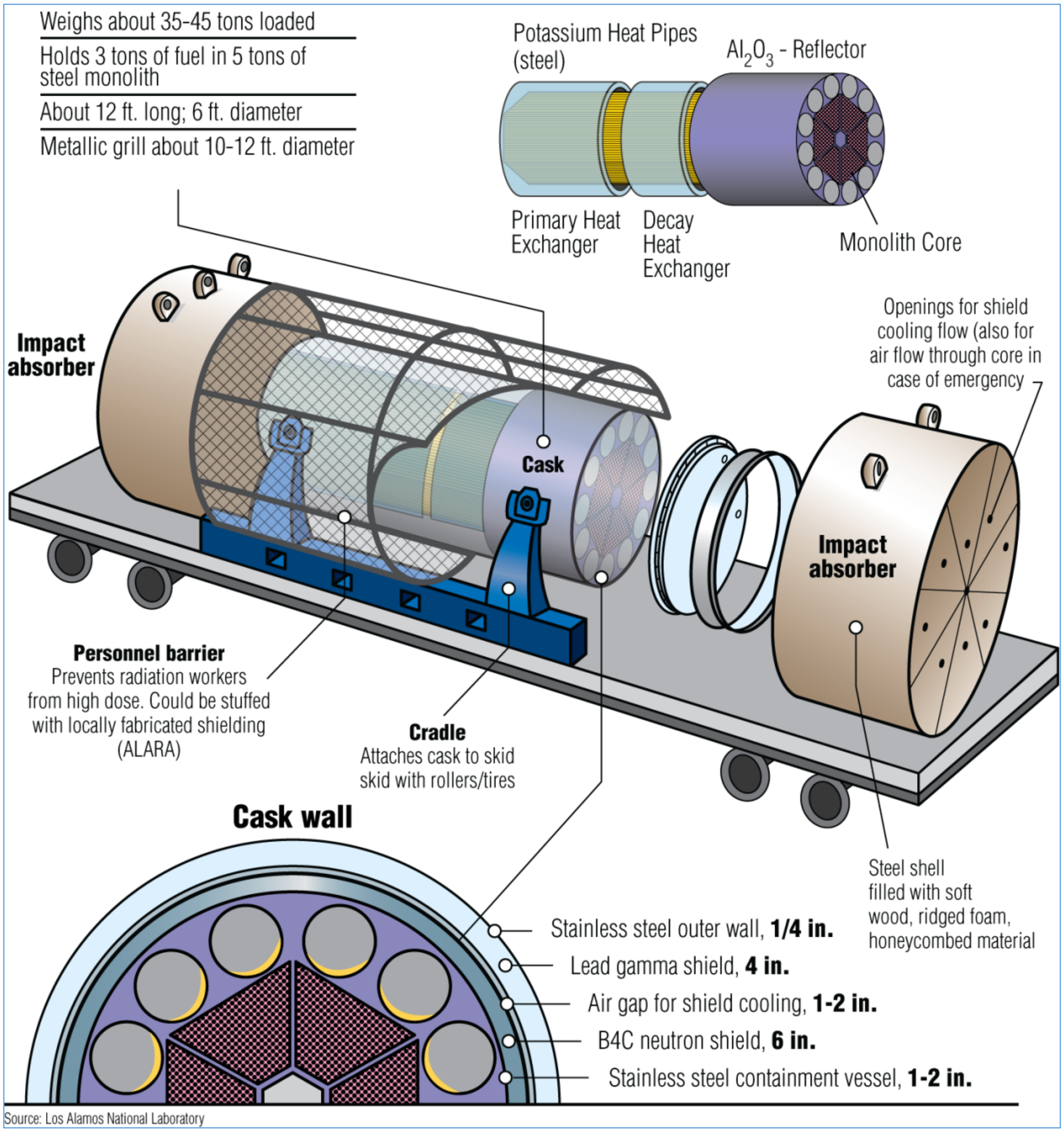


**Figure 23 Mobile heat pipe reactor in shipping cask**

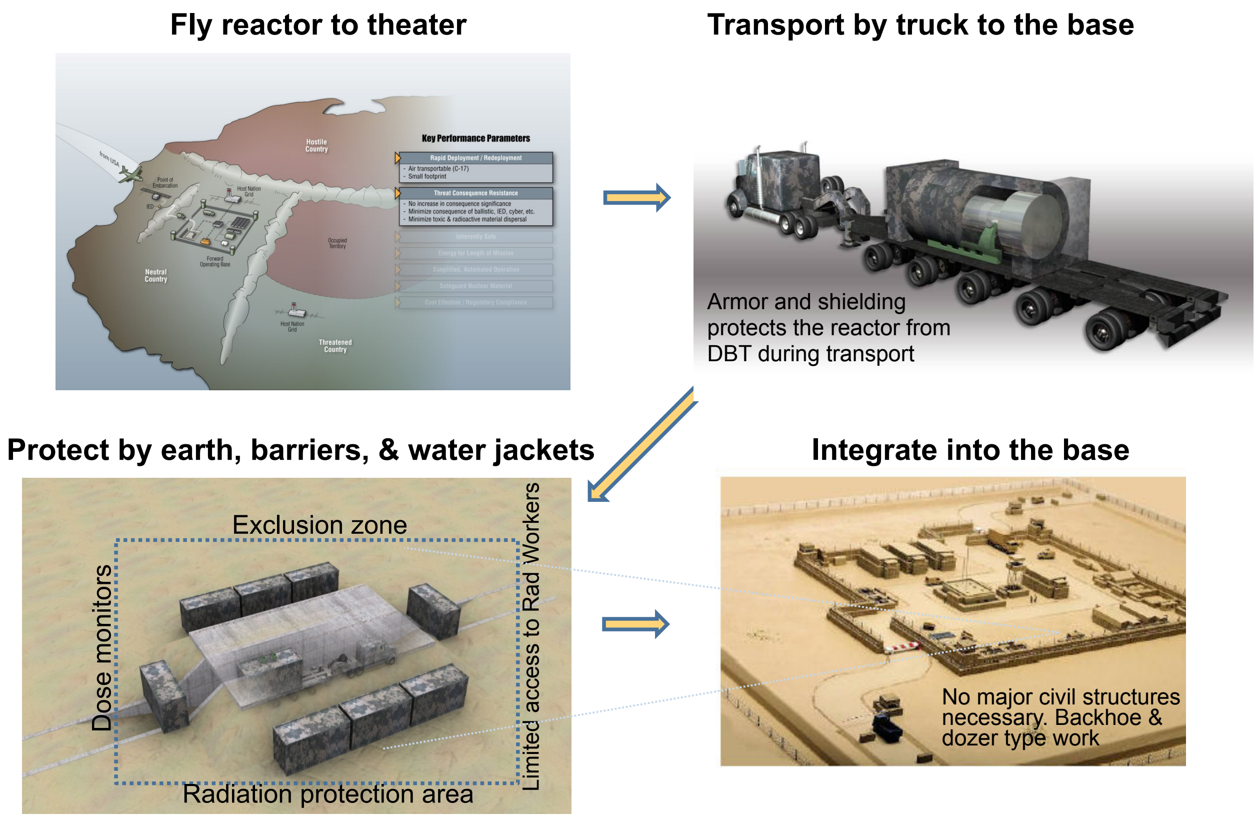


**Figure 24 Concept for mobile reactor deployment**

# Appendix A

## Methodology for Analysis

Los Alamos has a methodology that involves a preprocessor know as ALLGEN. ALLGEN has been developed so that it can be used for almost any reactor technology, and its reactor system model generation capability is automated, so it can perform rapid design or trade study calculations. The versatility is in part provided by MCNP, a Monte Carlo neutronics tool. MCNP utilizes continuous energy nuclear cross sections, which can provide accurate calculations in any neutron spectrum, and a geometry package that can model almost any conceivable reactor in detail.

Initially, calculations are performed to identify a design or designs that meet reactivity design requirements (i.e. maintaining sufficient criticality throughout life as well as having sufficient shutdown margin to ensure subcriticality prior to deployment). For these calculations, a large number of criticality cases are set up in ALLGEN and run by MCNP. For each case, ALLGEN estimates the temperatures of the fuel pins, structure, core barrel and fuel and control components are thermally expanded in three dimensions, with corresponding material density reductions. The reactor coolant inventory is reduced according to the expansion-reduced free coolant volume space, and its material density reduced. Irradiation-induced fuel swelling is also estimated as a function of fuel temperature and burnup.

These calculations are performed in a simplified manner, but are accurate enough to produce designs that will generally require only minor design adjustments based on more detailed, time-history thermal and structural analyses.

Once a design is identified, a time-history nuclear analysis is performed. For these calculations, the MCNP core model and the MONTEBURNS depletion calculations are performed interactively. MONTEBURNS simulates the movement of control rods during the burnup calculation to maintain criticality during the calculation, and also to ensure that the rods have enough worth and margin at end-of-life. These calculations confirm the as-designed end-of-life reactivity margin calculated with ALLGEN. Also, temperature defect and reactivity coefficients are calculated, as are the time-history power and neutron flux core distributions.

In addition to the nuclear design calculations discussed above, simplified calculations can be performed by the FRINK code. Two analyses are performed: one that assesses overall system performance, and one that calculates core response to bounding transient events. MCNP provides heating rates and reactivity coefficients that are used to evaluate system temperatures during steady-state and the transients. In steady-state, the key parameters generated by FRINK represent the overall power balance of the system. The transient analyses investigate how the system responds to reactivity insertion, loss of heat sink, and loss of flow.

The FRINK system model is not intended for use in detailed plant analyses or licensing calculations. Its primary purpose is to assess the design early on for any potentially adverse transient responses, so that the design can be modified if needed in the early stages.

The fuel pin assembly thermal-hydraulic design analysis is performed using a Visual Basic-based computer code called TMSS, which is executed in Excel. The code is a general reactor design tool, which for any given reactor type, is adapted to analyze that system. It contains material properties libraries for coolants (gas and liquid), fuels and structural materials. For each fuel type there are separate routines to calculate fuel swelling and fission gas release. There are also routines for each structural material type to calculate clad stress, thermal and irradiation-induced creep. If there is gas in the fuel pin gap, the mixed gas concentrations and thermal properties are calculated for each time point.

The analysis performed uses as-calculated temperatures to determine thermal expansions, material properties, fuel performance, clad stress and stress limit, clad creep strain. Initially temperatures are guessed and used in these calculations. The final calculated temperatures are compared to the initial temperatures for each time point in the time history. If the maximum difference is greater than a specified input limit (typically set at 1°C), then the calculation is repeated with new input values until convergence is achieved.

Some of the features and options in the code are:

1. Process the MCNP cell powers and fluxes, identifying high power/burnup fuel pins, used to size the fuel pin dimensions. Option to average powers and fluxes at symmetric locations to improve statistics.
2. Size the fuel pin dimensions to meet design requirements such as minimum fuel pellet-to-clad gap throughout life (most restrictive of the assembly gap and the hot and cold operating gaps).
3. Calculate clad stresses and creep (thermal and/or irradiation-induced) as a function of temperatures, burnup (gas pressure) and fast fluences. For a selected time point(s), calculate clad stress and creep for a specified loss-of-flow event (simplified calculation).
4. Option to orifice the core (specified number of orifices, produces same maximum coolant temperature for each orifice type, but not necessarily at the same time point).
5. Options to account for manufacturing tolerances, corrosion allowance, uncertainty in fuel swelling.
6. Option to adjust the coolant temperatures to meet a maximum specified clad and/or fuel temperature.
7. Option to size the fission gas plenum height or initial gas pressure or the clad thickness to meet specified clad stress and creep strain limits.
8. Option to adjust the coolant geometry (fuel pin pitch-to-diameter ratio) to meet a specified core pressure drop, coolant velocity or pumping power limit.

The core geometry and cell power and flux data from the nuclear design are read in to the TMSS code from files generated in the nuclear design. These are processed in TMSS to produce fuel pin time-histories, which are used in the fuel pin performance calculations.

A time-history thermal analysis for a fuel pin is performed with a 1½ D model.  Any or all of the fuel pins may be analyzed, but for initial assessments typically a limiting fuel pin (either max power or max burnup) pin and a core average pin are analyzed. The limiting fuel pin is used to size the pin dimensions to meet design requirements (temperatures, stresses/strains, fuel clad gap, etc.).  The core average pin results are representative of core average in the fuel performance calculations, fuel swelling and fission gas release are calculated at several fuel pin elevations, for each time point in the time history.  The incremental fission gas releases are summed over the fuel pin height to give the total fission gas release in the pin.  The fission gas release accounts for, in large part, the pressure inside the fuel pin.