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Idaho National Laboratory in collaboration with Argonne National Laboratory has evaluated technology options for a new fast spectrum reactor to meet the fast-spectrum irradiation requirements for the USDOE Generation IV (Gen IV) and Advanced Fuel Cycle Initiative (AFCI) programs. The US currently has no capability for irradiation testing of large volumes of fuels or materials in a fast-spectrum reactor required to support the development of Gen IV fast reactor systems or to demonstrate actinide burning, a key element of the AFCI program. The technologies evaluated and the process used to select options for a fast irradiation test reactor (FITR) for further evaluation to support these programmatic objectives are outlined in this paper.

I. INTRODUCTION

To meet future energy needs, ten nations have agreed on a framework for international cooperation in research for an advanced generation of nuclear energy systems, known as Generation IV. These ten nations have joined together to form the Generation IV International Forum (GIF) to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, nuclear waste, proliferation of nuclear materials, and public concerns and perceptions on nuclear power. The U.S. Department of Energy (DOE) has recently reversed the earlier multi-decade trend of decreasing funding for advanced nuclear energy research and development and has initiated a program for development of Generation IV advanced nuclear energy systems to be deployable by ~2030. After developing a Technology Roadmap, the U.S. DOE and the GIF selected six technologies for further development. Nuclear energy systems based on fast-spectrum reactors are among the selected systems. This project represents a collaboration between Argonne National Laboratory (ANL) and the Idaho National Laboratory (INL) to develop concepts for a Fast Irradiation Test Reactor (FITR) that will be a key element in the research and development programs leading to Generation IV reactor systems development.

Fast-spectrum reactor concepts are proposed for sustainability (improved uranium utilization) and the mission of transmutation of long-lived nuclides present in spent nuclear fuel. Transmutation processes would significantly reduce the thermal burden on a geologic repository and the duration of radiotoxicity of disposed nuclear waste from tens of thousands of years to hundreds of years. Additionally, planners for future space missions are looking toward the use of space-based fission power systems to extend mankind's reach into our solar system and the universe. Fast-neutron-spectrum reactors feature prominently in all these applications. The proposed FITR can play a critical role in advancing these technologies and in resolving other important issues.

For over thirty years, DOE, universities, and research institutions have relied on infrastructure built in the 1950s and 1960s to develop new materials, create important medical isotopes, train scientists and engineers, and perform a wide array of basic research activities. This technology, based largely upon the use of research reactors fueled with enriched uranium, has proved invaluable and safe over the decades. However, these reactors are aging and their owners are facing the need to upgrade or give up the important, often one-of-a-kind research capabilities these facilities offer students, researchers, and physicians.

The United States currently has no capability for irradiation testing large volumes of fuels or materials in a fast-spectrum environment. In the U.S., the Advanced Test Reactor (ATR) at the INL provides the primary material irradiation capability in a thermal neutron spectrum (although several smaller research reactors provide additional small-scale irradiation capacity). However, the premature shutdown of the Fast Flux Test Facility (FFTF) and the Experimental Breeder Reactor-II (EBR-II) has left the U.S. dependent upon a small number of foreign sources with limited access for any testing requiring a fast neutron spectrum. Worldwide, only a few fast reactors are currently available for materials testing. These include the PHENIX reactor in France, the JOYO and MONJU reactors in Japan, and the BOR60 reactor in Russia. The PHENIX reactor is scheduled for final shutdown in 2008 and thus will not be available for further irradiations beyond that

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time. The U.S. Advanced Fuel Cycle Initiative program is involved in the last irradiation campaign planned for the reactor prior to shutdown. The JOYO reactor has recently completed refurbishment and will begin to allow access by foreign customers for their irradiation testing needs. The JOYO planned operations includes extensive maintenance between each run cycle, making the reactor's overall availability less than 50%. The MONJU reactor in Japan is a prototype sodium-cooled fast reactor built in the mid-1980's. Although it is currently in standby awaiting safety-related modifications, when restarted, it may be available for fuel demonstration tests. The BOR60 fast reactor in Russia offers irradiation services. However, because of ongoing policy issues, the Russian facilities are not currently available to U.S. experimenters. Furthermore, the ability to control and monitor the test conditions (controlled temperature, gas purge, continuous monitoring) is limited compared to what is needed for the U.S. programs.

Testing in foreign reactors also comes at a significant cost and schedule penalty. Planning and negotiations for an irradiation campaign must begin many (3-5) years prior to test insertion. Also, the legal restrictions on the transport of irradiated materials can be an issue. For example, in Japan, samples extracted from a JOYO irradiation would not be allowed to leave the country. Therefore, post-irradiation examination of samples must then take place locally at an unspecified cost. To perform an irradiation of just a handful of experimental test pins in PHENIX, the cost premium for the irradiation services is ~\$5M, and the added time necessary to retrieve data ~ 2 years. A U.S. based irradiation capability would alleviate all of the issues associated with foreign reactor irradiations and therefore presents a compelling argument for developing and building a Fast Irradiation Test Reactor in the U.S.

The FITR concepts investigated under this project have been developed and evaluated through a coordinated effort between the INL and ANL. The work conducted encompasses both conception and feasibility studies of potential FITR systems. The integrated team of experts from the two Laboratories has explored novel as well as somewhat more conventional concept options. The team has developed a set of functional requirements as well as specific criteria, metrics, and importance factors related to achieving the mission objectives of the FITR consistent with the needs of existing and Generation IV reactor systems and future space reactor systems.

II. MISSIONS

Research and development of advanced nuclear fuels and materials for the Advanced Fuel Cycle Initiative and Generation-IV programs, requires an adequate fast-neutron irradiation capability to test candidate samples in a prototypic environment. Testing is necessary to prove the adequate performance of the fuels and materials prior to implementation of a demonstration reactor. The system must meet certain requirements for test volume, instrumentation, neutron spectrum, and physical conditions characteristic of the anticipated operating environment. For the fuels and materials being explored as part of the AFCI and GEN-IV initiatives, specific requirements include the ability to achieve a wide range of temperatures under tightly controlled conditions. It is recognized that a user facility of this type will also benefit other DOE programs such as the space reactor and fusion energy programs, therefore, to address the performance gap, the FITR will meet the following mission needs:

- Provide the capability to safely irradiate materials and fuels in a fast-neutron spectrum and a prototypic environment to meet the technology needs of the AFCI and GEN-IV programs and thereby allow them to perform the necessary proof-of-performance demonstrations.
- Provide a user facility for the DOE to perform fast-neutron spectrum irradiations for university and international collaborations, and other DOE programs.

The FITR will be a multi-mission facility that will support missions not currently possible in the U.S., and for which limited or no capability is available elsewhere. It is expected that the FITR will be built and operated at the Idaho National Laboratory (INL).

III. REQUIREMENTS

The FITR would be capable of supporting a variety of missions. However, a requirements list was developed to focus the conceptual design on the critical attributes. Table 1 summarizes which features were considered essential, desirable, and beneficial.

IV. REACTOR CONCEPTS

Four concepts for the FITR have been developed that utilize different reactor coolants, including sodium, lead-bismuth eutectic (LBE), and helium. The four concepts have include conventional sodium-, LBE-, and helium-cooled reactors as well as a new pressure tube reactor (PTR) concept.

Both the sodium- and LBE-cooled options utilize a traditional, vertical core with hexagonal fuel assemblies, while the PTR option utilizes a horizontal core based on a CANDU pressure tube/calandria tube configuration. Like the sodium- and LBE-cooled options, the gas-cooled option is based on a vertical core with hexagonal fuel assemblies.

IV.A. Sodium-Cooled Fast Reactor

Sodium as a reactor coolant has been the choice for nearly all fast reactors proposed and/or constructed. A sodium-cooled reactor core concept that meets the facility requirements has been developed. When compared in a consistent manner to other coolants considered in this report, the use of sodium generally results in a more compact, lower power, lower enrichment core while achieving the desired flux levels within the irradiation volume.

TABLE I

Facility Requirements

Priority	User Needs/Design Drivers
Essential	Must meet all applicable safety requirements
	High fast neutron flux, large sample volume, steady-state reactor: neutron flux (>1x10^{15} n/(cm^2s); E> 0.1 MeV) averaged over volume inside 2 helium-filled loops (1-m high, 8-cm diameter)
	Use of a standard fuel type to minimize research and development operations costs (i.e. oxide fuel or metal fuel are already qualified).
	Multiple (at least 2) in-core loops required (supporting variable dimensions, temperature, and coolants, i.e., Na, Pb, LBE, CO_2, and He)
	Easy access to loops and irradiation positions
	High availability and capacity factor
	Cost effective (<$1 billion) user facility for universities and others
	Low-enrichment fuel (<20%) to limit security issues
	The first fresh core will not contain any plutonium, but subsequent core reloads may contain plutonium as lead test assemblies.
Desirable	Optimized reactor control (number of control rods, etc.)
	On-line and/or easy maintenance/inspectability
	Long operational lifetime (40Š60 years for irreplaceable components)
	Optimal use of passive safety characteristics
Beneficial	In-core rabbit loops
	On-line experiment changes
	Participation from several outside users (DOE, DOD, NASA, universities, and industry)
	Dry heat exchange (if required by environment and cost)
	Use of system heat to offset operations costs (if cost effective overall)
	Core components and support structures that are easily replaced to respond to radiation embrittlement concerns

The nominal requirements for the FITR limit enrichment to 20% U-235. To help meet this requirement, the SFR uses metallic U-10Zr as the primary fuel choice to maximize heavy metal density within the core. An oxide (UO_2) core has also been evaluated; however parametric studies were confined to the metal core because it produces a higher fast flux at a given power level. Cladding is the ferritic steel HT-9, which shows virtually no swelling beyond a fast fluence of 3x10^{23} n/cm^2. Fuel pins are arranged along a tight triangular pitch within hexagonal assemblies so that a high fuel volume fraction can be achieved. Core inlet and outlet temperatures are assumed to be 350 and 500°C, respectively. Operating at temperatures typical of a power reactor helps minimize intermediate and dump heat exchanger sizes and will also provide a more prototypic environment under which irradiation testing can be conducted.

Core loading options were evaluated using the REBUS-3 fuel cycle analysis code in conjunction with the DIF3D nodal diffusion option (Toppel 1983, Derstine 1982). Equilibrium operating conditions were assumed based on a fixed cycle length of 180 days and a three-batch core. Fuel residence time was therefore 540 full-power days. To find the minimum core size for the SFR, the number of driver assemblies was increased until the charge enrichment for U-10Zr fuel dropped below 20%. At that point, the core power level was increased until experimental flux requirements were met. Using this approach with an assembly design that has a 100 cm active height, the equivalent core diameter is 119 cm and the charge enrichment for fresh, metal fuel was found to be 17.9%, which is well below the 20% enrichment limit. However, because of the relatively small core size, removing a row of driver assemblies causes the enrichment to increase to approximately 21%. Using the same core diameter, the charge enrichment for oxide fuel was found to be 19.9%. Therefore, the 119 cm core diameter represents the most compact core possible when using the assembly design. In the current design, longer fuel residence times could be considered for the metal core by either increasing the number of batches in the core or extending the cycle length. Both options would result in higher discharge burnup while reducing fuel handling requirements.

Core thermal power was adjusted to 135 MWth in order to meet the required average experimental fast flux level for the metal core. An experimental volume of 73 liters is provided by 10 assembly positions within the core. Seven of the positions are grouped at the center of the core, providing for the possibility of a single, large experimental region. Average fast neutron flux over the volume of the central experimental positions is 1.00x10^{15} n/cm^2/s. When all ten experimental positions are considered, the average is 0.99x10^{15} n/cm^2/s.

For the oxide core, thermal power was kept at 135 MWth for comparison with the metal core. Under these conditions the average fast flux over the central experimental positions is 0.84x10^{15} n/cm^2/s. Averaged over all ten experimental positions the average is 0.83x10^{15} n/cm^2/s.

IV.B. Lead-Bismuth-Cooled Fast Reactor

The feasibility of using lead-bismuth eutectic (LBE) as a reactor coolant was demonstrated in Russian Alfa-class submarines, the last of which was decommissioned in 1995 (Rawool-Sullivan et al 2002). An LBE-cooled reactor core concept that meets the facility requirements has been developed.

A constraint unique to the LFR is a limit on peak coolant velocity. Coolant velocity is kept below 2 m/s to avoid excessive erosion of core structural materials (Fomitchenko 1998). This leads to a more open pin lattice in the driver assemblies, which results in a larger core size than might otherwise be possible.

Because the requirements for the FITR limit enrichment to 20% U-235, the LFR assumes metallic U-10Zr fuel to maximize heavy metal density within the core, although compatibility with LBE is recognized as a potential problem. Cladding is the ferritic steel HT-9. Compatibility of both metallic fuel and HT-9 with LBE is not addressed in this study. These choices for fuel and cladding are used to provide consistency in comparing the impact of different coolant choices. In addition, core inlet and outlet temperatures are assumed to be 350 and 500°C, respectively, as with the SFR option.

LFR core loading options were evaluated using the REBUS-3 fuel cycle analysis code in conjunction with the DIF3D nodal diffusion option (Toppel 1983, Derstine 1982). Equilibrium operating conditions were assumed based on a fixed cycle length of 180 days and a three-batch core. To find the minimum core size for the LFR, the number of driver assemblies was increased until the charge enrichment for U-10Zr fuel dropped below 20%. At that point, the core power level was increased until experimental flux requirements were met. Using this approach, the final core layout had an equivalent core diameter of 154 cm, which represents an increase in core volume of 67% over the SFR option. (The active height is 100 cm in both cases.) Despite this, the heavy metal inventory for the LFR option is only 11% higher due to the more open pin lattice in the LFR option. The charge enrichment for fresh fuel was found to be 19.7%. Because there is little margin to the 20% enrichment limit, modifications to the assembly design (to be closer to the velocity limit) could be used to develop an improved overall core design.

Core thermal power is 140 MWth in order to achieve an average fast flux of 1.00×10^{15} n/cm^2/s over the central experimental positions. An experimental volume of 73 liters is provided by 10 assembly positions within the core. Seven of the positions are grouped at the center of the core, providing for the possibility of a single, large experimental region. The average fast neutron flux over the remaining three positions is 0.99×10^{15} n/cm^2/s, but when combined with the central positions (which dominate the average) the average over all ten experimental positions remains at 1.00×10^{15} n/cm^2/s.

IV.C. Pressure Tube Reactor

The pressure tube reactor (PTR) is horizontal in orientation with many pressure tubes running the entire length of a scattering-medium tank. (Not all pressure tubes will be fueled.) Fuel bundles are positioned inside the fueled pressure tubes. A shroud tube surrounds each pressure tube to allow for an annular gas gap. This gap configuration allows change-out of pressure tubes without draining the scattering-medium tank. The entire scattering-medium tank is filled with molten lead-bismuth eutectic (LBE) surrounding all of the shroud tubes. Coolant is pumped through the fuel bundles inside the pressure tubes to remove fission heat.

Two in-pile test loops exist inside the reactor that allow experimenters to perform integrated concept testing with prototypic fuel and coolant conditions. The loops are separated from each other so that simultaneous tests can be run using, for example, sodium and supercritical water in the different loops. The presence of the sample handling machine allows removal of samples (in the pressure tubes) while the reactor is at power. The coolants explored were helium, light and heavy water, steam, and LBE.

The PTR models used a standard CANDU pressure tube (PT)/calandria tube (CT) configuration with CO_2 in the insulation gap. The MCNP4B code was used for beginning-of-life (BOL) reactivity and fast flux calculations. Only 1/8 and 1/12 sections of the core are modeled for the square-lattice and hexagonal-lattice configurations, respectively. The total number of neutron histories followed for each run is 3×10^5. Thirteen different core, coolant, and fuel configurations were explored.

The results indicate that a pressure-tube fast-test reactor can achieve greater than 1×10^{15} n/cm^2/s fast flux, with acceptable reactivity and reasonably low thermal power. However, the design is a significant departure from a traditional CANDU reactor.

IV.D. Gas-Cooled Fast Reactor

Two advanced helium-cooled reactor systems, one thermal and one fast, are being considered as part of the Generation IV technology roadmap. Although helium has been used in a small number of thermal reactors, no gas-cooled fast reactor (GFR) has been constructed. A preliminary helium-cooled concept that meets the facility requirements is summarized in this section.

The initial assembly design used for the GFR option is based on the assembly design developed by Gulf General Atomics (GA) for the gas-cooled fast breeder reactor demonstration plant in the early 1970s (Pellaud 1971). This design utilized mixed oxide fuel and achieved an average fast flux of over 2.0×10^{15} n/cm^2/s; however it was also a relatively high-power core (826 MWth) compared to what one might expect for a test reactor. For the analyses presented here, UO_2 fuel is assumed.

Core loading options for the GFR were evaluated using the REBUS-3 fuel cycle analysis code in conjunction with the DIF3D nodal diffusion option (Toppel 1983, Derstine 1982). Equilibrium operating conditions were assumed to be identical to the SFR and LFR options. The assembly design from the demonstration plant combined with the requirement to use low-enriched uranium-based fuel resulted in a large core size. To assess the impact of a more compact assembly design, the pin pitch from the original assembly was arbitrarily reduced by 10% and the inter-assembly gap was cut in half. Once a core layout was developed to meet the enrichment requirement, core thermal power was adjusted until the flux requirement is met within the irradiation positions. The resulting power levels for the two designs are significantly higher than for the liquid-metal-cooled options.

Core thermal power for the design was 295 MWth. Under these conditions, the average fast flux within a single central experimental assembly meets the fast flux requirement of 1.00×10^{15} n/cm^2/s. Because of the larger assembly sizes, only a single experimental position is placed at the center of the core, but it provides 31 liters of experimental volume. Like the SFR and LFR designs, three additional experimental assemblies are placed at mid-core positions. However, due to higher leakage in the GFR, those positions produce an average fast flux of only 0.83×10^{15} n/cm^2/s, respectively. Careful evaluation of thermal-hydraulic performance is needed so that a more optimized assembly design can be developed in the future to reduce core size and power level while improving fast neutron flux in the experimental positions.

IV.E. Comparison of Reactor Concepts

The core loadings and performance of the four reactor concepts are compared in Tables 2 and 3. A comparison of the core loading for each concept in Table 2 shows that the liquid-metal cooled reactors achieve the desired fast flux production in a more compact, lower power core. Of the two liquid-metal cooled designs, the LFR core is larger than the SFR core because a more open pin lattice is used to reduce coolant velocity.

TABLE 2

Core Loading Description for the SFR, LFR, PTR, and GFR Concepts.

Thermal Power, MW	**135**	**140**	**267** [a]	**295**
Coolant	**Sodium**	**LBE**	**Helium**	**Helium**
Experimental Positions	10	10	2	4
Total Active Volume, liters	73	73	21	84
Average Fast Flux, 10^{15} n/cm^2/s	**0.99** [b]	**1.00** [b]	**1.3**	**0.83** [b]
Core Loading				
Fuel Composition	U-10Zr	U-10Zr	U-10Zr	UO_2
Number of Driver Assemblies	135	237	126 [c]	141
Lattice Pitch, cm	9.2	9.2	15.0	15.5
Equivalent Core Diameter, cm	119	154	200 [c]	200
Active Height, cm	100 [d]	100 [d]	135	100
Charge Enrichment, % U-235/HM	17.9	19.7	20	20.3
Heavy Metal Inventory, kgHM	5138	5716	6670	9791
Heavy Metal Loading, kgHM/yr	3126	3477	n/a	5956
Volume Fractions				
Fuel (smeared)	0.5055	0.3203	0.1454 [e]	0.3938
Fuel (as fabricated)	0.3791 [f]	0.2402 [f]	0.10905	0.3643 [g]
Bond/Gap (as fabricated)	0.1264	0.0801	0.03635	0.0295
Structure	0.2080	0.1812	0.0944 [e]	0.1992
Coolant	0.2865	0.4985	0.1447 [e]	0.4070
Other	n/a	n/a	0.6154 [e]	n/a

a. Thermal power for the PTR is estimated to be 205 MW for a fast flux of 1.0×10^{15} n/cm^2/s.
b. Average fast flux in the central position(s) is 1.00×10^{15} n/cm^2/s in the SFR, LFR, and GFR cases.
c. 126 pressure tubes, each containing three fuel bundles. Diameter is based on core tank diameter.
d. Active core height includes the length due to 5% axial swelling of the fuel.
e. PTR volume fractions are with respect to core tank. ÒOtherÓ represents internal LBE reflector
f. Fuel is assumed to be 100% theoretical density at fabrication.
g. Fuel is assumed to be 95% theoretical density at fabrication.

Nuclear, thermal, and hydraulic parameters for the four reactor concepts are compared in Table 3. An equilibrium fuel cycle was not evaluated for the PTR case, as shown by the BOEC and EOEC eigenvalues. Instead, depletion calculations of a startup core with 20% enriched fuel are reported for a 180 day burn time. To properly determine core geometry, fuel enrichment, cycle length, and other parameters, along with the power level needed to produce the required fast flux, an equilibrium cycle, including the effects of fission product buildup and the presence of fuel at different stages of depletion, will need to be evaluated for the PTR.

Because of the relatively short cycle length assumed for all concepts (180 days) burnup reactivity swing, discharge burnup, and peak fast fluence values are all very low. Some concepts are more amenable to longer fuel cycles and higher discharge burnup than others. For example, the relatively low enrichment for the SFR option suggests slightly higher enriched fuel could be used to extend the fuel cycle length, increase burnup, and reduce heavy metal loading requirements. The remaining concepts, however, are all very close to the 20% enrichment limit, and other improvements in the designs would be required before longer fuel cycles could be considered.

TABLE 3

Core Nuclear and Thermal Performance for the SFR, LFR, PTR, and GFR Concepts.

Thermal Power, MW	135	140	267 [a]	295
Coolant	**Sodium**	**LBE**	**Helium**	**Helium**
Nuclear Performance				
Core Residence Time, FPD	540	540	180 [b]	540
Eigenvalue, k_{eff}				
BOEC [c]	1.0056	1.007	1.054 [d]	1.007
EOEC [c]	1.000	1.001	1.044 [d]	0.999
Burnup Reactivity Swing, %Δk	0.57	0.62	0.95	0.75
Average Discharge Burnup, GWd/MT	14	13	n/a	16
Peak Fast Fluence, 10^{23} n/cm^2	0.6	0.6	n/a	0.6
Thermal Performance				
Average Specific Power, kW/kgHM	26	24	40	30
Average Power Density, kW/liter	132	78	198	98
Peak Power Density, kW/liter	225	141	302	182
Peak Linear Power, W/cm	184	171	392	141
Thermal Hydraulic Performance				
Reactor ΔT, C	150	150	150	200
Coolant Flow Rate, kg/s	705	6350	n/a	284
Peak Driver				
Thermal Power, MW	1.32	0.85	n/a	2.99
Coolant Flow Rate, kg/s	6.91	38.7	n/a	2.88
Coolant Flow Area (hot), cm^2	18.02	33.71	n/a	77.28
Coolant Velocity, m/s	4.5	1.13	n/a	62.5 [e]
Pressure Drop, MPa	0.22	0.042	n/a	0.42
Pumping Power, kW	1.76	0.159	n/a	168

a. Thermal power for the PTR is estimated to be 205 MW for a fast flux of 1.0×10^{15} n/cm^2/s.
b. Length of one cycle in PTR.
c. BOEC: Beginning of Equilibrium Cycle; EOEC: End of Equilibrium Cycle.
d. Based on startup core with 20% enriched fuel. Equilibrium fuel cycle has not been determined.
e. Coolant velocity at peak driver outlet.

The minimum pumping power reported in Table 5-2 was obtained in the LFR because of the relatively open pin lattice used to reduce coolant velocity. The highest pumping power is for the GFR case. This is the result of the low heat capacity and resulting high coolant velocity needed to remove heat.

IV.F. Impact of Lower Power and Plutonium Fuel Option

As described in previous sections, various core configuration options are being investigated to identify the most promising coolant choices for a Fast Irradiation Test Reactor. In order to limit the scope of the studies, several criteria were imposed on the preceding concepts. The two most important in terms of core design include the use of uranium-based fuel limited to 20% enrichment and a target fast flux of 1×10^{15} n/cm^2/s averaged over a substantial experimental volume.

Using these criteria, four primary core design options have been developed. The thermal power required to achieve the desired flux levels varies from around 140 MWth for the liquid-metal-cooled options to nearly 300 MWth for the helium-cooled options. Because of the concern over the high power levels needed to achieve the required fast flux levels, especially for the helium-cooled options, a sensitivity study was conducted where a core power of 100 MWth was established and the fuel was allowed to include plutonium to meet the minimum flux requirement. This sub-section presents the results of applying this new criterion to a series of parametric core designs, which have been evaluated to determine the impact on fast flux and fuel enrichment resulting from varying core diameter, fuel volume fraction, fuel type, and coolant. Because a REBUS model has not been developed for the PTR concept, it was not included in the comparisons presented here. Nevertheless, the comparisons presented here will be applicable to the PTR concept.

In summary, the average fast flux in a central experimental assembly position is most strongly influenced by core diameter (i.e. power density). With the exception of the LBE cases, the dependence of the fast neutron flux on other factors is relatively insignificant. As a result, the highest experimental fast flux will be obtained with the smallest, most compact, highest power density core, irregardless of other factors. However, to accomplish this with a reasonable fissile enrichment, a low leakage core with a high fuel volume fraction is needed.

IV.F.1. Methodology

Neutronics and fuel cycle analyses were carried out by ANL (King and Grandy, 2005) using the DIF3D/REBUS3 fuel cycle analysis code (Toppel 1983, Derstine 1982). A fixed, equilibrium fuel cycle based on a three-batch core with six month refueling intervals was imposed on all core designs considered. All active-core material compositions and dimensions were evaluated at a core-average temperature of 425°C. Flux calculations were carried out using the hexagonal-z nodal diffusion theory option of DIF3D. The fresh fuel charge enrichment was determined so that the reactor would have an end-of-equilibrium-cycle (EOEC) eigenvalue of 1.0. The axial height of each active core was fixed at 100 cm. Experimental fast flux was averaged over the active height of one or more empty, central assemblies. The composition of the experimental assemblies consisted of the assembly duct and coolant only.

For the evaluations presented here, the SFR, LFR, and GFR options were reevaluated at 100 MWth. Core diameter was then increased or decreased by the addition or removal of whole rings of driver assemblies from the core layout. Since the goal of this study is to identify the sensitivity of flux and enrichment to various design parameters, no assessment has been made as to the viability of the different cases evaluated. For example, in the 91-pin LBE case the peak coolant velocity is expected to exceed a 2 m/s limit. Overall, however, the cases presented here have relatively lower power densities, and thermal limits should not be a concern.

A total of 28 configurations were evaluated and are summarized in Table 4. For the sodium-cooled option, both U-10Zr and U-Pu-10Zr metallic fuel was used. For configurations with U-10Zr fuel, four different core diameters were used, while three different core diameters were used for the U-Pu-10Zr cases. The largest diameter represents eight rings of driver assemblies. Smaller diameter cores were created by removing either one or two rows of driver fuel. For the smallest core, the control-rod locations were repositioned so they were not on the core periphery. An additional seven-ring case was evaluated by substituting the 61-pin assembly design developed for the original LBE option. This was done to evaluate the effect of a reduction in the fuel volume fraction. In all cases, the core diameters reported in Table 4 include the space occupied by experimental assemblies and control rod positions.

For the LBE-cooled option, three different core diameters were evaluated with U 10Zr fuel. These include the original nine-ring core and two smaller core diameters. The smallest, seven-ring core was evaluated a second time by substituting the 91-pin assembly design developed for the sodium option to measure the effect of increasing the fuel volume fraction on the LBE cases.

The helium-cooled option based on the original GA assembly design is referred to as the "standard" case; while the option based on a reduced pin pitch is referred to as the "compact" case. For the standard, helium-cooled option, both UO_2 and $(U,Pu)O_2$ fuel was used. The GFR cores tend to be significantly larger than the sodium or LBE cases because the individual assemblies are larger. Both larger (nine-ring) and smaller (down to four-ring) cores were evaluated for the standard helium case. Control rod locations were altered for the two smallest cores so that they were not on or outside the core periphery.

For the compact, helium-cooled option, only UO_2 fuel was used. The compact case had a 100 cm active height and a seven-ring layout. Both larger (eight-ring) and smaller (down to five-ring) cores were evaluated. Again, for the smallest core, the control-rod locations were repositioned so they were not on the core periphery.

TABLE 4
. Summary of Fissile Enrichment and Fast Flux as a Function of Core Size, Coolant Choice, Fuel Type, and Fuel Volume Fraction.

Coolant	Fuel Form	Volume Fraction [a]	Diameter [b] (cm)	Enrichment [c] (%)	Fast Flux [d] (10^{15} n/cm^2/s)
Sodium	U-10Zr	0.5055	89	23.9	1.09
			101	20.5	0.933
			119	17.8	0.737
			136	16.3	0.605
		0.3203	119	26.5	0.779
	U-Pu-10Zr	0.5055	89	16.2	1.25
			101	13.9	1.06
			119	12.1	0.828
LBE	U-10Zr	0.3203	119	23.9	1.03
			136	21.2	0.851
			154	19.5	0.719
		0.5505	119	16.8	0.884
Helium	UO_2	0.3080	136	39.1	0.639
			170	30.0	0.468
			192	26.5	0.389
			226	23.8	0.292
			259	22.1	0.227
			292	20.9	0.181
		0.3938	150	24.7	0.550
			170	21.9	0.454
			200	19.9	0.339
			229	18.6	0.262
	$(U,Pu)O_2$	0.3080	136	24.9	0.754
			170	19.7	0.545
			192	17.8	0.450
			226	16.3	0.336
			259	15.3	0.259
			292	14.6	0.205

a. Fuel volume fraction represents the smeared volume fraction in driver assemblies only.
b. Core diameter represents a circular area that is equivalent to the combined area occupied by driver assemblies in addition to experimental and control rod positions.
c. For plutonium-fueled cases, fissile enrichment includes contributions from Pu-239 and Pu-241.
d. Experimental fast flux is averaged over the active axial length of the central (group of) experimental position(s).

IV.F.2. Results

The fast flux at EOEC, averaged over the central experimental position(s) is plotted in Figure 1 for all cases. Fissile enrichment requirements are plotted in Figure 2. The obvious trend is that one can achieve higher flux values for smaller diameter cores because power density increases as core volume decreases. Similarly, smaller cores generally have higher fissile enrichment requirements, but the enrichment also depends on fuel type, fuel volume fraction, and coolant.

When considering the uranium-fueled results alone (Figure 5-1, closed symbols) for sodium and helium, the average fast flux in the central experimental position(s) shows virtually no dependence on fuel type (metallic or oxide), coolant type (sodium or helium), or fuel volume fraction (standard or compact; 61-pin or 91-pin). Unlike the fast flux, the fissile enrichment depends strongly on fuel volume fraction and coolant.

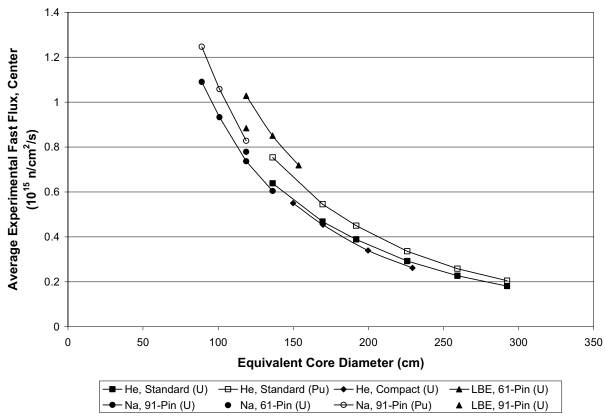

Fig. 1. Average Fast Neutron Flux in the Central Experimental Position(s) as a Function of Core Diameter for Various Configuration Options at 100 MWth.

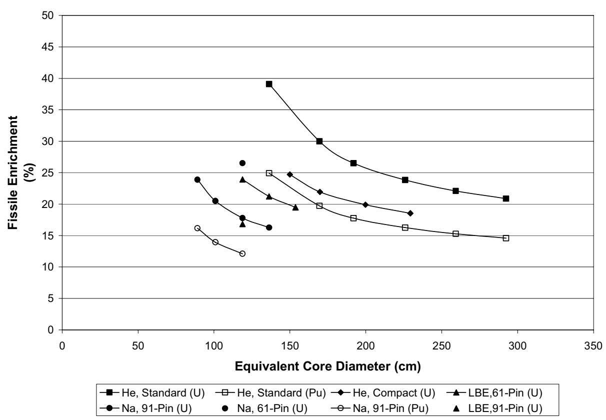

Fig. 2. Fissile Enrichment Requirements as a Function of Core Diameter for Various Configuration Options at 100 MWth.

Average power density in a reactor can be written as

$$p = C\Sigma_f \phi,$$

where C is a constant that represents the energy released per fission, Σ_f is the macroscopic fission cross section, and ϕ is the *total* neutron flux. For any given concept, the total flux will be directly proportional to the power density, and inversely proportional to the macroscopic fission cross section. The macroscopic fission cross section is defined as,

$$\Sigma_f = \sum_i N_i \sigma_f^i ,$$

where N_i is the isotopic number density and σ_f is the appropriate one-group microscopic fission cross section for fissile isotope *i*. The macroscopic fission cross section is related to the fissile inventory, *I*, by $I=NV$, where V is the core volume. Since the power level has been fixed at 100 MWth in all cases, power density will increase with decreasing core diameter. For a given core volume and power density, then, the case with the lowest macroscopic fission cross section (lowest fissile inventory) will tend to have the highest flux. Assuming the flux spectra are similar, these trends will also apply to the fast flux. Sensitivity of the fast flux and enrichment to various design options are described next.

IV.F.3. Conclusions

It is no surprise that in order to achieve a high fast flux, one must develop a core with a high power density, irregardless of other factors such as coolant, fuel form, and fuel volume fraction. However, in order to limit fuel enrichment requirements, a high heavy metal loading is needed in a core with reduced leakage. High heavy metal loadings can best be achieved with metallic fuel and a high fuel volume fraction. Reducing core leakage can be achieved with a high fuel volume fraction and an effective reflector. Further increases in flux levels can be achieved by utilizing plutonium-based fuels, but this benefit can be obtained regardless of the option under consideration.

Because of their superior thermophysical properties, liquid metal coolants are an obvious choice for cooling a high power density core with a high fuel volume fraction and low coolant volume fraction. In the absence of any other considerations, LBE provides an advantage over sodium in that it is an effective reflector and reduces core leakage. This in turn reduces enrichment requirements and enhances the experimental flux. However, LBE is subject to flow velocity limitations, which leads to increased coolant volume fractions and a larger core. As a result, one can expect sodium and LBE cases to be similar in their abilities to deliver a desired flux level.

In addition to achieving a higher fast flux, developing a small core size has several other advantages. These include a reduction in physical plant size and improved economics. But a smaller core also has the advantage of being on the steeper part of the flux curve shown in Figure 5-1. In this region, one might envision future plant modifications that significantly enhance fast flux production.

V. COMPARISON OF COOLANT PROPERTIES

The choice of the reactor coolant will likely play the most significant role in the design of the FITR. It will strongly influence the physical layout of the reactor, neutronic behavior of the core, the choice of fuel, cladding and other structural materials and mechanical components. This section discusses the comparison of the properties of the coolants that were proposed for FITR design concepts. In addition, it supports and explains the Coolant Comparison Matrix, Table 5. The groups of properties considered in this evaluation include heat transfer performance, neutronic performance, inherent safety characteristics, compatibility with materials, experience, maintenance, and cost. Each coolant was given a relative score from 1 to 9 for each property, with 9 representing the highest or most favorable ranking for that property. An importance factor was also assigned to each property. The total score for each coolant for each group of properties is arrived at by multiplying the score for the coolant for a property by its importance factor, then summing the resulting products for that coolant. The total score for each coolant is a summation of its scores for each group of properties.

TABLE 5
Coolant Comparison Matrix

PROPERTY	METRIC	IMPORTANCE FACTOR	Sodium	LBE	Steam	He
HEAT TRANSPORT PERFORMANCE						
Thermal conductivity	Excellent/Medium/Poor (9/5/1)	7	9	7	1	3
Volumetric Heat Capacity	Excellent/Medium/Poor (9/5/1)	7	8	9	3	1
Pumping Power Š for each concept	Low/Medium/High (9/5/1)	2	7	9	2	1
Pressurization	No/Yes (9/1)	4	9	9	1	1
Total Score for HTP		20	169	166	36	34
NEUTRONIC PERFORMANCE						
Spectrum	Hard/Soft (9/1)	6	8	9	5	7
Parasitic Absorption	Low/High (9/1)	4	8	6	5	9
Leakage	Low/High (9/1)	4	8	9	3	3
Total for Neutronic Performance		14	112	114	62	90
INHERENT SAFETY CHARACTERISTICS						
Reactivity void coefficient	Relative Ranking (9/1)	4	5	9	1	3
Possibility of phase change	No/Yes (9/1)	5	6	8	1	9
Emergency Decay Heat Removal	Readily established Yes/No(9/1)	8	9	9	1	1
Total Score for ISC		17	122	148	17	65
COMPATIBILITY WITH MATŐLS						
Compatibility with Oxide Fuel System Š UO2 and MOX	Yes/No (9/1)	4	8	7	7	9
Compatibility with metallic fuel system Š U10Zr (UPu-10Zr)	Yes/No (9/1)	4	9	4	3	9
Corrosion of Steels	None/Medium/High (9/5/1)	7	8	4	5	9
Erosion	No/Yes (9/1)	4	9	1	5	5
Liquid Metal Embrittlement of Materials	No/Yes (9/1)	1	9	1	9	9
Total Score for Compatibility W/M		20	169	77	104	164
EXPERIENCE						
Experience base applicable for reactor system	High/Medium/Low (9/5/1)	10	9	3	5	5
Total Score for Experience		10	90	30	50	50
MAINTENANCE						
Coolant Activation	Low/Medium/High (9/5/1)	2	5	1	7	9
Transparency	Yes/No (9/1)	2	1	1	9	9
Purification and chemistry control system	None/Average/Extensive (9/5/1)	2	5	1	5	8
Requires heaters	No/Yes (9/1)	2	1	1	1	9
Coolant Leakage control	Easy/Medium/ Difficult (9/5/1)	2	9	9	5	1
Chemical reactivity with air/water	Low/Medium/High (9/5/1)	2	1	7	9	9
Chemical Toxicity	No/Yes (9/1)	2	5	1	9	9
Removal of residual coolant for maintenance	Easy/Medium/ Complex (9/5/1)	2	5	1	9	9
Total Score for Maintenance		16	64	44	108	126
COST						
Coolant Cost (volume x $/liter)	Low/Medium/High (9/5/1)	1	5	1	9	7
Coolant Replacement Cost	Low/Medium/High (9/5/1)	1	9	9	9	1
Availability	Abundant/Limited (9/1)	1	9	1	9	5
Total Score for Cost		3	23	11	27	13
Overall Score		100	749	590	404	542

V.A. Heat Transport Performance

The primary function of the coolant is to remove heat from the reactor core. The major heat transport characteristics of the proposed reactor coolants are summarized in Table 6. From this table it is clear that sodium and LBE have significantly better heat transport characteristics than steam and helium, which will result in the lower cladding surface temperature in SFR and LFR cases. The high ratings for sodium and LBE for thermal conductivity (9 and 7) and for volumetric heat capacity (8 and 9 respectively) reflect those properties relative to the two gases, which rated either 1 or 3 for those properties. Additionally, sodium and LBE are low pressure systems whereas helium and steam both require substantially higher pressures, thus the highest ratings for the liquid metals, and the lowest for the gases.

It should be noted that the pumping power shown in Table 6 was calculated specifically for each proposed FITR design concept. In particular, the open pin lattice was used in the LBE model due to the 2 m/s limit imposed on coolant velocity to avoid excessive erosion of core structural materials. This contributed to a reduction of the pumping power value for LBE coolant compared to what it would be with the tight pin lattice cores.

TABLE 6

Heat Transport Characteristics of Reactor Coolants

Property	Na	LBE	Steam	He
Thermal conductivity (W/m K)	70.8	14.0	0.0649	0.284
Volumetric heat capacity[a] (J/m^3 K)	1.08E+06	1.49E+06	6.12E+04	2.47E+04
Pumping power (W)	3.409E+04	1.415E+05	2.969E+06	1.250E+07
Pressurization, (MPa)	~0.1	~0.1	~7	~8.6

a. Evaluated at 425EC. The pressure was 7.0 MPa for the gases and 0.1 MPa for the liquid metals

V.B. Neutronic Performance

The neutron absorption and scattering properties of the different coolant choices will impact core design by affecting core size, enrichment, and the hardness of the flux spectrum. Since the primary purpose of the FITR is to produce fast neutrons (i.e. a hard spectrum) the choice of coolant must be consistent with this mission. With the possible exception of steam, all of the coolants in Table 5 are suitable for use in a fast-spectrum reactor. In addition, the thermophysical properties, which characterize the ability of a coolant to remove heat, will have a larger impact on core size and enrichment. Therefore, the significance of neutronic performance in the ranking of the different coolants is relatively minor and is limited to approximately 14% of the total coolant ranking score. However, there are still differences between the coolant choices, and those differences are characterized by the following metrics.

V.B.1. Flux Spectrum

As described above, the primary purpose of the FITR is to produce a fast neutron flux of at least 1.0×10^{15} n/cm^2/s for irradiation testing purposes. Fast neutrons are those that have kinetic energy above 0.1 MeV. The fraction of the neutron flux that is above this energy threshold is a measure of the "hardness" of the flux spectrum. In a fast reactor, the coolant must not act as a moderator; therefore, high atomic weight coolants are desirable to prevent neutron slowing down.

To measure the hardness of the flux spectrum, the fast neutron flux fraction is evaluated at the middle axial position of the central experimental assembly for the SFR, LFR, and GFR concepts. Because of its high atomic mass (for both lead and bismuth), LBE has the highest fraction at 71.5% and is given a ranking of 9. For sodium and helium, it is 69.4% and 67.2% respectively. Consistent with this, sodium is given a rank of 8 and helium is given a rank of 7. Steam results in a softer spectrum than helium and is rated a 5.

V.B.2. Parasitic Neutron Absorption

Ideally, the only source of neutron absorption in a reactor would be the heavy metal in the fuel. Coolant and structural materials are usually chosen to limit the amount of parasitic absorption, as it has a negative impact on enrichment and core size. In the case of the FITR, parasitic absorption by the coolant in experimental channels is also undesirable. Therefore, the extent of parasitic absorption introduced by each coolant is measured by evaluating the flux-weighted macroscopic capture cross section at the middle axial position of the central experimental assembly for the SFR, LFR, and GFR concepts.

Helium clearly deserves a ranking of 9 for this metric because its capture cross section is zero. It introduces no parasitic absorption. Sodium introduces a small amount of parasitic capture, but it is less than 25% of the total parasitic capture in the experimental assembly, which includes the effects of the duct wall. (The experimental positions are modeled as duct wall and coolant only). Therefore, it is given a ranking of 8.

LBE introduces a larger amount of parasitic absorption than does sodium. Its macroscopic capture cross section is 5.5 times larger than sodium. Most (roughly 60%) comes from Bi-209. Combined with the structural components, LBE represents two-thirds of the parasitic absorption in the experimental assembly. Therefore, it is given a ranking of 6. Steam was arbitrarily given a mid-point ranking due to lack of information.

V.B.3. Neutron Leakage

Coolants with high atomic masses and large scattering cross sections make effective reflectors to prevent leakage from the core. Like absorption, leakage is a parasitic effect that has a negative impact on enrichment and core size. Within experimental assemblies, long channels of coolant provide axial streaming paths for leakage, which diminishes the available flux for testing. Furthermore, leakage tends to have a stronger effect on high-energy (fast) neutrons. Therefore, the amount of axial leakage in the central experimental assembly of the SFR, LFR, and GFR concepts is used to rank this metric.

LBE, because of its high atomic mass and effective scattering cross section, is the most effective reflector and is given a ranking of 9. Axial leakage in the case of the SFR (sodium) is only slightly higher than for LBE (less than a factor of two) and is given a ranking of 8. Helium, because of its relatively low number density, is a very poor reflector. Axial leakage in the GFR concept is six times higher than in the LFR concept. When helium is removed from the experimental assemblies, axial leakage increases by less than 4%. This shows that helium provides very little axial reflection within the experimental assembly, and it is given a ranking of 3. Steam was given an arbitrary rating equal to helium.

V.C. Inherent Safety Characteristics

Inherent safety as used herein is defined as reactor system response to design basis or beyond design basis accident events (e.g. loss of flow, loss of heat sink, with and without scram, etc.), in which the reactor simply shuts itself down and passively rejects decay heat with no operator actions or active safety system activation. Inherent reactor safety is a direct result of a combination of reactor fuel and coolant properties and reactor geometry.

V.C.1. Reactivity Void Coefficient

The value of the coolant void reactivity worth is strongly influenced by the type of fuel and reactor core geometry. Though very low and even negative reactivity void coefficients are possible with sodium coolant for certain reactor and fuel combinations, in general, the reactivity effect of coolant voiding is considered more positive for sodium than for LBE due to the smaller neutron loss in the LBE. In addition, LBE has higher boiling temperature (1670°C) compared to sodium (883°C), which helps to avoid positive reactivity insertions that are often associated with coolant boiling during accidental scenarios. Thus, LBE is given a 9, and sodium a 5. Helium has minimal void coefficient, however certain fuel and reactor geometry combinations can lead to positive reactivity insertions upon helium voiding. Steam has unfavorable coolant reactivity coefficient. Both helium and steam are given rankings of 1.

V.C.2. Possibility of Phase Change

Helium has a significant advantage relative to steam since it will not undergo phase transformation at reactor operating temperatures, and therefore has a rating of 9. Steam, on the other hand, could easily turn into a liquid if it were overcooled, resulting in a rating of 1. Similarly, LBE has an advantage over Na, since the latter has lower boiling temperature, but do not need pressurization to prevent boiling under normal and most abnormal operating conditions, and are therefore rated as 8 and 6 respectively.

V.C.3. Emergency Decay Heat Removal

Both sodium and LBE cooled reactors have the excellent heat transfer characteristics and are effective at removing heat through natural circulation thus providing ability to passively reject decay heat in emergency situations. For instance, it was experimentally demonstrated that the sodium cooled EBR-II reactor was capable to safely shutdown and remove decay heat effectively in loss of flow and loss of heat sink accidents (Planchon et al. 1988). LBE has an additional advantage with a very high boiling point. Both the SFR and LFR receive ratings of 9 while emergency decay heat removal for both helium and steam have not been established, but will most likely require engineered backup systems for emergency cooling and therefore are given ratings of 1.

V.C.4. Compatibility with Materials

Compatibility of reactor materials plays an important role in the design of any reactor. It is a key element in the decision making process during selection of fuels and structural materials. The objective in this undertaking is to minimize chemical interaction between materials so that the structural integrity of the components is not jeopardized.

The list of the candidate fuels and structural materials for proposed Fast Irradiation Test Reactor (FITR) concepts is rather limited since the choice was mainly based on the requirement of previous qualification (or near qualification) for fast reactor applications. It includes metallic U10Zr (UPu10Zr) and oxide UO_2 (MOX) fuels, HT9 and AISI 316 steel claddings and sodium or helium thermal bonds. The remaining reactor components, such as containment vessel, pumps, pipes, valves, etc., are typically made out of austenitic stainless steels, such as type 304 and 316. Other types of materials were also considered but it was concluded that their development and qualification for use in a reactor will require significant effort and time.

Despite the fact that all of the materials considered were used in fast reactors, they were tested or used only in specific reactor systems, with unique fuel, coolant and operating condition combinations. In fact, most of them were qualified or otherwise accepted for use in sodium-cooled fast reactors, which were predominant in the past (Ryskamp et al. 2004). Lead-bismuth cooled reactors were only built and operated in Russia; therefore no practical reactor experience with lead-bismuth (lead) coolant is available in the U.S. Experience with helium coolant is limited to just a few reactors around the world. These include Fort St. Vrain (USA), Unit 1 at Peach Bottom Power Station (USA), Dragon (UK), AVR (Germany), THTR (Germany), HTTR (Japan). Most of the experience with water/saturated steam coolants comes from BWR/LWR thermal reactors and there is no experience with single-phase (superheated) steam coolant for fast reactor applications.

For the FITR, each specific fuel, cladding, and structural material combination considered will require more extensive evaluation for the selected coolant and set of operating conditions established. The matrix rankings for compatibility with materials shown in Table 5 are based on operating experience, experiments, and tests performed, where available, and information from researchers on studies conducted to evaluate the effects of interest.

V.C.4.i. Compatibility with Oxide Fuel System

Oxide fuel systems considered should perform reasonably well with any of the coolants under consideration, as reflected in the ratings which go from 7 to 9. Helium is ranked the highest at 9 due to its inert nature.

V.C.4.ii. Compatibility with Metallic Fuel System

Sodium and helium are both highly compatible with metallic fuel based on extensive experience and are both rated at 9. Tests and studies with LBE indicate that solubility of metallic fuel system materials in LBE will be an issue, although some materials variations and precise oxygen may alleviate some problems. It is rated at 4. Much more development is required. Although there is no data on the compatibility of superheated steam with metallic fuel, it is known that uranium metal oxidizes in water vapor, and the oxidation rate of uranium with water vapor is significantly higher than the reaction rate of uranium with dry air or oxygen gas. Steam is therefore given a 3 in this category.

V.C.4.iii. Corrosion

Sodium is compatible with austenitic steels (type 316 and 304) in the temperature ranges expected for the FITR. Elevated temperatures with oxygen contamination of the sodium will create conditions for corrosion, however, thus the score of 8 versus 9 for helium. Corrosion of structural materials is a significant concern for LBE but at the temperatures ranges considered for FITR, should be controllable with oxygen control schemes. Thus the LBE rating is below the mid-point at a 4. Steam is rated at the mid-point. At higher steam temperatures there will be a significant increase in the rate of formation of the protective oxide scale. Because of the difference in the expansion coefficients between the base metal and oxide scale, the later would have a tendency to spall-off, which in turn might lead to solid-particle erosion damage of steel surfaces.

V.C.4.iv. Erosion

At higher flow rates, erosion can become an issue with both steam and helium, thus the mid-range rating of 5. For LBE, however, erosion is a concern and the flow velocities must be kept below 2 meters/sec which impacts core lattice spacing and heat removal, hence the rating of 1 for LBE. Erosion has not been observed as a concern for sodium, thus the rating of 9.

V.C.4.v. Liquid Metal Embrittlement

Liquid metal embrittlement is not an issue with either helium or steam, and has not been observed in sodium systems at or above the temperatures expected for the FITR, thus ratings of 9 are given for these coolants. LBE, however, is susceptible to causing liquid-metal embrittlement and is given a rating of 1.

V.C.5. Experience

There is extensive experience with the use of sodium as a fast-reactor coolant in the US and abroad, hence a rating of 9. There is also a great deal of experience with water/steam as a reactor coolant, but not in a fast reactor. Helium has also been used as a reactor coolant in the US, but not in a fast reactor. The two gases are given a mid-range rating. LBE has an experience base as a fast reactor coolant in Russia, but not in the U.S. LBE ranks lowest of the four, due to its lack of experience in the U.S. as a reactor coolant but at 3 is only two points below the gases since an experience base does exist.

V.C.6. Maintenance

V.C.6.i. Coolant Activation

Coolant activation presents a special concern in the design of the FITR since additional measures should be taken to protect personnel from exposure to radiological hazards during reactor normal operation and maintenance. Such protective measures might include additional shielding of the primary loop, online purification systems of the coolant to remove radioactive products, use of protective clothing and equipment, etc.

Helium will not be activated. Steam will become slightly radioactive mainly through production of ^{17}O. Activation of sodium is more significant due to production of short lived gamma emitting 24Na isotope (T1/2=15 hrs). This has shown to be easily accommodated by delaying hands-on maintenance activities for 2-3 days following reactor shutdown. LBE will be the most radioactive due to accumulation of polonium isotopes.

Therefore the range of ratings goes from helium at 9 to LBE at 1 with the others in between.

V.C.6.ii Transparency

Compared to helium and steam, sodium and LBE are non-transparent liquids and will require special instrumentation to assist in various reactor maintenance operations and during refueling. Thus, the highest ratings are given to the gases and lowest to the liquid metals.

V.C.6.iii. Purification and Chemical Control

The primary goal of coolant purification system is to keep the coolant free from chemical and radioactive contaminants. The majority of the chemical contaminants will be associated with corrosion and erosion of material surfaces, failed fuel/cladding, and coolant oxides, while coolant activation products will represent the main fraction of radioactive contaminants with some potential for fuel particles and cesium from fuel failures.

Helium is inert and non-corrosive to structural materials, therefore it will not require a sophisticated coolant purification system, thus earning a high rating.

Corrosion of steels is not a significant concern in sodium-cooled reactors and it can be successfully mitigated by keeping the concentration of oxygen below ~10 ppm. Purification of sodium is easily accomplished by means of the cold trap (Spencer 2000). Even though these are proven systems with a high level of confidence, since measures such as these are required, sodium is given a mid-point rating of 5.

A more complex purification system is required in LBE case, due to severe corrosion problem of structural materials in LBE. In addition, to maintain protective oxide films on the surface of steels, oxygen concentration in the coolant should be maintained in the optimal range. The LFR will require additional coolant purification systems to remove radioactive polonium. For these reasons, LBE is rated the lowest.

Steam can also be quite corrosive to structural materials and will require a special purification system. It should, however, be less complex that LBE requires. Steam is given a mid-rating.

V.C.6.iv. Requirement for Heaters

Both sodium and LBE are solid at room temperature therefore heat must be supplied to these coolants at all times to prevent them from accidental freezing. Heating must also be supplied to steam to prevent it from overcooling and changing into liquid. Helium is rated high at 9 while the other three received the lowest rating for this criterion.

V.C.6.v. Coolant Leakage Control

Considering significant pressurization of the helium coolant (~8.6 MPa) and its small molecule size, it will be extremely difficult to prevent the leakage of helium gas, resulting in the lowest rating. A steam-cooled reactor will experience similar coolant leakage problems but should be somewhat more manageable given the vast experience with steam systems, hence a mid-rating for steam. There should be no significant problems associated with the leakage control of sodium and LBE, since both coolants are liquids and will be kept at almost ambient pressure, thus the highest ratings for the liquid metals.

V.C.6.vi. Chemical Reactivity with Air/Water

Helium is inert and does not react with either air or water. Steam is also compatible with air and water. However sodium reacts rapidly with water or moist air according to the following reaction: $Na + H2O = NaOH + 0.5\ H2$. For this reason sodium systems use an inert cover gas over any free surface and special measures are taken to prevent and detect sodium leakage. Sodium is given the lowest rating for this criterion, while steam and helium are rated the highest at 9.

LBE mildly reacts with moist air, leading to formation of the oxide scale in the coolant. Reaction of LBE with water results in formation of hydroxides. Both reactions are not violent. Thus, LBE is rated a 7 which is below steam and helium, but significantly higher than sodium (1) in this category.

V.C.6.vii. Chemical Toxicity

Due to toxicity of lead-bismuth special protective measures should be taken by reactor personnel to avoid poisoning during repairs and maintenance operations and is rated as 1. Sodium does not have the toxicity problem of lead-bismuth, but since sodium hydroxide is formed from the reaction with sodium and moisture, personnel precautions are typically required for working with sodium systems. Thus sodium is given a mid-rating, and the two gases, helium and steam, which are non-toxic (but do of course have their own personnel hazards) are rated at 9.

V.C.6.viii. Removal of Residual Coolant During Maintenance

Since FITR is an irradiation test facility, special attention will be given to the development of techniques for removal of residual liquid metals from the surface of test materials/rigs. Sodium is traditionally removed with steam or alcohol in a straightforward manner and is given a rating of 5. Residual LBE can be removed by boiling in hot glycerol, washing with sodium or using ethanol-acetic acid-hydrogen peroxide mixture and is somewhat more complex to remove than sodium, thus is rated at 1. This should not be an issue with the gases and they are therefore rated at 9.

V.C.7. Cost

While coolant cost is a consideration in the selection of the best coolant for the FITR, for a one-of-a-kind facility it should not be a significant deciding factor, unless the cost would be a significant fraction of the cost of the facility. This is not the case for the coolants options discussed in this section. Due to the minor significance of coolant cost to this project, coolant cost including availability and replacement cost makes up only 3% of the coolant rating score. The low weighting of this factor also reflects the lack of development of each concept, so quantity of coolant required is unknown at this time.

V.C.7.i.Coolant Cost

In 1998 the cost of cubic meter of sodium and LBE was \$0.145 and \$5.62, respectively (Spencer 2000). In 2003 the cost of commercial grade lead was 43.8 cents per pound (North American Producer) and the cost of bismuth was \$2.87 per pound (USGS 2005). The cost of grade A helium (99.995% or greater purity) was between \$2.16 and \$2.34 per cubic meter in 2003. The ratings reflect a volumetric cost with steam given the highest rating (lowest cost) and LBE the lowest rating (highest cost). The importance factor is low.

V.C.7.ii. Coolant Replacement Cost

A periodic replacement of gas coolants might be required due to their leakage from the reactor system. The score reflects the thinking that helium may have to be replaced, or made up at a higher rate and higher cost than the other options, thus it is rated a 1 with the others at 9.

V.C.8. Conclusion

As shown in Table 5, the total relative rankings of the coolants evaluated indicate that sodium ranks the highest followed by LBE, helium, and steam. These ratings reflect to a large degree the superior heat transfer qualities and inherent safety characteristics of both liquid metals compared with the gases, and the relatively broad US experience base with sodium as a reactor coolant. Sodium also gains an additional edge over LBE in the area of materials compatibility, even though it gives up some ground in chemical reactivity with air and water.

While the ratings include some degree of subjectivity, the exercise itself is very helpful in comparing the differences among the coolants. Also, the overall rating scores have sufficient spread between coolants to provide a level of confidence that the order of the rankings is likely correct given the criteria, the metrics assigned, and the importance factors assigned to each criterion.

V.D. Down-Selection Process, Criteria, and Results

V.D.1.Introduction

A major portion of the project effort was directed toward establishing sufficient technical information for the variety of concept options to perform a ranking of the options based on their ability to meet the criteria developed for the FITR facility. One of the tasks was to develop specific criteria and a rating system so that comparisons among the options could be made on objective technical and programmatic bases. This section discusses the down-selection process, the criteria, metrics, importance factors, and the results of the process.

V.D.2. Criteria Development

The drivers behind development of the FITR are many as previously discussed. However, there is a set of key requirements that the FITR facility must meet to be able to accomplish its objectives and to be an attractive investment for the DOE and other potential stakeholders. These basic requirements were separated into two categories of criteria: 1) Essential; and 2) Important. The "Essential" category was used as the primary discriminator to determine whether or not a specific concept option would be considered for further evaluation. The "Essential" category was made up of a short list of necessary attributes or criteria that a concept was required to have or to be able to meet. If a concept option could not meet a specific element in this category, then it was eliminated from consideration for further evaluation at least as part of the continuation of this project. Since the concepts being evaluated have had little development, a score of "Maybe" was allowed which allowed a concept to pass that particular criterion if a clear "Yes" or "No" answer was not possible. Note that if a concept was eliminated as a result of this screening process, that does not indicate that further evaluation of that concept should not be pursued, just that it did not meet the criteria for continuation as part of this project.

The "Important" category included more detailed technical criteria for which metrics and relative importance factors were developed. This provided a means to establish a numerical score for each concept option for each criteria such that by summing the scores for each concept option, a relative ranking would be established. Although it was not necessary to establish a relative ranking for those concept options that did not pass the "Essential" screen, a relative ranking score was provided for all options to help show the relative strengths and weaknesses, and potential viability of each one. While the cost of the FITR is an important element (both construction and operating and maintenance costs), it was determined that the concepts were not sufficiently developed at this time to provide rankings. Therefore, the criteria were left in the table, but no rank or score was given.

Additionally, it was felt that the option for the FITR to support a prototype demonstration of a Gen IV reactor may be an important factor to DOE, so a score was provided for that as a criterion. However, since prototype demonstration was not a stated objective for the FITR, this scoring was not included in the total score and rank for the concept options. It is included for information only.

Development of the criteria metrics and the importance factors was done by roundtable discussions and debates among technical experts from INL and ANL. While this process inherently involves a degree of subjectivity, the logic used in selection of the criteria should be understandable, and the metrics and the importance factors assigned do provide a quantitative method to capture differences among options. While the exact numbers for these metrics and factors could be debated, the relative differences derived from this process have solid technical grounds given information available at this time.

The down-selection criteria and metrics were developed from the following list:

1. Flux level averaged over primary irradiation position volume(s) (fast flux must be greater than 1E15 n/cm2-s average (>0.1 Mev) over primary irradiation volume.
2. Estimated reactor startup date assuming a project start date of October 1, 2005 and the schedule includes duration for R&D development. This metric provides an indication regarding whether the irradiation test reactor concept can meet the irradiation start dates required of the various Generation-IV concepts. No more than three years R&D required to proceed with design.
3. Dedicated irradiation volume available. Must have irradiation volume of at least 2 liters that meets minimum average flux requirement.
4. Type of irradiation facilities supported. Must have at least one primary irradiation facility that has provisions for an independent coolant system isolated from primary coolant.
5. Accessibility for experiment handling including potential for on-line experiment insertion/removal.
6. Selected fuel form and composition, status of development and qualification. Minimal qualification required.
7. Coolant properties, e.g., toxicity, flammability, compatibility with fuels and materials, transparency, etc.

8. Coolant experience base.
9. Structural materials, status of development or qualification. See No. 2.
10. List of research and development tasks required to establish a high-confidence design including estimated time to complete R&D.
11. Availability/reliability factors, e.g., expected shutdown frequency and duration for refueling, component change-out, required maintenance. (Annual availability factor greater than 80 %.)
12. Transient performance and passive safety characteristics.
13. Potential to support prototype reactor testing and demonstration.

TABLE 7

Concept Down-Selection Matrix—Essential Criteria

Essential Criteria	Metric	Importance Factor	PTR-(G)	PTR-(L)	SFR	LFR	GFR
Ave Fast Flux > 1E15 over a 10 liter or greater volume. Two 1m x 8cm test volumes.	Yes/No	Must	Y	Y	Y	Y	Y
Support test loop with separate coolant at ave. fast flux > 1E15	Yes/No	Must	Y	Y	Y	Y	Y
Systems and Components R&D reqÕd < 3 years Š This criteria supports the target of a 2015 reactor startup for GenIV irradiation testing needs. Structural materials fall under this criteria.*	Yes/No	Must	No, level of design maturity precludes a Yes	No, level of design maturity precludes a Yes	Y	Y	No, level of design maturity precludes a Yes
Fuel Qualification Status for the first core load Š assume that startup is under DOE regulations. U10Zr, Mixed Oxide, UO2 are available for initial core load.*	FITR Reactor Startup with a qualified or near qualified fuel under DOE	Must	Y	Maybe, uncertainty in compatibility may preclude achieving qualification status within the 3-year target	Yes Š both U10Zr and Oxide have been used as fast reactor fuels	Maybe, uncertainty in compatibility may preclude achieving qualification status within the 3-year target	UO2 or MOX

*3 years of R&D budgeted to confirm feasibility, compatibility of fuel forms with coolants

TABLE 8

Concept Down Selection Matrix--Important Characteristics

Important Characteristic	Metric	Importance Factor	PTR-(G)		PTR-(L)		SFR		LFR		GFR	
Availability > 80%	Highly likely = 9 Not likely = 0 (9/5/0)	10	3	9	3	9	9	9	7	9	9	9
Passive Safety Characteristics Š able to accommodate protected accidents without fuel/cladding/core damage	Highly likely = 9 Not likely = 0	10	2	5	4	7	9	9	9	9	2	5
On-line Experimental Handle or Rabbit System	Highly likely = 9 Not likely = 0	5	9	9	9	9	3	7	3	7	2	6
Coolant Properties **	Relative rank from Coolant Matrix	15	5	5	7	7	9	9	7	7	5	5
Structural Materials Available	Available (yes) or R&D reqÕd <1 year (no) 9/1	10	7	7	7	7	9	9	7	7	9	9
Overnight Capital Costs	TBD, but must be <<$1B	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD
Operating and Maintenance Costs	TBD, but no more than ATR	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD	TBD
Support Gen-IV Prototype Demo	Direct Š Yes = 9, Indirect Š No = 0	0	3	3	3	3	9	9	9	9	7	7
Total Score			240	330	290	380	420	440	350	390	285	335
Average Score			285		335		430		370		310	
Delta (High-Low)			90		90		20		40		50	

V.D.3. Down-Selection Matrix

The various concept options, the essential and important criteria, the metrics assigned to each criterion, the criterion importance factors, and the computed ranking scores are shown in the FITR Down-Selection Matrix, Tables 7 and.8. Table 7 shows the essential criteria established for FITR, Table 8 shows the important criteria established as ranking measures for FITR options.

V.D.4. Down Selection Results

As shown in Table 7, the gas-cooled option (GFR) did not meet the essential requirements screen primarily due to the technical uncertainties that will require several years of research and development before a gas-cooled fast reactor could be considered a viable option. This is re-enforced by discussions in the Gen IV R&D Roadmap, and by discussions with experts in the fuel and safety analysis areas. While the GFR may be a viable technology at some point in the future, it is not a suitable candidate for the FITR which requires relatively near-term deployment implying limited required R&D and essentially no questions of feasibility. Even though the GFR did not pass the initial screen, scores in the "Important" criteria area (Table 8) were still developed to provide a more complete comparison of all options considered in this year's project.

The PTR was evaluated with helium (PTR-G) and LBE (PTR-L) as coolant options. Previously the FITR project evaluated both a helium and steam-cooled PTR. This year, steam was dropped from consideration for the PTR option due to inferior characteristics relative to helium and LBE was added as a potential coolant option for the PTR. The PTR is a novel concept that has worthwhile features that could be advantageous particularly in an irradiation/test reactor including the potential for at-power experiment changeout and re-fueling similar to the Canadian CANDU reactors from which the concept is derived. However, the PTR did not meet the essential requirements screen primarily due to the time required to develop the concept and address feasibility issues such as reactivity control, and at-power re-fueling in a fast spectrum with either LBE or He as the coolant. Even though the PTR did not pass the essential criteria screen, the PTR-L option ranking score was fairly close to the LFR score.

The two highest-rated concepts were the SFR and the LFR. The ranking of the SFR reflects the maturity of the technology involved and the relatively extensive experience base that sodium as a reactor coolant has in the US and elsewhere. The fact that both metal and oxide fuel have been developed, qualified, and operated in SFRs in the US, also tips the scales in favor of SFRs. The top ranking is reflective of the high degree of technical certainty about the technology and the minimal amount of R&D required to make the SFR a viable option to meet the FITR performance criteria. Benefits of sodium also include its compatibility with conventional structural materials. Drawbacks of sodium include its rapid reaction with water including moisture in the air, which requires development of design features to minimize the potential for exposure of sodium to air, and to detect sodium leakage if does occur.

The LFR came in second in the rankings. It also has considerable promise and a significant experience base in Russia. Certain characteristics of LBE rate higher than sodium, particularly its high boiling point and its relative compatibility with air and water. Its drawbacks include corrosiveness with structural materials, and production of polonium as a by-product of irradiation. There is some concern about the time required for the additional fuel development and qualification that would be required compared to the SFR.

V.D.5. Conclusions

The SFR and LFR both met the essential criteria established for the FITR project, and were also the two highest ranking concept options. Further development and evaluation of these two concept options is recommended. The GFR is far behind in technology development and will not be considered further as a viable candidate for the FITR. The PTR concept also did not meet the essential criterion related to near-term availability. Both PTR options finished below the SFR and LFR concept options in the overall scoring. However, the PTR-L concept was fairly close to the LFR in the rankings and it is recommended that some additional development and evaluation effort be put into the PTR given the potential benefits of the concept.

VI. Conclusions and Path Forward

The main objective of this project was the evaluation of various options for a Fast Irradiation Test Reactor (FITR) to address the specific and compelling future needs for fast neutron irradiation testing, and to choose a limited number of options for further development and evaluation. Argonne National Laboratory (ANL) and the Idaho National Laboratory (INL) teamed together to analyze and evaluate options, and develop a set of technical and programmatic criteria that would be used to select these options. The team worked together, across DOE contractor organizations, in an integrated, collaborative manner. This project is succeeding along these lines and will continue along this same direction in the future.

This report documents the team's evaluations of various options for a FITR and compared them against the functional performance requirements and missions for the test reactor.

The two options that were selected from the down-selection process are the sodium-cooled fast reactor (SFR) option and the lead-bismuth-eutectic-cooled fast reactor (LFR). These two options met the essential criteria and scored the highest in the ratings based on importance criteria. The novel pressure tube reactor (PTR) concept did not meet the essential criteria based on the expected research and development required, and it's rating below the two selected options. However, it scored high enough (just below the LFR) to justify some additional evaluation effort given the potential benefits of the concept.

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