Appendix 4.0
Lead-Cooled Fast Reactor

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Acronyms

ADS	accelerator-driven system
ANL	Argonne National Laboratory
BOC	beginning of cycle
CO_2	Carbon Dioxide
CRIEPI	Central Research Institute of Electric Power Industry
DELTA	Development of Lead-Bismuth Target Applications
DOE	Department of Energy
DRACS	Direct Reactor Auxiliary Cooling System
EDX	energy-dispersive x-ray
ELSY	European Lead-cooled System
EOC	end of cycle
ETD/EFIT	European Transmutation Demonstrator/European Facility for Industrial Transmutation
EURATOM FP6	European Atomic Energy Community 6th Framework Programme
F/M	ferritic-martensitic
FY	fiscal year
GIF	Generation IV International Forum
HM	heavy metal
HTR	high-temperature recuperator
HX	heat exchanger
I-NERI	International Nuclear Energy Research Initiative
IP EUROTRANS	European Integrated Project on transmutation of high-level nuclear waste in an accelerator driven system
IPPE	Institute of Physics and Power Engineering
IRACS	Intermediate Reactor Auxiliary Cooling System
ISI	in-service inspection
ISTC	International Science & Technology Center
JAERI	Japan Atomic Energy Research Institute
JNC	Japan Nuclear Cycle Development Institute
KAERI	Korean Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LBE	lead-bismuth eutectic
LETF	Lead Engineering Test Facility
LFR	Lead-Cooled Fast Reactor
LTR	low temperature recuperator
LWR	light water reactor
MW_e	megawatt electric
MW_t	megawatt thermal
ODS	oxide dispersion strengthening
PCHE	printed circuit heat exchanger
PEACER	Proliferation-resistant, Environment-friendly, Accident-tolerant, Continuable-energy, Economical Reactor
RVACS	Reactor Vessel Auxiliary Cooling System
S-CO_2	supercritical carbon dioxide
SEM	scanning electron microscope
SSTAR	Small Secure Transportable Autonomous Reactor
TRU	transuranic

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A4.1 INTRODUCTION AND BACKGROUND

The LFR has the potential to meet many of the Generation IV mission interests. The LFR is mainly envisioned for electricity and hydrogen production, and actinide management. Options for the LFR also include a range of plant ratings and sizes from small modular systems to monolithic plants.

Two key technical aspects of the envisioned Lead-Cooled Fast Reactor (LFR) that offer the prospect for achieving the Generation IV goals of non-proliferation, sustainability, safety and reliability, and economics are (1) the use of lead (Pb) coolant and (2) a long-life, cartridge-core architecture in a small, modular system intended for deployment with small grids or remote locations. Figure A4.1 shows the conceptual LFR system. The Pb coolant is a poor absorber of fast neutrons and enables the traditional sustainability and fuel cycle benefits of a liquid metal-cooled fast spectrum core to be realized. Pb does not interact vigorously with air, water/steam, or carbon dioxide, eliminating concerns about exothermic reactions. It has a high boiling temperature (1,740°C), so the prospect of boiling or flashing of the ambient pressure coolant is realistically eliminated. Two land prototypes and ten submarine reactors utilizing lead-bismuth eutectic (LBE) coolant were operated in Russia providing about eighty reactor years of experience together with the supporting development of coolant technology and control of structural material corrosion.

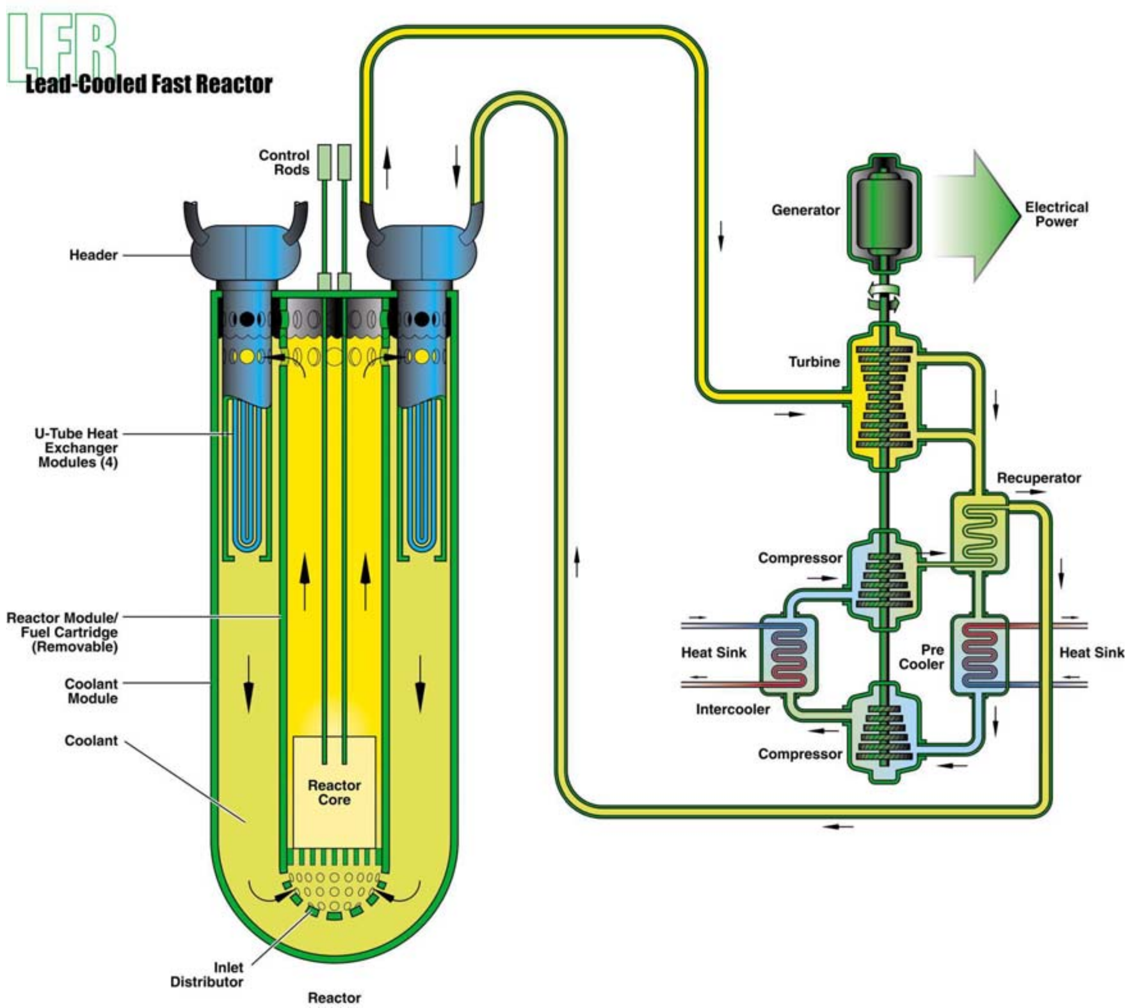


Figure A4.1. Conceptual LFR system.

The LFR envisioned in the Generation IV Program is the Small Secure Transportable Autonomous Reactor (SSTAR) concept, which is a small modular fast reactor. The main mission of the 20 MW_e (45 MW_t) SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities without electrical grid connections, such as those that exist in Alaska or Hawaii, Ulung Island in the Republic of Korea, island nations of the Pacific Basin (e.g., Indonesia), and elsewhere. This gives early LFR designs a unique niche market within which costs for competitive systems are typically higher and large-scale nuclear power plants are not competitive, while later evolution of the LFR technology to larger sizes may broaden the market. Design features of the reference SSTAR include a 20- to 30-year-lifetime sealed core; a natural circulation primary, autonomous load following without control rod motion; and use of an innovative supercritical carbon dioxide (S-CO_2) energy conversion system. The incorporation of inherent thermo-structural feedbacks imparts walk-away passive safety, while the use of a sealed cartridge core with a 20-year or longer cycle time between refueling imparts strong proliferation resistance. If these technical innovations can be realized, the LFR will provide a unique and attractive nuclear energy system that meets Generation IV goals.

The SSTAR-type LFR also provides a unique capability for management of actinides. The reactor can be fueled with the plutonium and minor actinide mixture that results from processing Light Water Reactor (LWR) fuel (plus some of the separated uranium). With a conversion ratio very near unity to enable the long core life, it serves as neither a breeder nor a burner. It uses the actinides for 20 to 30 years and then returns it upon recycle of the core by the supplier. The fissile material can then be either reformed into more SSTAR fuel for another 20 to 30 years of working storage or be made available for other reactors, either thermal or fast spectrum. Early deployment of such reactors could absorb all the fissile material from LWR recycle and return it in the future when fissile material supply could be a limiting factor in nuclear energy growth rate.

A4.1.1 System Description

SSTAR utilizes transuranic (TRU) nitride fuel enriched to nearly 100% in ^{15}N in a compact core. Heat is removed from the core and transported to in-vessel Pb-to-CO_2 heat exchangers by single-phase natural circulation of the Pb coolant; the need for main coolant pumps is eliminated. The fast spectrum core with nitride fuel and Pb coolant has strong reactivity feedbacks that enable autonomous load-following and provide passive power shutdown in the event of loss-of-normal heat removal. The core has a long lifetime/refueling interval of 20 years during which access to the core is restricted, providing proliferation resistance; the TRU fuel is self-protective in the safeguards sense. The Pb coolant and nitride fuel provide for enhanced passive safety whereby the core and in-reactor heat exchangers remain covered by ambient pressure, single-phase, primary coolant inside the reactor vessel, and single-phase natural circulation removes the core power under all operational and postulated accident conditions. The reference SSTAR reactor system is coupled to an S-CO_2 gas turbine Brayton cycle power converter that enables potential improvements and cost savings over the traditional Rankine saturated steam cycle including higher cycle efficiency at temperatures attainable with Pb primary coolant and nitride fuel (650°C peak cladding temperature and 561°C core outlet temperature for a 405°C inlet temperature) as well as remarkably small compressors and turbine, and a smaller plant footprint with simpler secondary side components.

The SSTAR reference reactor system fits inside of a reactor vessel that is about 18 m tall and 3.2 m in diameter—small enough to be transported either by rail or barge. The compact ~1.0 m diameter/0.8 m height active core is located near the bottom of the vessel. Large diameter (2.7 cm) fuel pins are arranged on a triangular pitch. The core is not composed of individual removable assemblies but is a single proliferation resistant cassette that can be accessed only when refueling equipment is brought to the site at the end of the core lifetime. The fuel pins consist of TRU nitride (enriched to nearly 100% ^{15}N) pellets bonded by molten Pb to the silicon-enhanced ferritic-martensitic (F/M) stainless steel cladding. A tall fission gas plenum (1.75 times the active core height) is provided at the top of each fuel pin. The molten Pb coolant flows upward through the core and the overlying riser region inside of a cylindrical shroud. Near the free surface at the top of the Pb, the coolant enters modular Pb-to-CO_2 heat exchangers located in the annulus between the shroud and reactor vessel to flow downward over the exterior of double-walled tubes containing the upward flowing carbon dioxide (CO_2). The Pb continues through the downcomer region beneath the heat exchangers and enters the lower plenum below the core where a flow distributor tends to equalize the pressure at the core inlet.

The Pb flow is driven solely by natural circulation. The low core pressure drop reflecting a large coolant hydraulic diameter and short fuel pin height is key. The Pb coolant enters the core at 405°C (providing adequate margin above the Pb freezing temperature of 327°C) and exits the core at a 561°C mixed mean outlet temperature. The maximum temperature at the cladding inner surface is 650°C. Corrosion control is maintained through the formation and maintenance of protective oxide (Fe_3O_4 at lower temperatures) layers upon the steel structural surfaces through maintenance of the dissolved oxygen potential in the Pb coolant. Shutdown rods provide for startup and shutdown while compensation rods offset small reactivity changes during the 20-year core lifetime. Control rods are not needed to effect power changes during autonomous load following due to the strong reactivity feedbacks of the fast spectrum core. The reactor vessel is surrounded by a guard vessel. The exterior of the guard vessel is passively cooled by upward flowing air driven by natural convection; passive air cooling provides for emergency heat removal in the event that neither the normal operational nor shutdown heat removal paths are available. The reactor system is coupled to an S-CO_2 power converter. S-CO_2 at 20 MPa pressure is heated to 541°C in the in-reactor Pb-to-CO_2 heat exchangers. It expands to about 7.4 MPa in a remarkably small turbine that drives the generator and then passes through two recuperators (a high temperature recuperator followed by a low temperature recuperator) where a portion of the remaining thermal energy is extracted to preheat the compressed CO_2 that is returned to the in-reactor heat exchangers.

Upon exiting the low temperature recuperator, about 67% of the CO_2 passes through the cooler where heat is rejected from the cycle and the CO_2 is cooled to 31.25°C, compressed in a small compressor to 20 MPa, and preheated in the low temperature recuperator. The remaining 33% of the CO_2 is directly recompressed in a second compact compressor and merged with the other flow stream between the low and high temperature recuperators. This flow split/merge approach is necessitated by the significantly greater specific heat of the higher pressure CO_2 over the temperature range of the low temperature recuperator. The recuperators and cooler incorporate printed circuit heat exchangers (PCHEs) to further reduce component volumes. The cycle efficiency of 44% provides about 20 MW_e of electricity for 45 MW_t of core thermal power.

A concept for and early technology demonstration reactor is currently being developed to provide a shorter, and perhaps less costly, path forward for SSTAR development. The early demonstrator would back off from certain SSTAR design features to permit near-term construction with low technology risk, while demonstrating key SSTAR features and providing a test bed for other SSTAR features. Reduction in operating temperature and core lifetime would enable construction with existing materials. while allowing for advanced material testing. Use of oxide or metal driver fuel might avoid long fuel qualification for nitride fuel, while the facility could be used for nitride fuel testing to qualify future core fuel. Modest increase in neutron flux would reduce test time for fuels and materials.

A4.1.2 System Timeline

The schedule proposed for LFR development is illustrated in Figure A4.2. The plan described in this section reflects 10 years of a 20-year development program leading to startup of a LFR demonstration unit. Key dates in the current Ten-Year Program Plan include a fast reactor option selection in 2010 and a decision in 2014 whether to proceed to construction of the LFR demonstration plant. The option of an early demonstration reactor is not reflected in this baseline schedule.

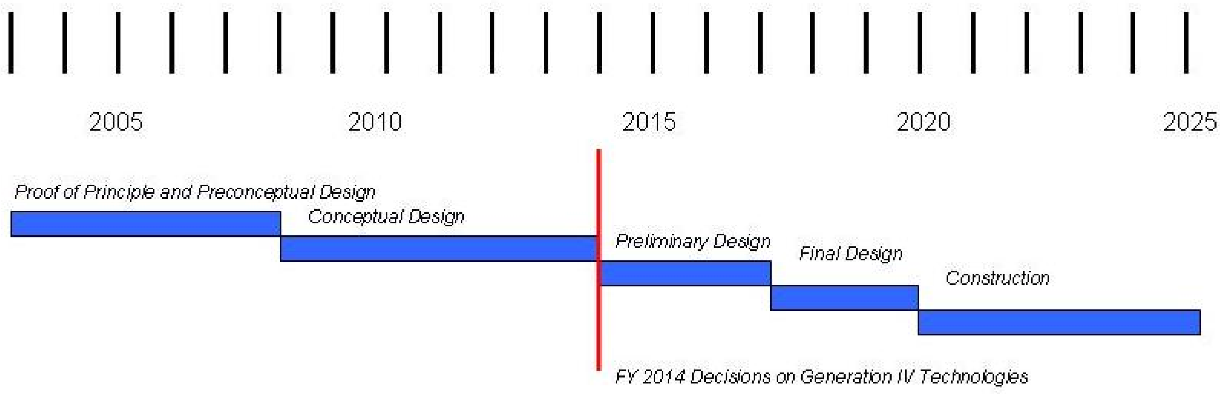


Figure A4.2. Proposed schedule for LFR/SSTAR development.

A4.2 RESEARCH AND DEVELOPMENT STRATEGY

The LFR development strategy incorporates a near-term focus on the technologies for a small, simple modular design for specialized markets. Targeting this market need offers an additional benefit of manageable research and development (R&D) and demonstration costs. If market conditions motivate it, the technology base can be applied later to larger LFR concepts as needed to support a long-term sustainable fuel cycle. Current and near-term R&D is planned to address key viability issues leading to the 2010 decision, while subsequent R&D will address issues leading to demonstration of the LFR concept.

A4.2.1 Objectives

LFR viability R&D objectives can be grouped according to two purposes:

1. Objectives demonstrating the viability of LFR/SSTAR features to meet all of the Generation IV goals
2. Objectives demonstrating the viability of satisfying requirements for a commercial LFR/SSTAR nuclear power plant that meets all of the Generation IV goals.

Objectives directly related to meeting each of the Generation IV goals are:

- *Sustainability-1 (S-1) and Sustainability-2 (S-2):* Analyses indicate that the LFR with a fast neutron spectrum core with TRU nitride fuel and Pb coolant is fissile self-sufficient with a core conversion ratio of unity. This enables a closed fuel cycle in which there is a fertile feed stream of depleted or natural uranium and a minimal volume waste stream comprised only of fission products. All fissile material including minor actinides can be recycled in the fabrication of new fuel cores and burned as fuel in reactors. Objectives are to confirm the viability of these attributes.

- *Economics-1 (E-1):* Overnight and generation costs remain to be estimated. Objectives are to demonstrate the viability of reducing costs by taking advantage of LFR system attributes that enable savings such as system simplification through elimination of the need for an intermediate heat transport circuit; elimination of main coolant pumps; autonomous load following that simplifies the control system and reduces operator requirements; utilization of S-CO_2 Brayton cycle power conversion that offers higher plant efficiency together with smaller, simpler, and fewer balance-of-plant components; and a small plant footprint, factory fabrication that reduces component costs, and modular transport and installation at the site that reduces construction time and costs.

- *Economics-2 (E-2):* Financial risk remains to be quantified. The small modular plant requires a smaller outlay of funds and provides a shorter construction time. When the plant goes online it becomes a source of positive cash flow that can be applied to financing the construction of the next module and so on. An objective is to establish the viability of this approach. Another objective of passive safety is to demonstrate the viability of minimizing the threat to investment in the plant due to postulated accidents or sabotage.

- *Safety and Reliability-1 (SR-1):* Objectives are to show the viability of taking advantage of the highly favorable LFR attributes, including properties of Pb coolant (inertness with respect to interactions with CO_2, water/steam, and air, and high boiling temperature), properties of nitride fuel (compatibility with cladding, bond, and coolant; high melting temperature; high decomposition temperature; and large thermal conductivity), natural circulation heat transport at power levels exceeding 100% nominal, and large reactivity feedbacks from the fast spectrum core that provide passive power reduction to achieve ‘walk-away’ passive safety.

- *Safety and Reliability-2 (SR-2):* The SSTAR core and heat exchangers will remain covered by ambient pressure, single-phase, primary coolant inside the reactor vessel, and single-phase natural circulation will remove the core power under all operational and postulated accident conditions. Objectives are to show that it is viable to assert that this is the nominal behavior and to show that it is viable to ensure that scenarios that could conceivably result in core damage, such as the simultaneous penetration of both the reactor vessel and guard vessel, have an extraordinarily low probability.

- *Safety and Reliability-3 (SR-3):* Objectives are a viable licensing approach that effectively uses “Walk Away” passive safety and a very low likelihood of core damage to possibly eliminate the licensing need for offsite emergency response. A specific objective is the acceptance of passive safety as a barrier in the defense-in-depth approach that is a bedrock principle of licensing to argue for elimination of the need for emergency planning.

- *Proliferation Resistance and Physical Protection-1 (PR-1):* The TRU fuel with incomplete fission product removal is self-protective in the safeguards sense. Objectives are to establish the viability of achieving a very long core lifetime (which has been shown to be neutronically viable) together with the viability of design measures to restrict access to the fuel or neutrons, to effectively refuel the core and transport it in a coolable and shielded state while maintaining a high resistance to theft, and to provide a plant that is resistant to sabotage or malevolent human-induced acts (e.g., airplane crashes).

Table A4.1 summarizes how specific features of the LFR/SSTAR system can contribute to satisfaction of all of the Generation IV goals. Table A4.2 summarizes how specific requirements for the LFR/SSTAR system can contribute to satisfaction of all of the Generation IV goals.

Table A4.1. Contribution of specific LFR/SSTAR features to meeting of Generation IV goals.

LFR Features	S-1	S-2	E-1	E-2	SR-1	SR-2	SR-3	PR-1
Pb Properties	+	+	+	+	+	+	+	
Nitride Fuel Behavior	+	+			+	+	+	
Fast Spectrum Core—Long Core Lifetime			+	+				+
Fast Spectrum Core—Unity Conversion Ratio	+	+	+	+				+
Fast Spectrum Core—Burn-up Reactivity Swing <$1			+	+	+	+	+	
Fast Spectrum Core—Large Reactivity Feedbacks			+	+	+	+	+	
Simplification—Elimination of Intermediate Heat Transport System			+	+	+	+	+	
Simplification—Natural Circulation Primary Coolant Heat Transport			+	+	+	+	+	
Passive Safety			+	+	+	+	+	
TRU Fuel—Self Protection	+							+
S-CO_2 Brayton Cycle Power Conversion			+	+				
Small Power Level—20 MW_e (45 MW_t) to 100 MW_e			+	+				
Factory Fabricability			+	+				
Full Transportability and Modular Assembly and Installation at Site			+	+				

Table A4.2. Contribution of specific LFR/SSTAR requirements to meeting Generation IV goals.

LFR Features	S-1	S-2	E-1	E-2	SR-1	SR-2	SR-3	PR-1
Proliferation Resistance								+
Small Power Level - 20 to 100 MW_e			+	+				
Economical Overnight Capital and Generation Costs; Niche Market Conditions			+	+				
‘Walk Away’ Passive Safety			+	+	+	+	+	
Autonomous Operation			+	+	+	+	+	
Fissile Self-Sufficiency	+	+	+	+				+
Reactor Compatible with Advanced Energy Conversion			+	+				
Small Power Level—20 MW_e (45 MW_t)			+	+				
Factory Fabrication of All Reactor and Balance-of-Plant Components			+	+				
Full Transportability and Modular Assembly and Installation at Site			+	+				
Efficient Fuel Utilization	+	+	+	+				
60-Year Plant Lifetime			+	+				
Plant Licensable by U.S. Nuclear Regulatory Commission					+	+	+	
Near Worldwide Deployable	+	+	+	+	+	+	+	
In-Service Inspection			+	+	+	+	+	
Flexibility to Generate Other Energy Products—Desalinated Water			+	+				

A4.2.2 Scope

This R&D plan will address viability issues associated with the LFR leading to the Generation IV fast reactor selection in 2010 and a follow-on decision in 2014 to proceed with design and construction of the LFR demonstration plant. The plan reflects 10 years of a 20-year development program leading to startup of a LFR demonstration unit. Viability will be established through focused viability R&D tasks and with formulation of a technically defensible pre-conceptual design. Conceptual design will begin in 2009 and continue, given a decision for pursuing the LFR in the 2010 to 2014 timeframe. R&D tasks that support conceptual design will be defined in more detail at a later time in the viability R&D program, but will include analysis and experiments intended to reduce design uncertainty and to establish conceptual limiting conditions of operation.

A4.2.3 Viability Issues

Viability issues for the LFR as listed topically as follows, and are described further in Section A4.3.

- Core Neutronics
  - Fuel conversion sufficient to sustain a 20- to 30-year-life core
  - Identification of core parameters that provide feedback coefficients, and ensure passive safety and autonomous load following viability
- System Thermal-Hydraulics
  - Natural circulation within core parameter constraints necessary to meet conversion and thermo-structural feedback requirements
  - Thermal response to feedback to ensure passive safety and autonomous load following
  - Feasibility of elimination of an intermediate loop
  - Identification of Pb-S-CO_2 heat-exchanger parameters
  - Safety issues arising from S-CO_2 tube rupture and identification of mitigation concepts
  - Concepts for passive decay heat removal.
- Structural Design
  - Stress and temperature conditions for structural materials are compatible with expected materials performance.
- Materials
  - Materials structural reliability in the anticipated service environment (i.e., high-temperature Pb in a fast neutron flux)
  - Materials compatibility with and corrosion behavior on high-temperature Pb
  - Conditions of operation required to ensure materials integrity and compatibility are practically achievable in an operating plant.
- Nitride Fuel
  - Uranium nature and TRU nitride compatibility with Pb at elevated temperatures
  - High-burnup potential of nitride fuel during an extended core life is not yet established
  - Transient behavior must be shown to not exacerbate off-normal events.

- Passive Safety Characteristics
  - Thermo-structural feedback coefficients required to ensure passively safe response must be practically achievable
  - Decay heat removal after accidents must be sufficient to prevent core damage
  - Pb-S-CO_2 heat exchanger tube rupture must not prevent heat removal from the core or introduce positive reactivity insertion.
- Containment and Building Structures
  - Containment necessary to prevent release to environment must be small enough to reduce economy of scale
  - Containment necessary to prevent release to environment must allow a path for passive removal of decay heat.
- In-service Inspection
  - Reactor vessel and safety system integrity in the vessel sealed for 20 to 30 years must be verified using a viable and practical means.
- Cost Impacts
  - Design features to achieve necessary safety and proliferation resistance must not impact capital costs to render the LFR uneconomical for the envisioned deployments
  - Operating strategy must be compatible with requirements to ensure operating costs are acceptable for the envisioned deployments
  - Cost-essential design features must be identified to ensure compatibility with design for performance.
- Whole-core Cassette Refueling
  - Concepts for safe and secure refueling must be identified and must be practical
  - Shielding and cooling of a spent core must be practically achievable
  - Design features to allow refueling must not add significantly to capital cost and must not compromise safety of containment.
- S-CO_2 Brayton Cycle Energy Conversion
  - The energy conversion system must be developed and demonstrated
  - The smaller, innovative conversion plant components must be developed and demonstrated.

A4.2.4 Research Interfaces

The LFR R&D program interfaces with a number of domestic and international partners, as described below. In addition, the current LFR concept calls for use of an S-CO_2 energy conversion system, and this plan assumes that the U.S. Generation IV Energy Conversion R&D will address development needs of that technology. However, the LFR program will identify and address aspects of S-CO_2 energy conversion that are specific to the LFR.

A4.2.4.1 Relationship to Generation IV International Forum Research and Development Projects and Other International Cooperation

A Generation IV International Forum (GIF) LFR Steering Committee has been formed with members from the European Atomic Energy Community (EURATOM), Japan, Republic of Korea, and the U.S. A working draft of an LFR R&D Plan has been developed that describes the R&D interests of each participant and identifies opportunities for collaborative studies. The R&D Plan includes tracks toward both small and medium-large-sized LFRs, with potential for a joint demonstration reactor of intermediate size.

Lead coolant R&D activities in Europe are well established in the framework of accelerator-driven system (ADS) development. A group of 29 organizations plus many universities has presented to the European Community the IP EUROTRANS project to develop a preliminary design for a European Transmutation Demonstrator/European Facility for Industrial Transmutation (ETD/EFIT). In addition, there is renewed interest in reactor R&D, with twelve organizations joining the new European Lead-Cooled System (ELSY) program in the European Atomic Energy Community 6th Framework Programme (EURATOM FP6) to study a medium-size (600 MW_e) lead-cooled, forced convection fast reactor. U.S. organizations have been invited to participate in ELSY.

Work with Japan includes cooperation on topics common to small modular fast reactors, with focus on lead coolant in the U.S. and sodium coolant in Japan. Periodic coordination meetings are held with the Central Research Institute of Electric Power Industry (CRIEPI) and Toshiba. In addition, technical collaborations and information exchanges exist with the Japan Atomic Energy Research Institute (JAERI), Japan Nuclear Cycle Development Institute (JNC), and Tokyo Institute of Technology.

Although Russia is not currently part of the GIF, there has been a long history of cooperation with Russian Pb-Bi and Pb reactor technology experts, in particular the Institute of Physics and Power Engineering (IPPE). Several on-going International Science and Technology Center (ISTC) projects support further development of coolant technology and materials.

A4.2.4.2 University Collaborations

The University of California-Berkeley has been a long-standing member of the LFR R&D community with contributions in innovative core and component design and analysis.

A Ph.D. candidate at Ohio State University is working on a two-year Department of Energy (DOE) Nuclear Engineering Graduate Fellowship in collaboration with Argonne National Laboratory (ANL) on the development of an approach to cool the LFR spent cassette core during refueling and transport.

Partnership in material science topics includes work with Massachusetts Institute of Technology, University of Nevada-Las Vegas, University of Illinois, and University of Wisconsin.

A4.2.4.3 Industry Interactions

In addition to work with Toshiba via CRIEPI, discussions on potential cooperation have been held with General Electric and Westinghouse. Further industry participation is expected when funding permits.

A4.2.4.4 International Nuclear Energy Research Initiatives/Nuclear Energy Research Initiatives

A U.S. DOE/Euratom Joint International Nuclear Energy Research Initiative (I-NERI), “Lead Fast Reactor Engineering and Analysis,” is in progress between ANL and the Joint Research Center of the European Commission, Institute for Energy in Petten, the Netherlands.

There is an on-going I-NERI between Los Alamos National Laboratory (LANL) and the Korean Atomic Energy Research Institute (KAERI) and Seoul National University in the Republic of Korea on fuel cladding materials development and testing as well as improved oxygen sensors. The Republic of Korea program, PEACER, is developing a LBE-cooled transmutation reactor.

There is an ongoing I-NERI between ANL and the KAERI on "Supercritical Carbon Dioxide Brayton Cycle Energy Conversion." Seoul National University is also a contributor to that collaboration through a subcontract from KAERI.

A proposed cooperation with the Japan Atomic Energy Agency would examine oxide dispersion strengthening (ODS) steel irradiation performance and compatibility with Pb-alloys. Another proposed cooperation with CRIEPI would involve participation in retrieval and examination of archived Fast Flux Test Facility material irradiation specimens.

A4.3 HIGHLIGHTS OF RESEARCH AND DEVELOPMENT

A4.3.1 Core Neutronics

Motivation:

Core design is essential to establishing the necessary features of a 20- to 30-year-life core, and determining core parameters that impact feedback coefficients, which are essential inputs for establishing passive safety and autonomous load-following viability.

Tasks:

- Further optimize the core configuration
  - Determine the size of the central low enrichment zone to reduce radial power peaking and improve time-dependent conversion behavior for the long-life core strategy.
- Establish startup/shutdown rod and control rod strategy
  - Assess number and location of rods
  - Satisfy diversity and redundancy requirements.
- Calculate reactivity feedback coefficients
  - Support autonomous load following evaluation
  - Support passive safety evaluation.

A4.3.2 System Thermal-Hydraulics

Motivation:

Studies of system thermal-hydraulics are essential to establish the parameters for potential natural circulation cooling in the primary system, identifying any safety issues to be addressed in subsequent design, and establishing parameters for ensuring passively-safe response.

Tasks:

- Conduct autonomous load following evaluation for reactor using the calculated reactivity feedback coefficients
  - Determine need for any enhancement of core radial expansion feedback

- Develop and evaluate preconceptual control strategy for S-CO_2 Brayton cycle to match the heat removal from the reactor to the load demand from the electrical grid over power levels from near zero to above nominal.

- Determine viability of elimination of intermediate heat transport system
  - Evaluate effect of tube rupture in Pb-to-CO_2 heat exchanger (HX) and blow-down of CO_2 into reactor system
  - Develop pressure relief strategy for reactor coolant system
  - Evaluate need to contain CO_2 and entrained radionuclides released from reactor coolant system
  - Assess impact upon containment configuration, size, capability, and other requirements
  - Determine viability of failure-resistant HX concepts.
- Ensure viability of startup using natural circulation
  - Evaluate possible need for small flow assist during startup or shutdown
  - Assess options for startup flow: mechanical, electromagnetic, lift (non-condensable gas injection), or jet pumps
  - Assess options for melting the Pb and heating the primary coolant system to maintain the Pb in a molten state prior to and during initial neutronic startup.
- Establish viability of emergency heat removal concept
  - Evaluate safety grade system
  - Assess performance of passive air cooling of outside of guard vessel
  - Compare relative merits of alternate in-reactor cooling systems (e.g., Direct Reactor Auxiliary Cooling System [DRACS] or Intermediate Reactor Auxiliary Cooling System [IRACS]) versus Reactor Vessel Auxiliary Cooling System (RVACS) approach – performance, reliability, cost, resistance to attack or sabotage
  - Determine final selection of emergency heat removal approach.

A4.3.3 Structural Design

Motivation:

Viability of the long-life core and passive safety under all abnormal conditions (including seismic events that might unacceptably reconfigure a core) requires materials that can withstand stresses at high temperatures and, for some components, contact with liquid lead. The range of expected stresses and temperatures, and the potential materials must be identified. Establishing actual materials and conditions of operation are design functions to be accomplished later in a development program. However, ranges of conditions must be identified to provide requirements for materials and to determine that such material performance can be achieved within an engineering development program.

Tasks:

- Evaluate pre-conceptual structural design to ensure viability at projected system temperatures up to 650°C peak cladding
  - Identify suitable structural materials for core, in-vessel structures, reactor, and guard vessels using the materials at projected system temperatures

- Evaluate concepts for core support, core clamping, and restraint
- Evaluate effect of seismic requirements on structures including reactor and guard vessel thicknesses
- Evaluate effects of accidental freezing and thawing events upon structures.

A4.3.4 Materials

Motivation:

Prior experience with heavy liquid metals and with fast reactors indicates that construction materials will be challenged in the envisioned LFR environments. Viability of long core lifetime, passive safety, and economic performance (both capital and operating costs) will depend upon identifying materials with the potential to meet service requirements.

Tasks:

- Identify candidate silicon-enhanced F/M steels, F/M steels, ODS F/M steels, carbides, amorphous materials, and other candidate materials
- Conduct compatibility testing of candidate materials with heavy liquid-metal coolants
- Demonstrate control of corrosion to ensure adequate thickness of cladding and structural elements at operating temperatures over long core and reactor lifetimes
- Prepare code cases for selected cladding and structural materials throughout the operating temperature range.

A4.3.5 Nitride Fuel

Motivation:

Achieving long core life, walk-away passive safety, and reliable operation will require robust and predictable fuel performance for long durations under service conditions. Nitride fuel has many properties and characteristics that render it well suited for LFR application; however, there is very little data on nitride fuel performance to confirm the designer's current assumptions regarding this fuel type for TRU nitride fuel or for transient fuel performance. In addition, although operation with failed fuel must be a low-probability circumstance, it must be accommodated, so sufficient compatibility of irradiated nitride fuel with lead at high temperatures must be demonstrated.

Tasks:

- Perform irradiation testing and demonstration to projected burnup (>13 atomic weight %) under operating conditions
    - Include TRU nitride with volatile minor actinide constituents.
- Perform transient testing including accident conditions to verify acceptable fuel behavior.

A4.3.6 Passive Safety Evaluation

Motivation:

Passively-safe response can be designed into the reactor core and plant based on current experience and passive safety design principles. However, the magnitudes of feedback coefficients for a given design and integral behavior of a reactor plant must be verified through further analysis. It is anticipated that some coefficients may require enhancement through design modification, and those design impacts must be determined acceptable at the preconceptual level through follow-on analysis. Eventually, inherent response of components (i.e., the magnitude of the coefficients for certain design configuration) must be verified with single-effect experiments and through integral testing and demonstration with a reactor plant. These experimental tasks, however, are not necessary for the viability phase.

Tasks:

- Evaluate operational transients and postulated accidents
  - Apply coupled LFR thermal-hydraulics, neutron-kinetics S-CO_2 Brayton cycle energy converter plant dynamics analysis code
  - Model both reactor system and S-CO_2 Brayton cycle.
- Evaluate potential for flow instability
- Evaluate potential for flow reversal
- Use calculations to demonstrate that core and Pb-to-CO_2 heat exchangers remain covered by ambient pressure, single-phase, primary coolant inside the reactor vessel and single-phase natural circulation removes the core power under all operational and postulated accident conditions with the exception of postulated beyond design basis accidents having an extraordinarily low probability
- Evaluate removal of afterheat during postulated accidents

A4.3.7 Containment and Building Structures

Motivation:

Use of a small, closely-coupled containment is essential for reducing the per-MW capital cost of the LFR. Experience with LWRs and previous fast reactor plants and concepts indicates that large containments necessary to contain a fair amount of gaseous reaction and fission products drove such plants to large economies of scale. This must be avoided if the LFR is to be financially viable. Therefore, the factors that would drive containment design must be evaluated as part of a viability R&D program to ensure that the design, if technically achievable, can avoid large-size containment requirements.

Tasks:

- Evaluate requirements for containment
  - Determine ranges of radionuclide contents generated in coolant or released to coolant from postulated failed cladding
  - Assess potential need to contain CO_2 and entrained radionuclides released from reactor coolant system.

- Evaluate containment configuration, size, and capability
  - Determine external events for consideration
  - Evaluate need for and conceptual design of decay heat removal system for postulated accidents.
- Consider industrial health aspects of operation with Pb and CO_2
  - Pb vapor and aerosols following leaks/spills
  - CO_2 release
  - CO_2 toxicity.
- Identify decontamination and decommissioning issues that would impact design.

A4.3.8 In-Service Inspection

Motivation:

Twenty- to thirty-year operation of a plant with a sealed core will require a means of inspection and verification of key safety structures and boundaries. If such integrity cannot be verified, then the LFR concept is not likely to be licensed. Therefore, concepts for inspection and verification (in-service inspection) must be identified during the viability R&D phase for subsequent engineering development.

Tasks:

- Identify in-service inspection (ISI) approaches for operation over long core lifetimes of 20 years or more or propose and evaluate approaches (e.g., robust core support) that significantly reduce or minimize the requirements for ISI
- Assess capability to operate with failed cladding over long core lifetime.

A4.3.9 Assessing Cost Impacts

Motivation:

Because the envisioned LFR concept will not have the benefit of economy of scale, the identified opportunities to reduce capital and operating costs below those of larger, base-load plants must be evaluated. In particular, additional design features with strong cost impacts must be identified and considered for subsequent changes to design requirements.

Tasks:

- Determine basis for credible estimate of plant costs
  - Estimate plant capital and generation cost factors, with consideration of LFR-specific attributes (e.g., experience with factory construction of modules, etc.)
  - Account for benefits of design simplification, passive safety, factory fabrication, modular assembly, reduced construction time at site, and reduced staffing
  - Account for the significantly higher efficiency and other potential benefits of S-CO2 Brayton cycle power conversion.
- Evaluate economic conditions for niche market applications.

A4.3.10 Whole-Core Cassette Refueling

Motivation:

If the proliferation-resistant LFR system is to be viable as envisioned, with refueling occurring only at 20- to 30-year intervals and with equipment that is brought onsite temporarily rather than maintained onsite, credible concepts for emplacing and exchanging fueled core cartridges must be proposed and considered. Preconceptual designs for such systems and identification of the requirements those systems would place on the reactor primary system as well as the containment and buildings must be evaluated.

Tasks:

- Determine viability of cooling spent cassette during retrieval and shipment following short cooldown period
- Identify spent-fuel-cassette shielding concepts
- Evaluate in-cask cassette cooling concepts
- Evaluate safeguards considerations
- Determine impact on plant containment and building structures.

A4.3.11 Supercritical-Carbon Dioxide Brayton Cycle

Motivation:

Use of an S-CO_2 Brayton cycle for energy conversion offers the prospect of significantly higher efficiencies at the reference LFR core outlet temperature and acceptable efficiencies with lower Pb coolant outlet temperatures, which reduces the challenges for materials in a near-term demonstration. Furthermore, the economic viability of the LFR may depend on reduction of capital cost achieved by incorporation of an S-CO_2 Brayton cycle rather than a steam Rankine cycle. Therefore, several R&D tasks associated with S-CO_2 Brayton cycle conversion are identified as viability tasks. Some of these tasks are expected to be addressed as part of Generation IV Energy Conversion R&D, but LFR-specific issues involving impact on reactor operation and design and heat exchange with lead coolant will be considered as part of the LFR scope.

Tasks:

- Evaluate innovative design concepts for compressors, turbine, PCHEs (by Heatric, a subsidiary of Meggitt, Ltd.), and other components
- Design, test, and demonstrate compressors, turbine, printed circuit heat exchangers, and other components
- Demonstrate long-term operation of components with small channels (e.g., PCHEs) without fouling or corrosion
- Demonstrate operation of an integral S-CO_2 Brayton cycle at sufficiently large scale.

A4.4 PROJECT COST AND SCHEDULE

The time and resources to conduct the planned R&D to prepare the LFR technology and design for Generation IV down-selection, and provide a basis for a decision to proceed with a prototype construction is shown below with a known budget for fiscal year (FY) 2006, and estimated budgets in later years.

A4.4.1 Fiscal Year 2006 Project Budget

The FY 2006 budget to begin the LFR R&D described in the previous sections is provided in Table A4.3.

Table A4.3. FY 2006 budget profile for LFR activities ($K).

Task	FY-06[a]
System Design and Evaluation	575
Materials	775
Total	1,350

a. FY 2006 funding includes FY 2005 carryover funds.

A4.4.2 Ten-Year Project Schedule

The schedule proposed for LFR development is illustrated in Figure A4.3. If there is sufficient interest in an earlier demonstration than that identified in the current Generation IV schedule, then a critical decision-driven schedule for a demonstration project can be prepared. However, the plan described in this appendix reflects a ten-year development of the technical basis for a Generation IV down-selection and decision whether to proceed to construction of a LFR demonstration.

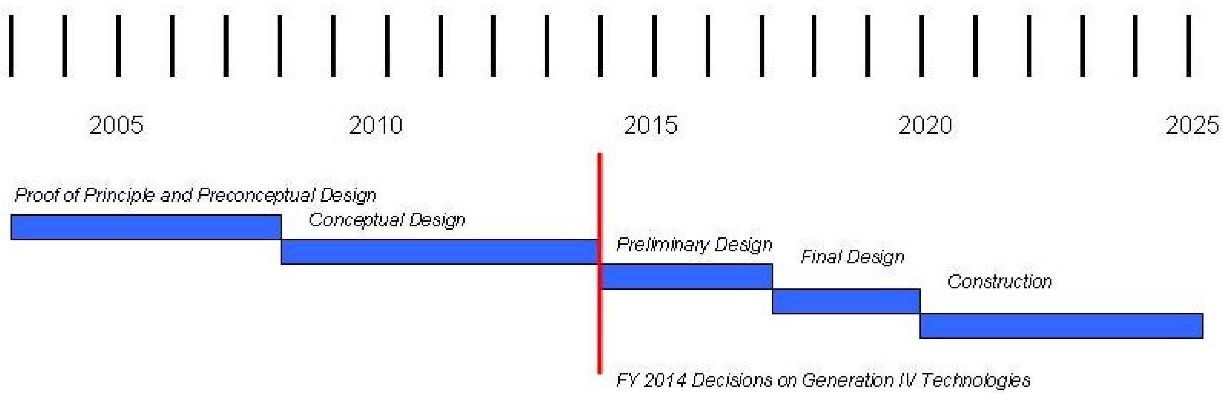


Figure A4.3. Proposed Schedule for LFR Development.

A4.4.3 Ten-Year Project Milestones

FY 2006

- Initiate studies of potential alloy modification, surface treatments, and amorphous metals for LFR environments
- Report Early Demonstration Reactor concept
- Complete design requirements for Lead Engineering Test Facility (LETF)
- Issue status report on preconceptual viability studies and evaluations, including reactivity control strategy, system heat transport and emergency heat removal, containment, and structural viability.

FY 2007

- Complete initial assessment of mechanical and corrosion properties of primary candidate LFR materials in as-received condition
- Complete preliminary selection of primary candidate materials for LFR system
- Issue status report on preconceptual viability studies and evaluations including structural viability assessment, containment approach viability, initiation of plant transient analyses and safety evaluation, and initiation of evaluation of core refueling and transport options.

FY 2008

- Establish reference cladding design and material specifications
- Complete facility design for LETF
- Issue status report on preconceptual viability studies and evaluations including plant transient analyses and safety evaluation, viability of core refueling and transport approach, and plan for utilization of LETF and other LFR experiment facilities.

FY 2009

- Complete initial aging and irradiation resistance assessment of candidate materials
- Begin LETF construction
- Issue status report on preconceptual viability studies and evaluations including plant transient analyses and evaluations, initial analyses of available experiment data to calibrate analysis methods, and reduction of effects of assumptions and uncertainties in analyses.

FY 2010

- Establish initial design database for short-term mechanical and corrosion properties of primary candidate LFR materials in as-received condition
- Issue initial report on preconceptual design
- Begin LETF testing.

FY 2011

- Issue initial LFR Materials Handbook
- Issue status report on experiments performed in LETF and other LFR facilities
- Issue final report on preconceptual viability studies and overall viability assessment.

FY 2012

- Report qualification testing and modeling status
- Issue status report on experiments performed in LETF and other LFR facilities
- Begin fuel testing
- Issue interim status report on conceptual design.

FY 2013

- Report qualification testing and modeling status
- Issue status report on experiments performed in LETF and other LFR facilities
- Issue interim report on fuel testing
- Issue interim status report on conceptual design.

FY 2014

- Issue interim status report on conceptual design
- Issue interim report on fuel testing
- Issue interim status report on licensing basis and approach.

FY 2015

- Issue final status report on conceptual design
- Issue report on LFR/SSTAR licensing basis and approach.

ADDENDUM A4-1: SMALL SECURE TRANSPORTABLE AUTONOMOUS REACTOR REFERENCE CORE

The LFR program has developed an initial reference design described below. This design is used for further analyses to evaluate alternatives. The reference design is a 20 MW_e , 20-year core life, cassette core design with natural circulation. The SSTAR cassette core has been developed to meet the following requirements and constraints:

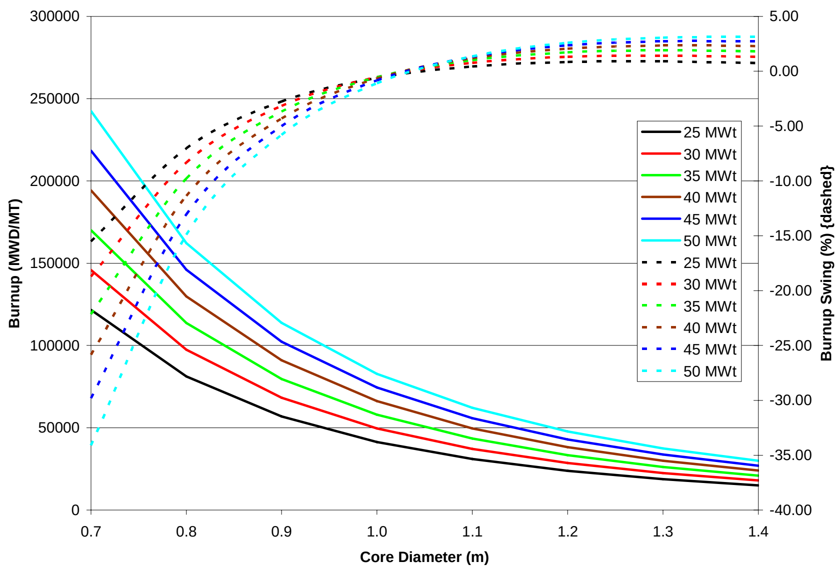
- Single batch fueling with TRU nitride fuel enriched to 100% in ^{15}N
- TRU fuel feed from LWR spent fuel following a 25-year cooldown time to reduce the effects of ^{241}Pu decay
- Core diameter small enough to meet the criterion for transportability by road as well as barge or rail
- Long fuel lifetime of 20 full power years
- Coolant volume fraction large enough to enable natural circulation heat transport of more than the full core power
- Minimization of burnup reactivity swing = $k_{eff,max} – k_{eff,min}$ during the cycle
- Maximization of average discharge burn-up
- Peak fluence less than or equal to 4×10^{23} fast neutrons/cm^2 for HT9 F/M cladding.

The use of ^{15}N eliminates parasitic (n, p) reactions in ^{14}N and waste disposal problems that would be associated with ^{14}C production. In order to reduce the core peak-to-average power ratio as well as the burn-up reactivity swing, five distinct TRU enrichment zones are employed including a central low enrichment zone.

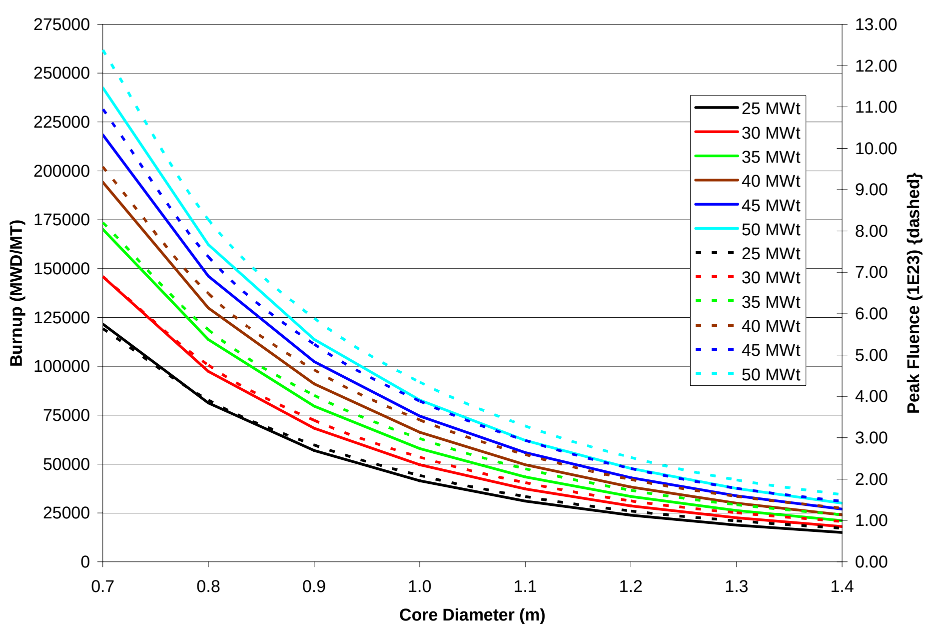
Addm A4-1: Figure 1 shows the results of calculations of the average discharge burnup and burnup reactivity swing versus active core diameter for a simplified cylindrical core geometry (height-to-diameter ratio = 0.8) assuming a fuel volume fraction of 0.55 and an 85% nitride fuel smeared density. It is observed that for this fuel volume fraction, the burnup reactivity swing exhibits a minimum at an active core diameter of about 1.0 m. Addm A4-1: Figure 2 plots the average discharge burnup as well as the peak fast fluence versus the active core diameter. Increasing the core thermal power directly increases the average discharge burnup. However, the maximum power is limited by the requirement that the peak fast fluence remain below the assumed limit of 4.0×10^{23} neutrons/cm^2. This limitation is encountered for core powers of about 45 to 50 MW_t. Thus, for the assumed 0.55 fuel volume fraction, a core diameter of about 1.0 m minimizes the burnup reactivity swing and a power level of about 45 MW_t maximizes the average discharge burn-up. More detailed calculations were performed using the DIF3D/REBUS-3 code package. Addm A4-1: Table 1 shows core conditions and the calculated core performance.

The reference fuel form consists of nitride pellets bonded by molten Pb to silicon-enhanced F/M stainless steel cladding. The fuel pins have a large diameter of 2.7 cm that provides a large hydraulic diameter for Pb coolant flow reducing the frictional pressure drop through the core as required for natural circulation. The fuel pins are arranged on a triangular pitch with a pitch-to-diameter ratio of 1.096. The core is a single cassette of fuel pins and is not composed of individual removable assemblies, providing a high degree of proliferation resistance. Nitride fuel has been selected for several reasons. First, it has a high melting temperature (e.g., 2,630°C for uranium nitride) and is compatible with the cladding as well as the Pb bond and coolant at high temperatures. It has a high atom density which makes feasible a compact fast spectrum core. A closed fuel cycle can be realized using electrometallurgical reprocessing.

The conversion ratio is near unity (for fissile self-sufficiency) over the 20-year lifetime. Nitride has a low volumetric swelling so, assuming a smeared density of 85%, the active core fuel volume fraction is equal to 0.55 at which the core power can be removed to in-reactor heat exchangers solely by single-phase natural circulation of the Pb coolant (i.e., main coolant pumps are eliminated).



Addm A4.1: Figure 1. Average discharge burnup and burnup reactivity swing versus active core diameter.



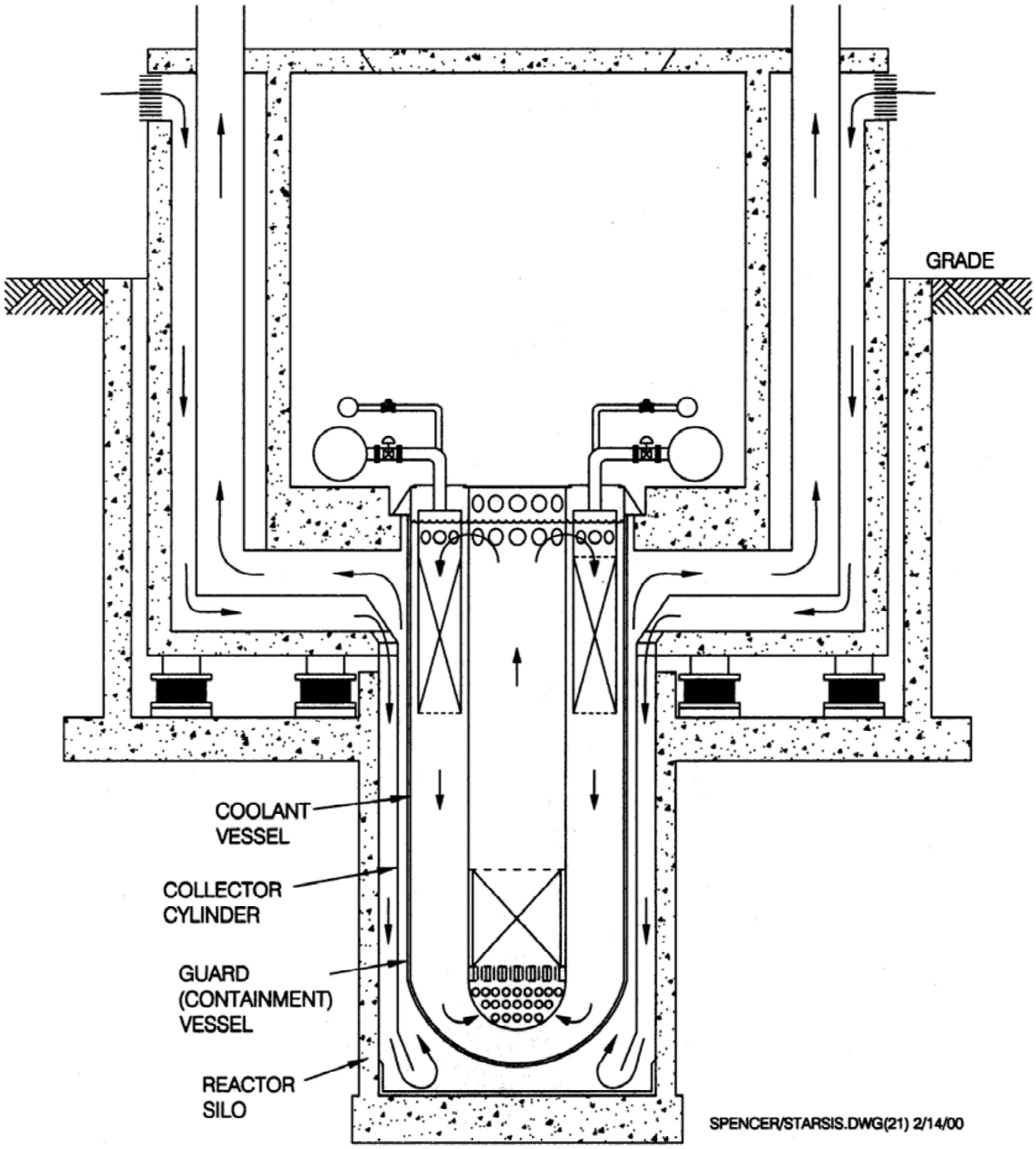
Addm A4.1: Figure 2. Average discharge burnup and peak fast fluence versus active core diameter.

Addm A4.1: Table 1. SSTAR core conditions and performance.

Core Conditions	Calculated Core Performance
Core Diameter, m	1.02
Active Core Height, m	0.8
Nitride Fuel Smeared Density, %	85
Fuel Volume Fraction	0.55
Cladding Volume Fraction	0.16
Bond Volume Fraction	0.10
Coolant Volume Fraction	0.16
Fuel Pin Diameter, cm	2.7
Fuel Pin Pitch-to-Diameter Ratio	1.096
Cladding Thickness, mm	1.0
Average Power Density, W/cm^3	69
Specific Power, KW/Kg heavy metal (HM)	10
Peak Power Density, W/cm^3	119
Average Discharge Burnup, MWd/Kg HM	72
Peak Discharge Burnup, MWd/Kg HM	120
Peak Fast Fluence, n/cm^2	4.0×10^{23}
Beginning of Cycle (BOC) to End of Cycle (EOC) Burnup Swing,% delta rho	0.13
Maximum Burnup Swing,% delta rho	0.36
Estimated Delayed Neutron Fraction	0.00375
BOC to EOC Burnup Swing, $	0.35
Maximum Burnup Swing, $	0.96

Reference Reactor System Development

Addm A4-1: Figure 3 shows the primary coolant system configuration. The Pb coolant flows upward through the core and the above-core riser region interior to the above-core shroud. Coolant flows through the holes in the shroud and enters the modular in-reactor heat exchangers to flow downward over the exterior of double-walled circular tubes arranged on a triangular pitch through which the S-CO_2 flows upward. Heat is thus transferred from Pb to S-CO_2 in a countercurrent regime. The Pb exits the heat exchangers to flow downward through the down-comer to enter the reactor vessel lower head. A flow distributor head provides for an approximately uniform pressure boundary condition beneath the core.



Addm A4.1: Figure 3. Illustration of reference LFR.

The SSTAR reactor system thermal-hydraulic development has been carried out to meet the following requirements and constraints:

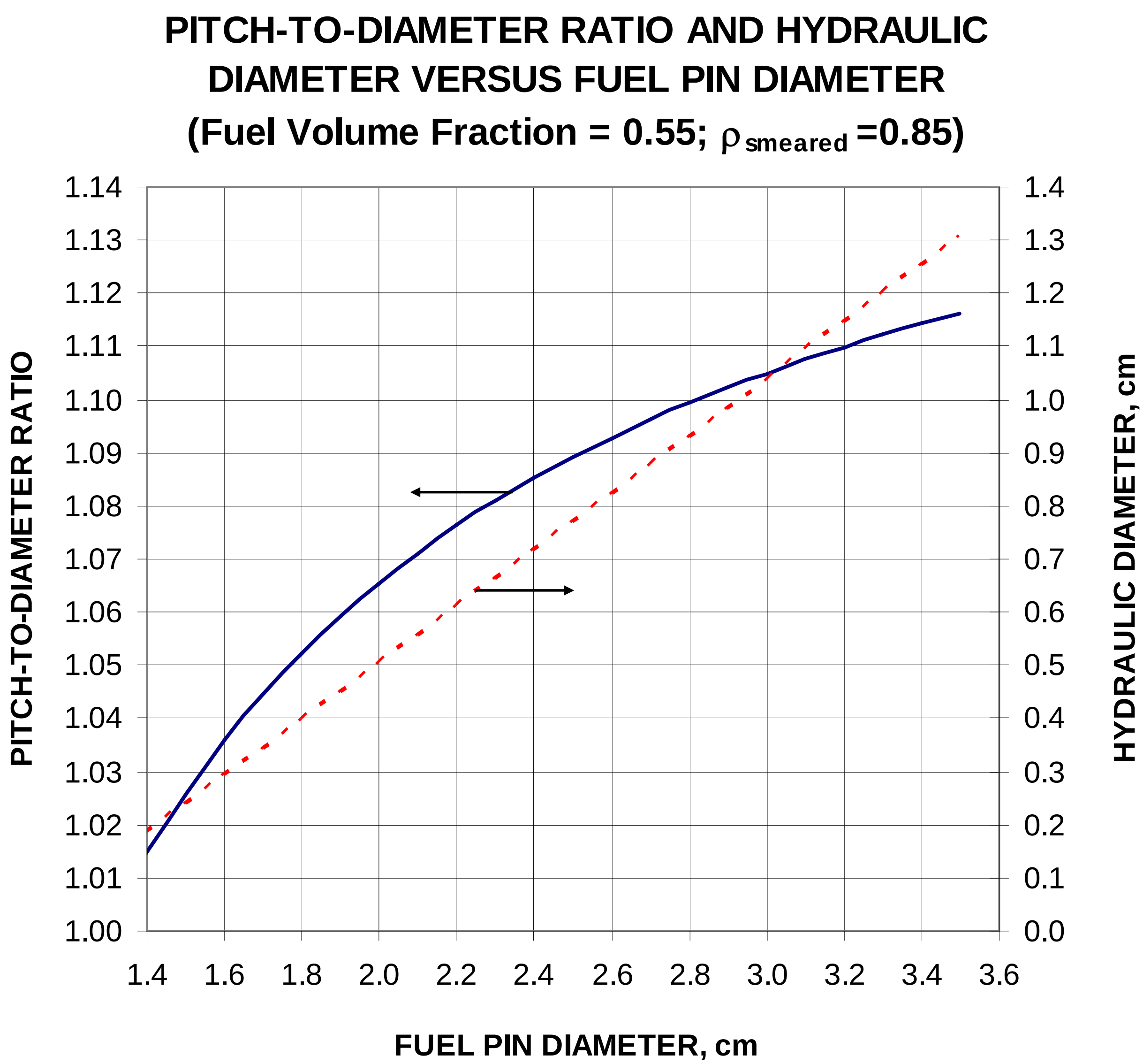
- Power level = 45 MW_t
- Full transportability by barge or rail
- Natural circulation heat transport of primary coolant at power levels up to and exceeding 100% nominal
- Core dimensions and fuel volume fraction from core neutronics analyses
- Power level = 45 MW_t

- Full transportability by barge or rail
- Natural circulation heat transport of primary coolant at power levels up to and exceeding 100% nominal
- Core dimensions and fuel volume fraction from core neutronics analyses
- Peak cladding temperature equal to 650°C
- Maximize S-CO_2 Brayton cycle efficiency
- Fission gas plenum height above active core equal to 1.75 times the active core height
- Pb coolant channels about 1 cm or more in diameter to reduce potential for plugging by contaminants
- Space for incorporation of cylindrical liner and annular gap escape path for CO_2 vapor/gas between in-vessel Pb-to-CO_2 heat exchangers and reactor vessel inner surface
- Space for multi-plate thermal radiation heat shield between bottom of upper head/cover and Pb-free surface
- Adequate coolant temperature margin above the freezing temperature
- Heat removal of decay heat from outside of guard/containment vessel to inexhaustible atmosphere heat sink by natural circulation of air.

The reactor vessel height is 18.3 m, with a diameter of 3.23 m. These dimensions meet the rail transportability size limitations and allow the following components to fit inside of the vessel to provide sufficient driving head for single-phase natural circulation heat transport between the elevations of the in-reactor heat exchangers and the active core:

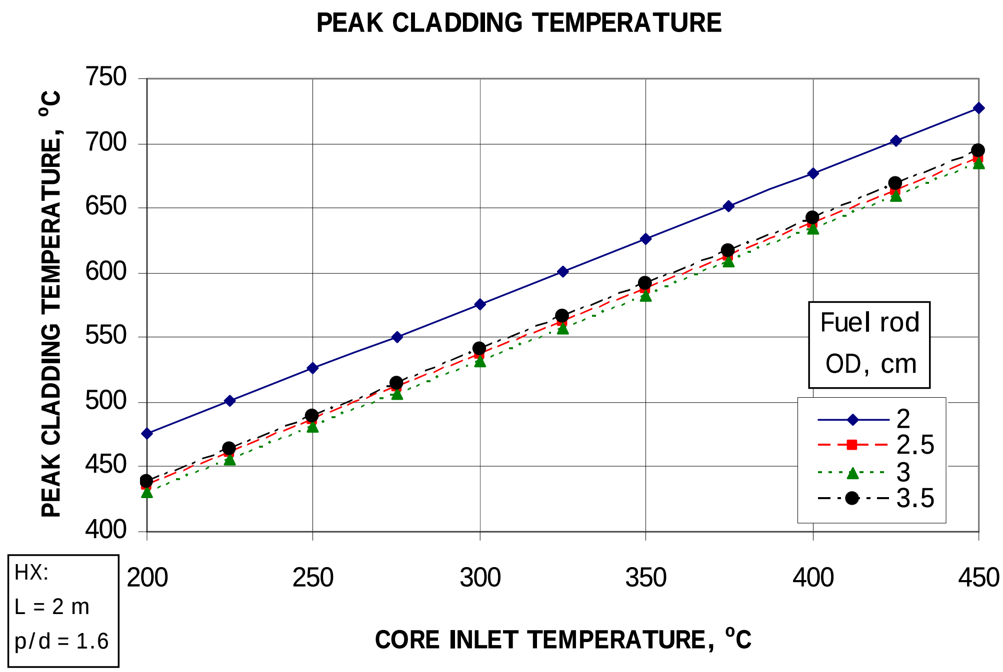
- 1.02-m-active core diameter
- 0.297-m-reflector thickness
- 2.54-cm-core shroud thickness interior to down-comer
- 5.72-cm-thick gap between reactor vessel inner surface and 1.27 cm thick cylindrical liner to provide escape path to Pb-free surface for CO_2 void, in the event of HX tube rupture
- 5.08-cm-thick reactor vessel
- Kidney-shaped Pb-to-CO_2 heat exchangers must fit inside the annulus between shroud and reactor vessel and provide sufficient heat exchange performance to realize a significant Brayton cycle efficiency.

The fission gas plenum height is based upon conservative accommodation of thermal creep at the peak cladding temperature of 650°C resulting from the fission gas pressure-induced hoop stress in the cladding over the 20-year core lifetime. The fuel volume fraction was held fixed in the thermal-hydraulic design analyses at the value of 0.55 determined by the core analyses. The fuel rod outer diameter and pitch-to-diameter ratio were varied to determine an optimum combination. Addm A4-1: Figure 4 shows the relationship between pitch-to-diameter ratio and rod diameter for a triangular lattice with a fixed fuel volume fraction of 0.55 and a fixed fuel smeared density of 85%.

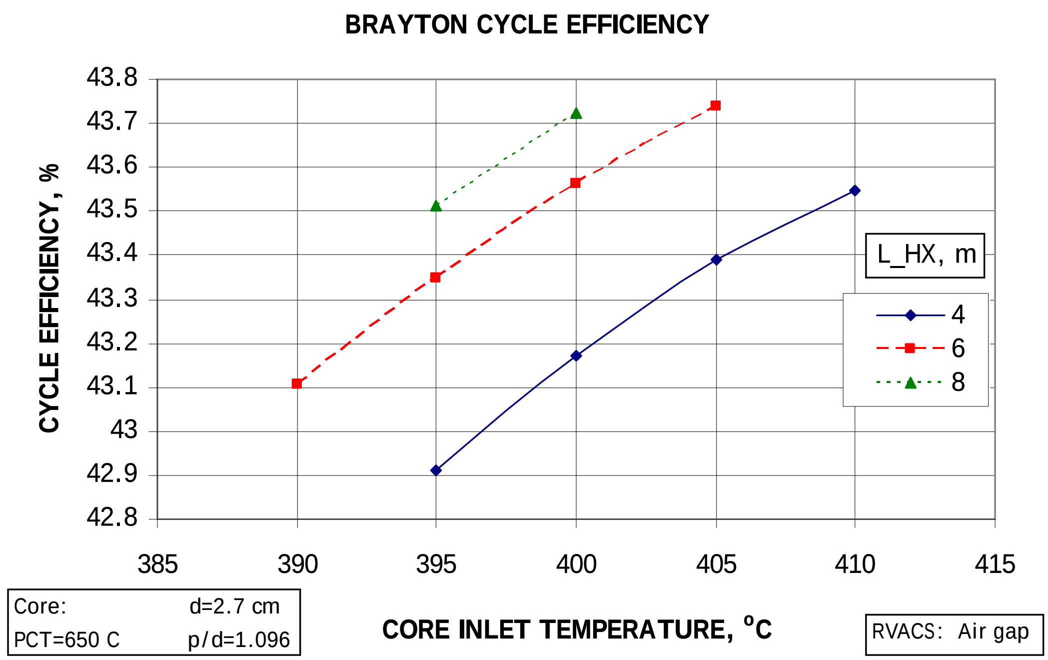


Addm A4.1: Figure 4. Relationship between fuel pin diameter and triangular pitch-to-diameter ratio.

Using this relationship, the fuel pin diameter is determined as the optimal value that minimizes the peak cladding inner surface temperature (assuming a 1.0 mm cladding thickness). Addm A4-1: Figure 5 shows the dependencies upon the fuel pin diameter and core inlet temperature with the frictional losses in the heat exchangers temporarily reduced. The heat exchanger tube height and pitch-to-diameter ratio are then determined to provide a 650°C peak cladding temperature and maximize the S-CO_2 Brayton cycle efficiency (Addm A4-1: Figure 6). Addm A4-1: Table 2 presents operating conditions for the 45 MW_t SSTAR coupled to an S-CO_2 Brayton cycle.



Addm A4.1: Figure 5. Dependencies of peak cladding temperature upon core inlet temperature and fuel pin diameter.



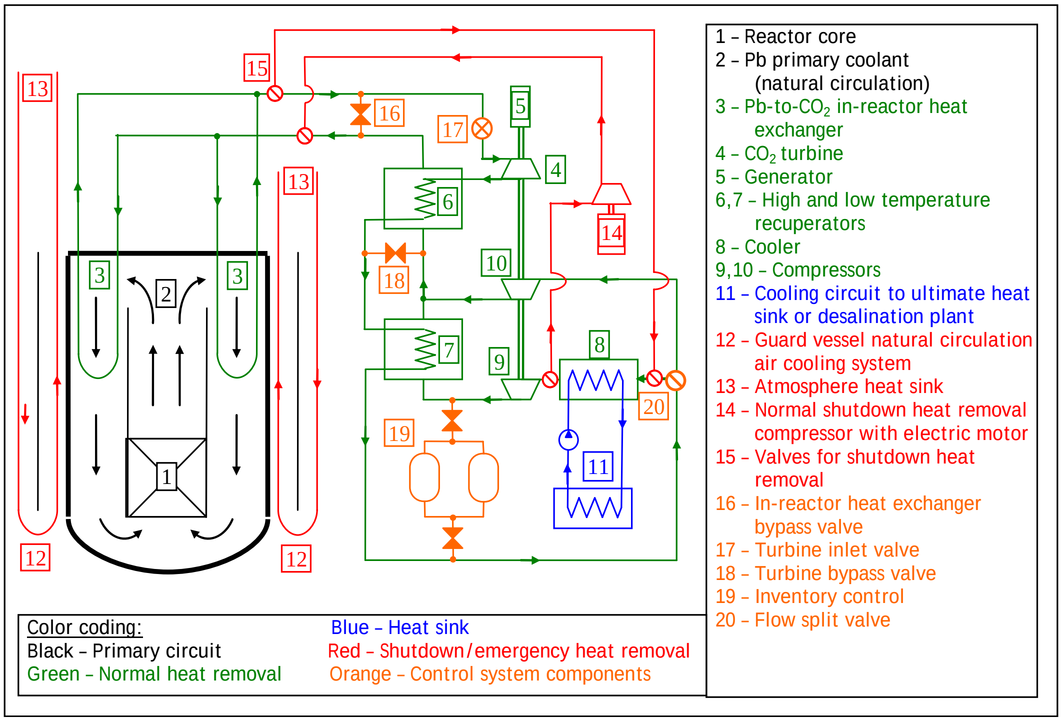
Addm A4.1: Figure 6. Dependencies of S-CO_2 Brayton cycle efficiency upon core inlet temperature and HX tube height.

Addm A4.1: Table 2. SSTAR operating conditions.

Parameter	Value
Power, MW_e (MW_t)	20 (45)
Reactor Vessel Height, m (feet)	18.3 (60.0)
Reactor Vessel Outer Diameter, m (feet)	3.23 (10.6)
Active Core Diameter, m (feet)	1.02 (3.35)
Active Core Height, m (feet)	0.80 (2.62)
Active Core Height-to-Diameter Ratio	0.8
Fuel Volume Fraction	0.55
Fuel Pin Outer Diameter, cm	2.7
Fuel Pin Pitch-to-Diameter Ratio	1.096
Core Hydraulic Diameter, cm	0.876
Cladding Thickness, mm	1.0
Fuel Smeared Density, %	85
HX Tube Height, m	6.0
HX Tube Outer Diameter, cm	1.4
HX Tube Inner Diameter, cm	1.0
HX Tube Pitch-to-Diameter Ratio	1.302
HX Hydraulic Diameter for Pb Flow, cm	1.22
HX-Core Thermal Centers Separation Height, m	12.2
Peak Fuel Temperature, °C	1009
Peak Cladding Temperature, °C	650
Core Outlet Temperature, °C	561
Maximum S-CO_2 Temperature, °C	541
Core Inlet Temperature, °C	405
Core Coolant Velocity, m/s	0.948
Pb Coolant Flow Rate, Kg/s	1983
CO_2 Flow Rate, Kg/s	245
S-CO_2 Brayton Cycle Efficiency,%	43.8

Supercritical-Carbon Dioxide Brayton Cycle Energy Conversion

The SSTAR reactor is coupled to an S-CO_2 Brayton cycle power converter that provides greater cycle efficiency at the Pb outlet temperature and has smaller, simpler, and fewer components as well as a smaller plant footprint relative to the traditional Rankine steam cycle. The general features of the S-CO_2 Brayton cycle are discussed elsewhere for the Energy Conversion Crosscut and in the literature. The present discussion shall therefore be limited to SSTAR-specific attributes. Addm A4-1: Figure 7 is a schematic of SSTAR coupled to the S-CO_2 Brayton cycle showing the heat transfer paths as well control mechanisms for the Brayton cycle. The turbine and two compressors are connected via a common shaft. This enhances the cycle efficiency and reduces the required generator power. Conditions for the turbine and compressors are presented in Addm A4-1: Table 3; the turbomachinery components are observed to have remarkably small sizes. The power conversion plant also incorporates a shutdown cooling compressor to circulate CO_2 through the in-reactor heat exchangers and the cooler to remove decay heat while allowing S-CO_2 Brayton cycle components to be isolated for maintenance or repair.



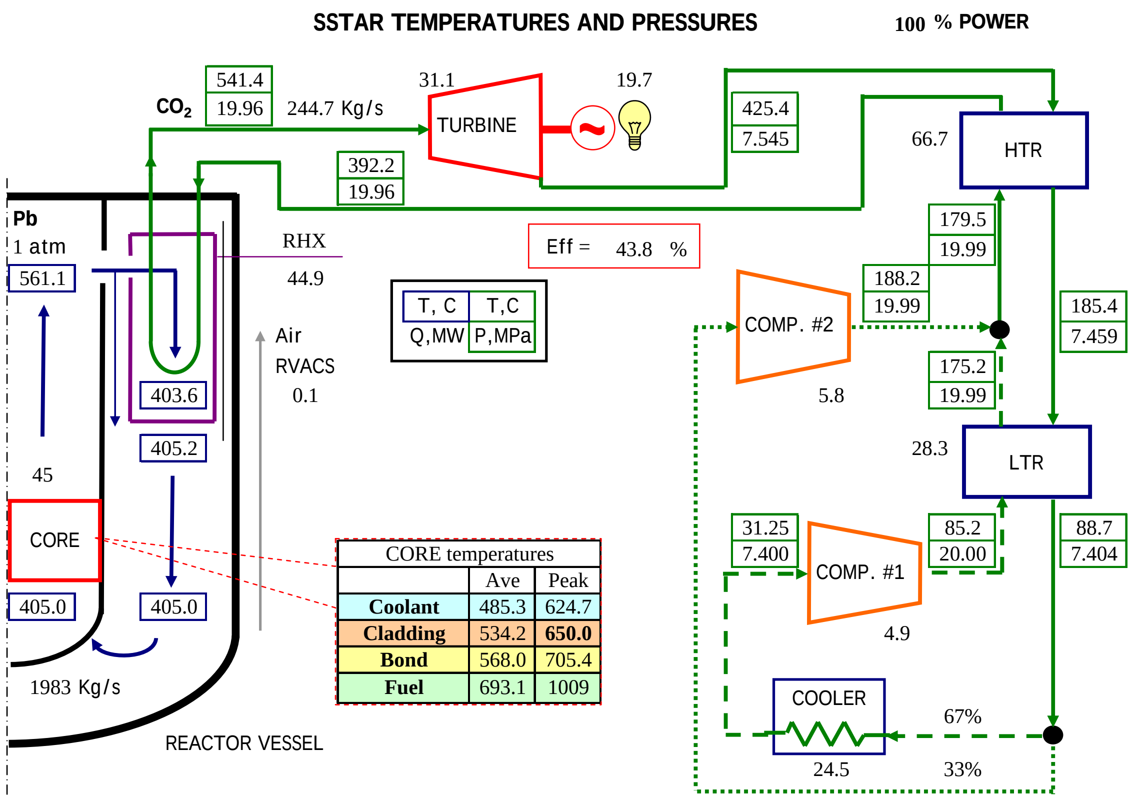
Addm A4.1: Figure 7. Schematic illustration of SSTAR coupled to S-CO_2 Brayton cycle showing normal, shutdown, and emergency heat transfer paths.

Addm A4.1: Table 3. Results of turbine and compressor analyses for 45 MW_t SSTAR.

	Turbine	Compressor No. 1	Compressor No. 2
Number of Stages	5	10	10
Length without Casing, m	0.41	0.26	0.14
Maximum Diameter without Casing, m	0.38	0.15	0.21
Efficiency without Secondary Losses,%	96.0	92.4	90.7
Assumed Secondary Losses,%	5.0	5.0	5.0
Net Efficiency,%	91.0	87.4	85.7

The two recuperators and cooler are assumed to consist of PCHEs (from Heatric, a subsidiary of Meggitt, Ltd.) in which millimeter-scale semicircular channels are chemically etched into plates that are hot isostatically pressed together at high temperature and pressure. Use of PCHEs offers the potential for savings in the recuperator and cooler volumes relative to shell-and-tube heat exchangers. However, it is assumed that the etched-plate manufacturing process limits the plate width to 0.6 m. To obtain the calculated required heat exchange area, twelve such PCHEs are incorporated to realize the high-temperature recuperator (HTR), low-temperature recuperator (LTR), and cooler. A concept was developed whereby the three components are assembled from three transportable modules. Each module consists of twelve PCHEs total: four 2.0-m-long PCHEs belonging to the high-temperature recuperator (located at the top); four 2.0-m-long PCHEs belonging to the low-temperature recuperator (in the middle); and four 0.72-m-long PCHEs of the cooler (at the bottom). The PCHEs are supported by a steel space frame.

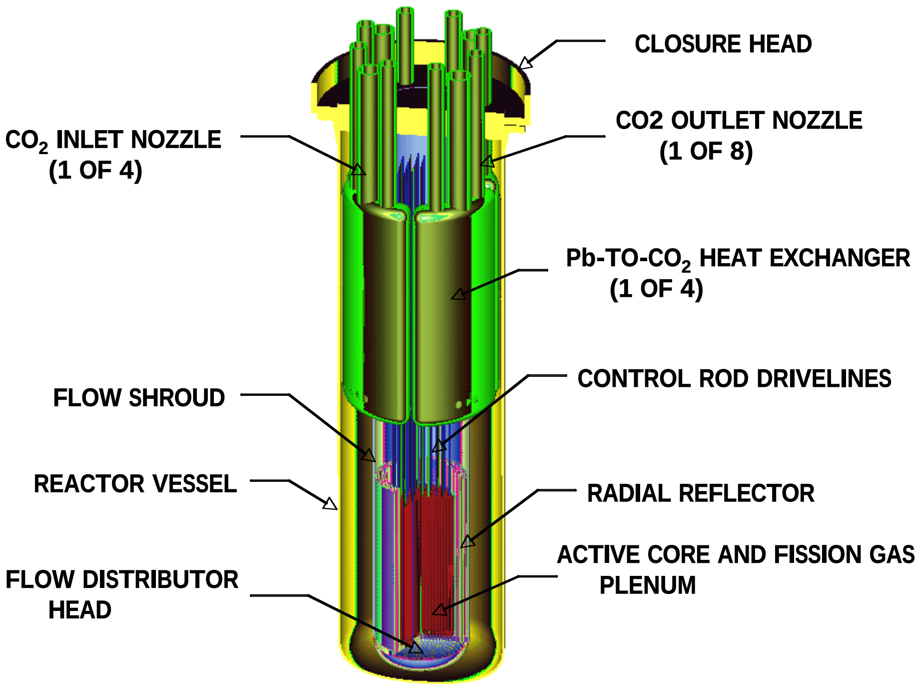
Pressures and temperatures for the Pb and S-CO_2 circuits are shown on the schematic in Addm A4-1: Figure 8.



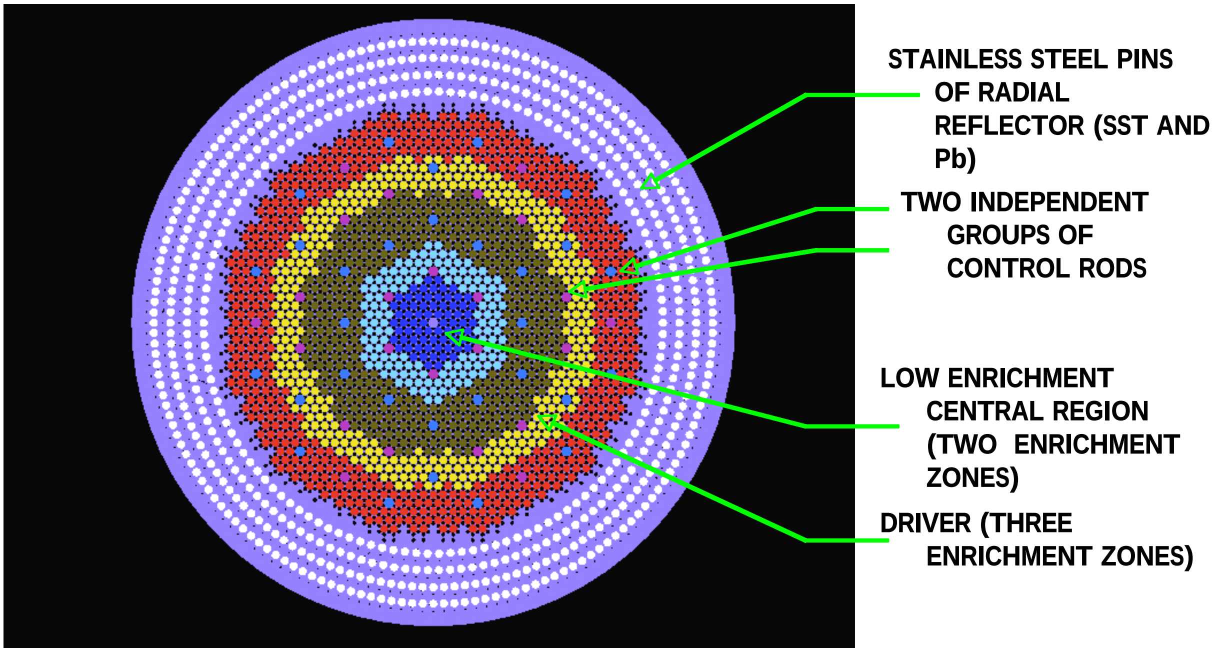
Addm A4.1: Figure 8. Schematic illustration of SSTAR coupled to S-CO_2 Brayton cycle showing temperatures, pressures, and heat exchange rates.

Thirty-Year Core Lifetime Lead-Cooled Fast Reactor Variant

Evolving from the LFR reference design concept, a variant has been evaluated to provide a 30-year core life and permit a shorter reactor vessel while retaining natural circulation (Addm A4-1: Figures 9 and 10). The design uses a core design with multiple enrichment zones and a lower fuel volume fraction of 0.45 that results in very little reactivity change throughout core life (less than $1 total burnup reactivity swing), flatter flux profiles and higher average discharge burnup, and a lower core pressure drop. The concept also uses compact high-efficiency in-vessel heat exchangers to retain natural circulation cooling while reducing overall vessel height from 18 m to 14 m.



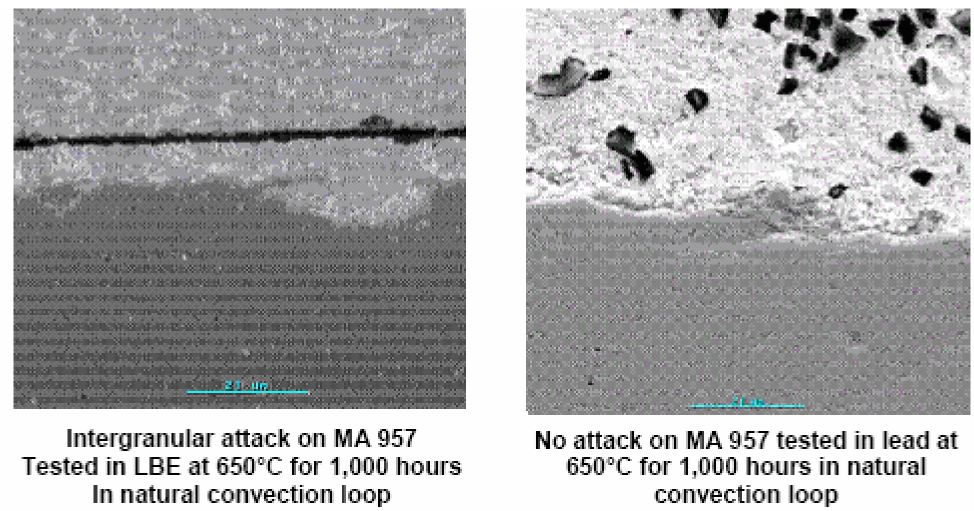
Addm A4.1: Figure 9. Illustration of 30-year core lifetime LFR-SSTAR variant.



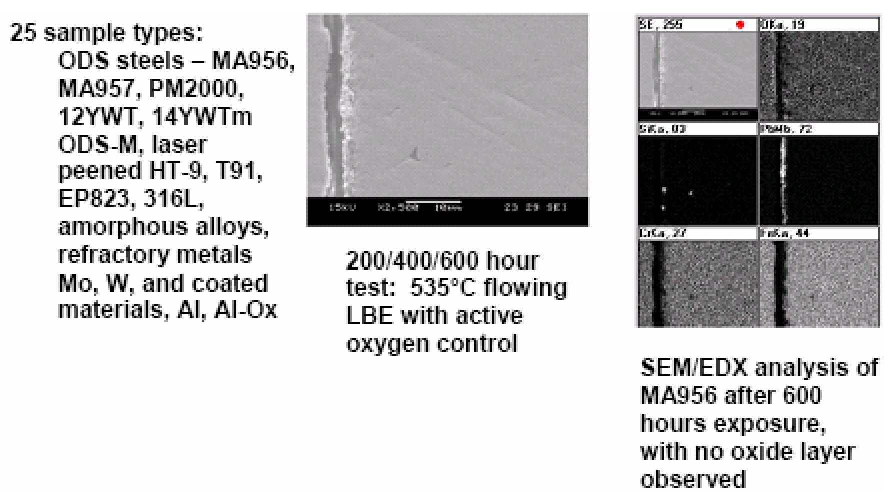
Addm A4.1: Figure 10. Thirty-year core lifetime LFR-SSTAR core variant.

Lead-Cooled Fast Reactor Materials Testing

Material corrosion in Pb or LBE is a significant issue for LFR cladding, in vessel components, and structural materials (Addm A4-1: Figure 11). Material development, testing, and modeling are major aspects of the LFR R&D plan. Test programs using the large forced flow LBE Development of Lead-Bismuth Target Applications (DELTA) Loop at LANL and small sealed Pb and LBE convection systems at ANL have produced encouraging results (Addm A4-1: Figure 12).



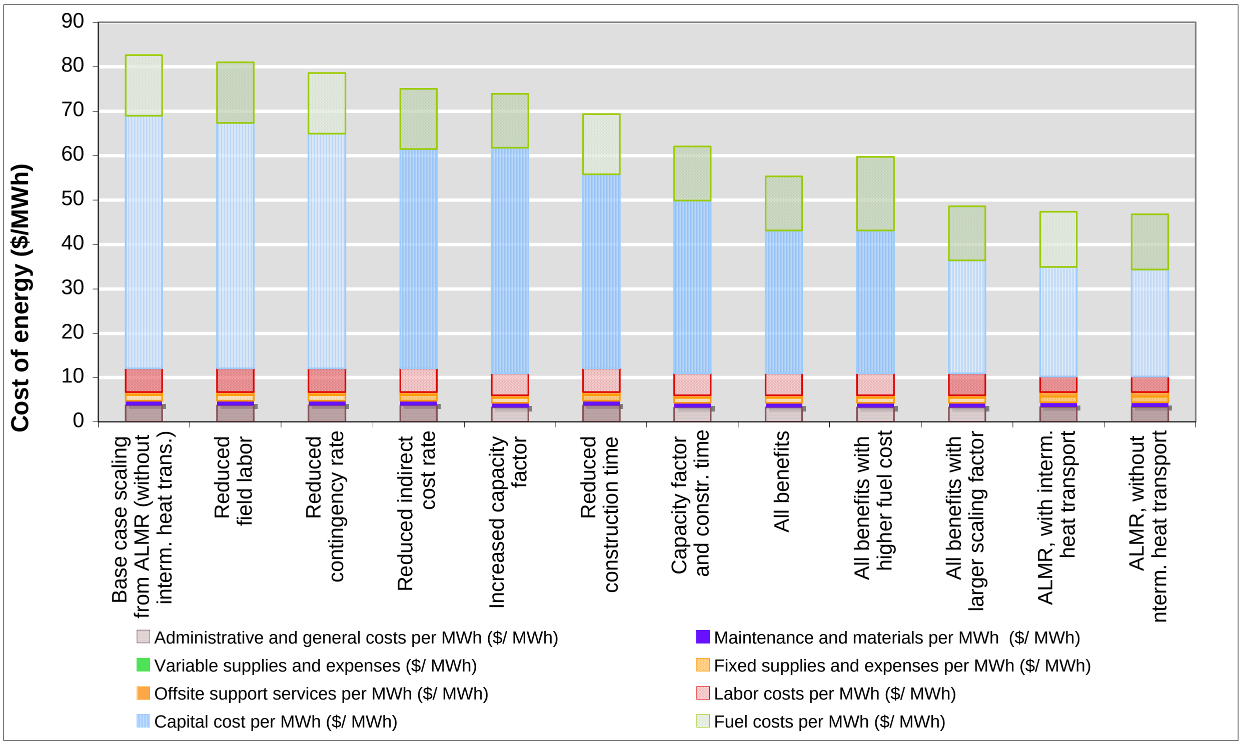
Addm A4.1: Figure 11. Material corrosion testing in Pb at up to 650°C.



Addm A4.1: Figure 12. Material testing in circulating LBE.

Lead-Cooled Fast Reactor Cost Factors

LFR cost factors and deployment costs are evaluated to better understand how LFR system design characteristics interact economically (Addm A4-1: Figure 13). Ongoing work suggests that one can expect a generation cost of \$50/MWh from 50 MW_e LFRs. The LFR receives cost benefits from factory production and rapid field installation, no fuel handling on site, no fuel outages, and simple reliable operations.



Addm A4.1: Figure 13. Cost factors for LFR-SSTAR.

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