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Supercritical CO_2 Direct Cycle Gas Fast Reactor (SC-GFR) Concept

Edward J. Parma, Steven A. Wright, Milton E. Vernon, Darryn D. Fleming,
Gary E. Rochau, Ahti J. Suo-Anttila, Ahmad Al Rashdan, and Pavel V. Tsvetkov

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

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Edward J. Parma, Steven A. Wright, Milton E. Vernon, Darryn D. Fleming, Gary E. Rochau
Sandia National Laboratories
P.O. Box 5800
Albuquerque, NM 87185-1136

Ahti J. Suo-Anttila
Computational Engineering Analysis
Albuquerque, NM 87112

Ahmad Al Rashdan, Pavel V. Tsvetkov
Texas A&M University
129 Zachary Engineering Center
College Station, TX 77843-3133

Abstract

This report describes the supercritical carbon dioxide (S-CO_2) direct cycle gas fast reactor (SC-GFR) concept. The SC-GFR reactor concept was developed to determine the feasibility of a right size reactor (RSR) type concept using S-CO_2 as the working fluid in a direct cycle fast reactor. Scoping analyses were performed for a 200 to 400 MWth reactor and an S-CO_2 Brayton cycle. Although a significant amount of work is still required, this type of reactor concept maintains some potentially significant advantages over ideal gas-cooled systems and liquid metal-cooled systems. The analyses presented in this report show that a relatively small long-life reactor core could be developed that maintains decay heat removal by natural circulation. The concept is based largely on the Advanced Gas Reactor (AGR) commercial power plants operated in the United Kingdom and other GFR concepts.

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Executive Summary

This report presents a relatively new transformational reactor concept that uses supercritical carbon dioxide (S-CO_2) as the coolant in a direct cycle gas fast reactor (SC-GFR). The concept is a combination of the CO_2-cooled Advanced Gas Reactor (AGR) developed and operated in the United Kingdom and the direct cycle Gas-Cooled Fast Reactor (GFR) concept.

The SC-GFR concept is a relatively small (200 MWth) fast reactor that is cooled with CO_2 at a pressure of 20 MPa. The CO_2 flows out of the reactor vessel at ~650°C directly into a turbine-generator unit to produce electrical power. The thermodynamic cycle that is used for the power conversion is a supercritical gas Brayton cycle with CO_2 as the working fluid. With the CO_2 gas near the critical point after the heat rejection portion of the cycle, it can be compressed with less power as compared to a standard gas Brayton cycle, thereby allowing for a higher thermal efficiency at the same turbine inlet temperature. A cycle efficiency of 45-50% is theoretically achievable for an optimized configuration.

The major advantages of the concept include the following:

- High thermal efficiency at relatively low reactor outlet temperatures;
- Compact, cost-effective, power conversion system;
- Non-flammable, stable, inert, non-toxic, inexpensive, and well-characterized coolant;
- Potential long-life core and closed fuel cycle;
- Small void reactivity worth from loss of coolant;
- Natural convection decay heat removal; and
- Feasible design using today's technologies.

The goal of this work was to develop a SC-GFR concept and perform scoping analyses, including a review of other similar concepts, to determine concept feasibility, advantages, disadvantages, and issues requiring further investigation. The scoping analyses included parametric thermal hydraulic and burnup analysis to determine core size and fuel pin dimensions, void reactivity worth, and the potential for natural circulation flow for decay heat removal from the reactor core. Overall, the SC-GFR concept as described in this report appears feasible and warrants further study. Additional research is required to determine resolution to important issues regarding the reactor and plant design, fuel burnup lifetime, safety, and economic viability.

This report documents the completion of the milestone in the work package "Argonne National Laboratory, MPO 0T-31542" due January 31, 2011.

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1. Introduction

The current trend in advanced power reactor concepts is to develop right size reactors (RSRs), grid appropriate reactors, and small modular reactors (SMRs) as alternatives to the current status quo, which are large (3000 MWth) light water reactors (LWRs) that will cost several billion dollars and many years to construct and license. Included in these advanced reactor concepts are small LWRs, liquid metal-cooled reactors (LMRs), high temperature gas-cooled reactors (HTGRs), molten salt-cooled reactors (MSCRs), and others. These advanced reactor concepts will use either a water-Rankine cycle or an advanced Brayton cycle for power conversion.

This report presents a relatively new RSR concept that uses supercritical carbon dioxide (S-CO_2) as the coolant in a direct cycle gas fast reactor (SC-GFR). The concept is a combination of the CO_2 cooled Advanced Gas Reactor (AGR) developed and operated in the United Kingdom (UK) (Shropshire, 2004) and the direct cycle Gas-Cooled Fast Reactor (GFR) concept (GIF, 2002).

The SC-GFR concept is presented in this report as a transformational reactor concept for electrical power generation and potential actinide burning. The nuclear reactor concept is a fast reactor that has the potential for a long burnup lifetime and operates as a direct cycle with an S-CO_2 power conversion system. The major advantages of the concept include the following:

- High thermal efficiency at relatively low reactor outlet temperatures;
- Compact, cost-effective power conversion system;
- Non-flammable, stable, inert, non-toxic, inexpensive, and well-characterized coolant;
- Potential long-life core and closed fuel cycle;
- Small void reactivity worth from loss of coolant;
- Natural convection decay heat removal; and
- Feasible design using today's technologies.

The reactor concept and fuel pin design are based largely on the AGR, a UK design which uses CO_2 coolant at 4.33 MPa (640 psia) and oxide fuel and stainless-steel cladding in the form of bundled fuel pins. The AGR design, however, is a thermal reactor using a graphite moderator matrix; it does not use a direct cycle and does not use a supercritical fluid. The CO_2 coolant circulates within a pressure vessel that contains the reactor, recirculators, and steam generators. The AGR CO_2 coolant has a mixed mean exit temperature of 650°C. The AGRs use a water-Rankine cycle that allows for thermal efficiencies of up to 40% (Shropshire, 2004). Although the AGRs and their predecessor Magnox reactors have been largely replaced by LWR technology, approximately fifty-two commercial power-producing reactors of this type have been built and operated throughout the world, and eighteen are still in operation (ANS, 2010). A wealth of information is, therefore, available regarding operational characteristics, safety issues, and the behavior of the fuel, cladding, and coolant.

The proposed SC-GFR concept operates at a power level of 200 MWth for 20 years. The direct cycle allows for a direct driven power turbine with no intermediate heat exchangers or recirculators. At a pressure of 20 MPa (3000 psia) and a reactor outlet temperature of 650°C,

thermal efficiencies of 45-50% can be achieved using the S-CO_2 cycle. At this operating pressure, the component hardware, including the heat-rejection heat exchanger and turbine, can be made orders of magnitude smaller as compared to a water-Rankine cycle. The rejection heat exchanger and recuperators would use advanced printed-circuit-type units that are compact and have a large surface area for heat transfer per unit volume. Since the reactor is a fast reactor, it can be designed to have high fuel conversion efficiency. Using a 12% enriched U-235 oxide fuel in the initial core loading, a small change in reactivity is calculated for the reactor operating at 200 MW for 20 years. After the core life is expended, the fuel's value remains high due to the remaining quantity of fissile material, which provides an economic incentive for reprocessing. The fuel would be reprocessed and recycled in subsequent core loadings. The lifetime of the core will ultimately depend on the amount of burnup that can be achieved in the fuel pins without significant leakers or failures. The reactor also maintains a small positive void reactivity worth from loss of coolant, which would only be observed for a major depressurization of the reactor vessel coolant.

One key advantage of the S-CO_2 direct cycle over a helium Brayton cycle is the capability to develop natural convection flow through the reactor and power conversion flow loop. This capability allows for decay heat removal from the reactor without the compressor operating. The CO_2 coolant is non-flammable, stable, inert, non-toxic, inexpensive, and well-characterized. Overall, this concept is feasible using today's technologies, materials, and fabrication techniques. The concept offers a potential cost-effective alternative to other advanced reactors that have been proposed. Concepts similar to this have been proposed by the Massachusetts Institute of Technology (Pope, 2004; Handwerk, 2007; Pope, et al., 2009) and the Tokyo Institute of Technology (Kato, et al., 2004).

Proposed Work

The goal of this work was to develop a SC-GFR concept and perform scoping analyses, including a review of other similar concepts, to determine concept feasibility, advantages, disadvantages, and issues requiring further investigation. The scoping analyses included the following:

- Review of other reactor systems that are similar to the SC-GFR concept;
- Review of the S-CO_2 cycle analysis;
- Parametric thermal hydraulic analysis to determine fuel pin dimensions and pitch;
- k effective and burnup analysis to determine reactor size and potential core lifetime;
- k effective analysis to determine void reactivity worth;
- Parametric analysis to determine natural convection flow and decay heat removal capabilities; and
- Plant layout, relative size of components, and cost effectiveness.

The intent of this work was to allow the reader to gain an understanding of the SC-GFR concept and its overall feasibility. Many issues and unknowns have been identified as a result of performing the scoping analysis. These issues will require further study and analyses and are delineated in a separate section of the report.

2. Review of Gas-Cooled Reactor Concepts

There have been at least seven previous programs that have evaluated fast gas-cooled reactors without moderating materials. These programs explored the use of helium, steam, and CO_2 (including S-CO_2) as the coolant, with pressures that varied from 0.5 MPa to 25 MPa. The most significant programs include the following:

(1) Germany, "The Gas Breeder Memorandum" (1969) – 3 concepts

(2) United States, General Atomics (1962)

(3) European Gas Breeder Reactor Association (1971) – 4 concepts

(4) Soviet Union (1970s) - concept used dissociating coolant;

(5) United Kingdom (1980s) – 2 concepts similar to the concept presented here but using larger reactors (up to 3600 MWth) with metal clad pins

(6) Japan (1970s and 1980s) – prismatic fuel reactor

(7) Generation IV International Forum (2002) - The GFR was selected as one of the six concepts to be evaluated as part of the Generation IV program for the Next Generation Nuclear Plant program.

All of these programs have produced extensive conceptual designs. In comparison with sodium, gas coolants have the following advantages for fast reactor applications:

- Chemical compatibility with water, obviating the need of an intermediate coolant loop, and generally good chemical compatibility with structural materials;
- Negligible activation of coolant;
- Optically transparent, simplifying fuel shuffling operations and inspection;
- Single phase coolants that cannot change phase in the core, reducing the potential for reactivity swings under accidental conditions;
- Reduction or elimination of the positive void effect typically associated with sodium; and
- Presence of a harder neutron spectrum, which increases the breeding and metal actinide burning potential of the reactor.

The disadvantages are generally considered to be the following:

- High pumping power;
- High pressure;
- Required surface roughening of cladding to enhance heat transfer;
- High coolant velocities; and
- Decay heat removal after shutdown and during a loss-of-coolant accident (LOCA).

Sandia National Laboratories (SNL) has gained a significant amount of experience in modeling, designing, building, and operating S-CO_2 power conversion research loops over the past several years. The SC-GFR concept was conceived from this work. Many of the disadvantages

identified for ideal gas GFRs may not be as detrimental for S-CO_2 systems. For example, the pumping power requirements are low for an S-CO_2 system because the fluid density of the coolant at the compressor inlet conditions is very high (60-70% the density of water). The fluid is nearly incompressible and, therefore, the pumping power is low, even with the expected core pressure drop. Likewise, high-pressure, high-density fluids mean lower gas velocities, smaller containment, smaller turbomachinery, smaller heat exchangers, and smaller piping, provided the reactor concept remains roughly below the 300-500 MWth power size. Similarly, decay heat removal can be addressed by using natural circulation mechanisms, in the event that the turbomachinery is inoperable. Large flow rates due solely to natural circulation mechanisms have been observed in the research loops. Through appropriate design of the power conversion system, natural circulation mechanisms could be implemented to provide passive decay heat removal features. There also exists the ability to provide large tanks of liquid CO_2 that can blow-down through the reactor to provide core cooling by venting through a break. These "blow-down" systems require no external pumps. Guard pressure vessels are often used in some GFR concepts to mitigate the effects of LOCA. CO_2 expansion cooling has been found to be extremely effective and is used to cool down hardware in the research loops. As observed in the S-CO_2 research loop, the venting and blow-down process through multiple ¼ inch valves can take hours. Appropriate design of emergency cooling systems could allow for effective core cooling available for long periods of time.

2.1 Comparison of the GFR Concepts

A comprehensive review of the GFR concepts is provided by van Rooijen (2009) covering the design concepts through 1980. More recently, the Department of Energy Office of Nuclear Energy (DOE-NE) Generation IV program has selected the GFR as one of its six preferred options. Research has identified literature describing four concepts from 2001 through 2009. One is by Idaho National Laboratory (INL) (Weaver and Khalil, 2002), another by the Tokyo Institute of Technology (TIT) (Kato, et al., 2004), a third by the Massachusetts Institute of Technology (MIT) (Pope, 2004; Handwerk, 2007; Pope, et al., 2009), and the last by the French Commissariat a l'Energy Atomique (CEA) (Dumas, et al., 2007). A fifth paper by British Nuclear Fuels (BNFL) (Newton and Smith, 2001) was reviewed that described a GFR based on the British Advanced Gas Fast Reactor. All of these systems provided basic design features for GFRs. Most used CO_2 as the primary coolant, although the CEA design used helium or a helium-nitrogen mixture. A summary of the major design features is provided in Table 1.

A review of this work confirms a number of design features. Most of the designs were developed for very large reactors, ~2400 MWth. The core sizes were large, 4-5 m in diameter, and most operated at power densities near 100 W/cc for the core. They all showed small reactivity consequences for a depressurization event (less than +$1.00 positive). Also, most of the concepts used active methods for decay heat removal. Several used guard vessels to mitigate the consequence of a LOCA to avoid full depressurization. Perhaps the most intriguing feature was that the concepts largely used non-conventional fuels. UC-SiC plates and UO_2-BeO fuel were proposed, even though these fuel types have not been fully developed, characterized, or tested. Only the TIT design and the BNFL design concepts used conventional fuel: metal fuel for the TIT concept and oxide for the BNFL GFR concept.

Table 1. Summary of Major GFR Design Features for a Variety of Different Concepts.

Program	British Energy	GEN-IV					
Org	BNFL	Japan TIT	INL	MIT	CEA	SC-GFR (SNL)	AGR
Author	Newton & Smith	Y. Kato	Weaver	Pope Lee 7 Hejzlar	Dumas	Wright & Parma	Shropeshire
Direct/Indirect	Indirect	Direct	Direct & Indirect	Direct	Direct	Direct	Indirect
Primary/2nd-PwrConv	CO2 /Steam	CO2	CO2 & He/CO2	CO2	He	CO2	CO2/Steam
Power MWth	3600 MWth	600 MWth	600 MWth	2400 MWth	600-2400 MWth	200-400 MWth	1500 MWth
Core Coolant Type	CO2	CO2	CO2	CO2		CO2	CO2
Core Pressure	4.2 Mpa	12.5 Mpa	7 Mpa	20 Mpa		13 Mpa	4.2 Mpa
Fuel Type	UO2/MOX	Metal	UPuC/SiC	UO2BeO	UC SiC	UO2/MOX	UO2 (Annular)
Clad Type	Austenitic SS	na (SS?)	SiC	ODS MA956	SiC	Austenitic SS	Austenitic SS
Mixed Mean Outlet Temp	525 C	527 C	550-650 C	650 C	850 C	650 C	650 C
Pressure Vessel Type	Steel Lined Pre-Stressed Concrete	Steel PV	Steel PV	Steel PV	Steel PV	Steel PV	Pre-Stressed Concrete
Pin Diameter	8.2 mm			Inverted Fuel	Plate Hexagonal	7 mm	15.76 mm
Pellet OD	7.14 mm			"NUT"			15 mm
Pellet ID	2 mm						5-6 mm
Core Diameter	not available	5.35 m	7 m (Vessel)	not available	4.44 m	2 m	
Core Height	1.5 m				1.55 m	2 m	
Power Density				20.7 kw/kg HM 85 kW/l core	100 kW/l-core	100 W/cc	2 kW/liter-core
Linear Heat Rating (w/cm)	230			151 kW/l fuel			20 W/cm2
Depressurizatoin Reactivity (pcm)	+ 160 to 300				+ 212 to 250	small +	
Reactivity Burnup Loss (pcm)	+1500 to 6400				-.07 Breeding Gain	< 1 $	
Burnup	20% h.a.		5 at%				
Fuel Re-Cycle Length efd	6 x 344		3 x 829		3 x 831	35 GWd/tonne	35 GWd/tonne
Cladding Limit	180 dpa		60 dpa			He Embrittlement	50,000(7yr) EFPH
Decay Heat Removal Method	Aux Motors on Redundant Main Circulators		Active and "Semi-passive"	Separate Nat Circ Decay Heat Removal Loop	Low Pwr Force Convection Nat Circ Long term	Natural Circ thru Brayton Cycle	Aux Motors on Redundant Main Circulators
Guard Containment				Yes 10 bar	Yes (10 bar)	Unknown Blow Down Tanks	No

For comparison purposes, the SC-GFR conceptual parameters are also listed in Table 1 along with the parameters for the AGR concept. The SC-GFR design is an SMR concept (200-400 MWth), using AGR oxide fuel technology and natural circulation for decay heat removal. Design features to deal with LOCAs have yet to be incorporated into the SC-GFR concept, but SNL's work explores the use of natural circulation, blow-down tanks, and guard vessels as potential safety features. The SC-GFR concept operates at power densities near 80-100 W/cc for the core, similar to the other concepts, but uses pin-type fuel with steel cladding. In general, the SC-GFR concept is most like the Japanese TIT concept, except that oxide fuel has been selected because the high core pressure may compress the cladding onto the fuel. At this operating pressure, it is doubtful that metal fuels could withstand the compressive stresses on the fuel without extruding the fuel into the fission gas plenum. The major difference between the SC-GFR concept and the others is that this effort proposes to use natural circulation as a passive safety mechanism to remove decay heat. This mechanism is described in a subsequent section of this report. Natural circulation for decay heat removal was proposed in the MIT concept for long-term cooling, while they relied on separate low-power pumps to provide forced cooling for short-term cooling.

2.2 Issues Generally Associated With GFR Concepts

The disadvantages that are generally associated with the GFR concepts are presented with a discussion of how they relate to the S-CO_2 concept.

High Pumping Power

Supercritical CO_2 reactors operate with the compressor very near the critical point of the working fluid. Under these operating conditions, the CO_2 entering the compressor is very dense – about 60-70% the density of water – and has little compressibility remaining in the fluid. As a consequence, the main compressor acts more like a pump than a compressor, allowing for the "pumping power" to be much lower than for ideal gas Brayton cycles. The small pumping power is one of the reasons for the potential high efficiency in S-CO_2 Brayton cycles.

High Pressure

The pressure in an S-CO_2 reactor is high, 20 MPa (3000 psia). There have been dual-turbine concepts proposed for an optimized S-CO_2 cycle (Muto and Kato, 2007) that can significantly reduce the reactor pressure. The pressure can be reduced by about 6-7 MPa (1000 psi) to allow for the operating pressure in the reactor to be near 13 MPa (2000 psi). This pressure is the same as for commercial pressurized water reactors.

Cladding Surface Roughening

Cladding surface roughening may be required to enhance heat transfer from the fuel pins. This is the approach that is use for the AGRs. More heat transfer and thermal hydraulic analysis is required to determine if cladding roughening is required for the SC-GFR concept.

High Coolant Velocities

The coolant velocities in particular portions of the S-CO_2 power conversion system can be high, depending on the piping size. From the reactor to the turbine, the coolant velocity is ~4.5 m/s for a 200 MWth plant and a pipe diameter of 1.6 m. Decreasing the pressure drop through the system is important in order to maintain a high thermal efficiency. Component integration and design will be important future research for S-CO_2 power conversion cycles.

Decay Heat Removal During Shutdown

Initial scoping calculations presented in this report indicate that natural circulation can provide enough cooling for a 200-400 MWth reactor core after shutdown. Further computational fluid dynamics analyses, along with experimental validation testing on the existing SNL S-CO_2 research loops, will be essential to show that decay heat removal via natural circulation is possible and can be designed into the SC-GFR concept.

Decay Heat Removal During a LOCA

Emergency core cooling and coolability of the reactor during a LOCA are significant design issues that must be addressed in future research. Blow-down concepts, evaporative cooling concepts, and guard vessels, in addition to natural circulation flow, are the major approaches to be considered for LOCAs.

3. Supercritical CO_2 Cycle

Supercritical CO_2 power conversion cycles have been studied significantly within the last decade as an alternate power conversion approach to couple to an advanced high-temperature nuclear reactor system. Water-Rankine cycles have historically been used in the power conversion system for all commercial nuclear power reactors and represent the current state-of-the-art technology. The S-CO_2 cycle, however, has been shown to have, at least theoretically, some significant advantages over the water-Rankine cycle that could allow it to be developed into a viable future technology, especially for advanced nuclear reactor systems.

A supercritical cycle is a gas Brayton cycle in which the working fluid is maintained near the critical point during the compression phase of the cycle. The supercritical properties near the critical point include higher gas densities, more similar to a liquid than a gas, allowing for the pumping power in the compressor to be significantly reduced as compared to a typical ideal gas Brayton cycle. This reduction in pumping power allows for the thermal efficiency to be significantly increased as compared to an ideal gas Brayton cycle at the same turbine inlet temperature. Another advantage of using a supercritical cycle is that the overall footprint of the power conversion system can be significantly reduced as compared to the same power output of a water-Rankine cycle. This is due to the high pressure in the system and resulting lower volumetric flow rate, which allows for the heat-rejection heat exchanger and turbine to be orders of magnitude smaller than for similar power output water-Rankine systems. Another potential advantage is the use of less water, not only due to the increased efficiency but also because the heat rejection temperature is significantly higher than for water-Rankine systems, allowing for significant heat rejection directly to air.

Different working fluids can be used in a supercritical power conversion system, including but not limited to CO_2, water, xenon, sulfur hexafluoride, sulfur dioxide, and ammonia. CO_2 is one of the most likely candidates as the working fluid because of a number of factors, including that the critical pressure is 7.38 MPa (1085 psia) and critical temperature is 31°C. The critical pressure for CO_2 is high but not an unattainable value; LWRs typically operate between 1000 and 2000 psia. The critical temperature is near the ambient temperature found world-wide. Other favorable characteristics of CO_2 are that it is non-flammable, stable, inert, non-toxic, inexpensive, well-characterized, and used in many industrial applications.

The S-CO_2 cycle offers a number of advantages over a water-steam Rankine cycle and other gas Brayton cycles in both efficiency and cost. The most significant advantages of using a S-CO_2 cycle include the following:

- Efficiency improvement due to smaller compression work required near the critical pressure, as compared to ideal gas Brayton cycles;
- High efficiencies at relatively low peak temperatures, ~650°C
- Relative compactness of component hardware, including the heat-rejection heat exchanger and the turbine, due to relatively large density of the gas at pressure;
- Critical point (31°C) is near the desired heat rejection temperature of 20°C;
- CO_2 is inexpensive, abundant, stable, non-toxic, inert, relatively non-corrosive, and is not flammable; and
- CO_2 is used in many industrial applications and is well-characterized.

Disadvantages of the S-CO_2 cycle include the following:

- CO_2 is corrosive at temperatures exceeding 500°C and requires components to be fabricated from high-cost metal alloys, such as stainless steel, or requires the use of passivating layers on components exposed to the high-temperature CO_2;
- S-CO_2 cycle requires high pressures to attain high efficiencies, ~20 MPa (~3000 psia).

Figure 1 shows the cycle thermal efficiency as a function of the heat source temperature for different cycles at typical conditions including water-Rankine (pink), helium Brayton recuperated with one turbine and one compressor (yellow), helium Brayton recuperated with three turbines and six compressors and interstage heating and cooling (light blue), and a S-CO_2 recuperated with split flow (dark blue). The S-CO_2 cycle has higher thermodynamic efficiency than for the water-Rankine cycle at temperatures greater than ~450°C. The S-CO_2 cycle efficiency is significantly greater that the nominal helium Brayton recuperated cycle with one turbine and one compressor over the complete temperature range. Only when the helium Brayton recuperated cycle has several interstage heating and cooling stages does it show greater efficiency than for the S-CO_2 cycle, and then only for temperatures greater than ~700°C. Hence the S-CO_2 cycle is clearly the cycle of choice for source temperatures greater than 450°C and lower than 700°C, if one considers efficiency improvement as the only factor in cycle selection.

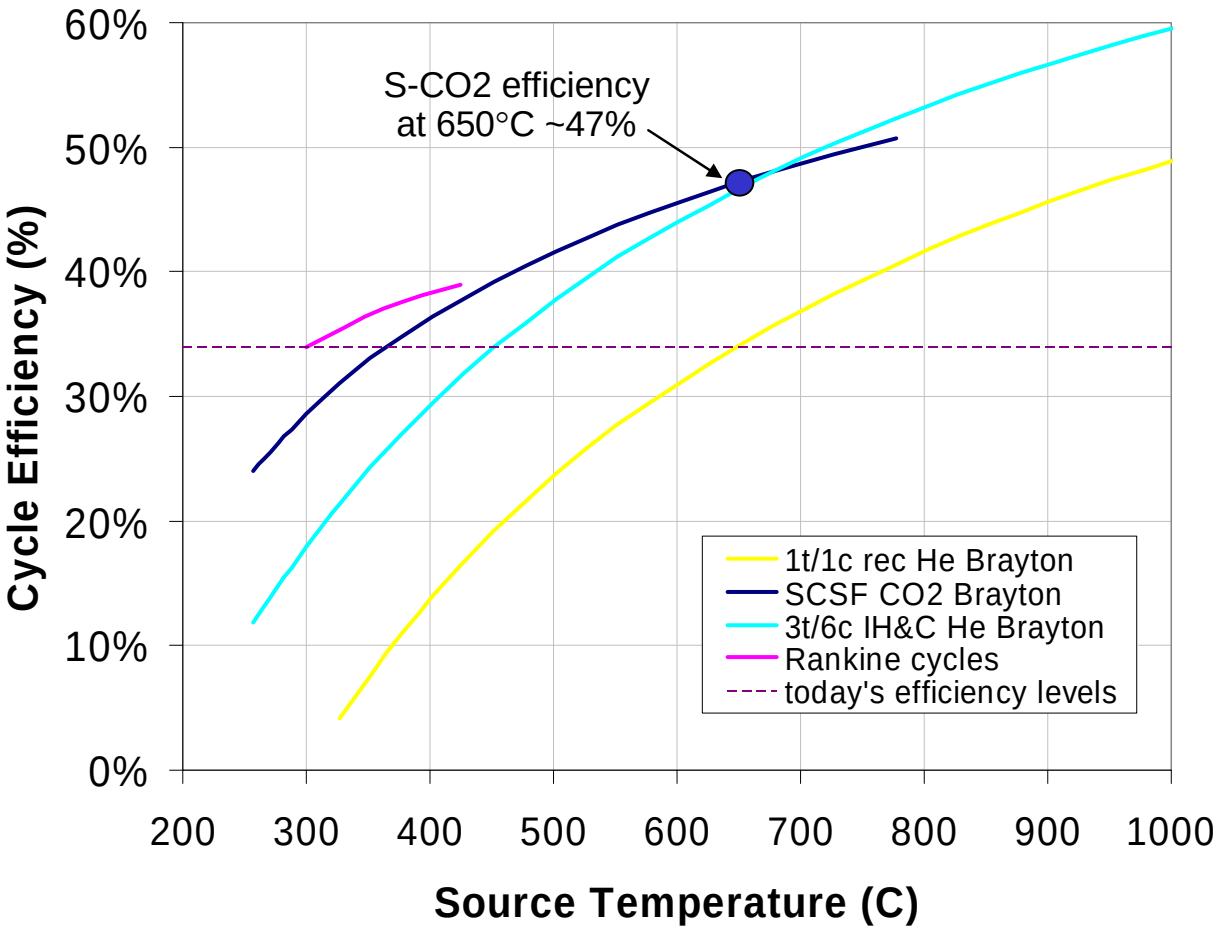

Figure 1. Cycle Thermal Efficiency as a Function of Heat Source Temperature.

SNL has two operating experimental S-CO_2 loops as part of the ongoing work to determine the feasibility of S-CO_2 power conversion systems. In addition, SNL has developed a number of computer codes to parametrically analyze thermodynamic cycles. The Excel spreadsheet code Flow Analysis Refprop was used to examine a typical S-CO_2 cycle with a split-flow configuration. The spreadsheet code uses the National Institute of Standards and Technology (NIST) Standard Reference Database 23 – REFPROP (NIST, 2007) for the S-CO_2 thermodynamic properties. The cycle is analyzed using input parameters including the desired output power level, heat rejection temperature, lower pressure value, compressor and turbine efficiencies, pressure ratio, fractional pressure drop in each component, heat exchanger effectiveness, and reactor exit coolant temperature.

Figure 2 shows a schematic diagram of the S-CO_2 cycle with a split-flow configuration and an annotated T-S diagram of the cycle. A split-flow configuration, referred to as flow recompression in some references, is the baseline configuration for this work. The split-flow configuration, proposed by Angelino (1968, 1969), allows for an increase in efficiency of several percentage points as compared to a simple recuperated cycle. The reason for this is that the heat capacities for CO_2 are significantly different as a function of temperature and pressure. By splitting the flow and allowing ~40% of the flow to bypass the heat-rejection heat exchanger, a more efficient cycle can be attained. The drawback to a split-flow configuration is the addition of a compressor and separate recuperator, adding more complexity and capital cost to the system.

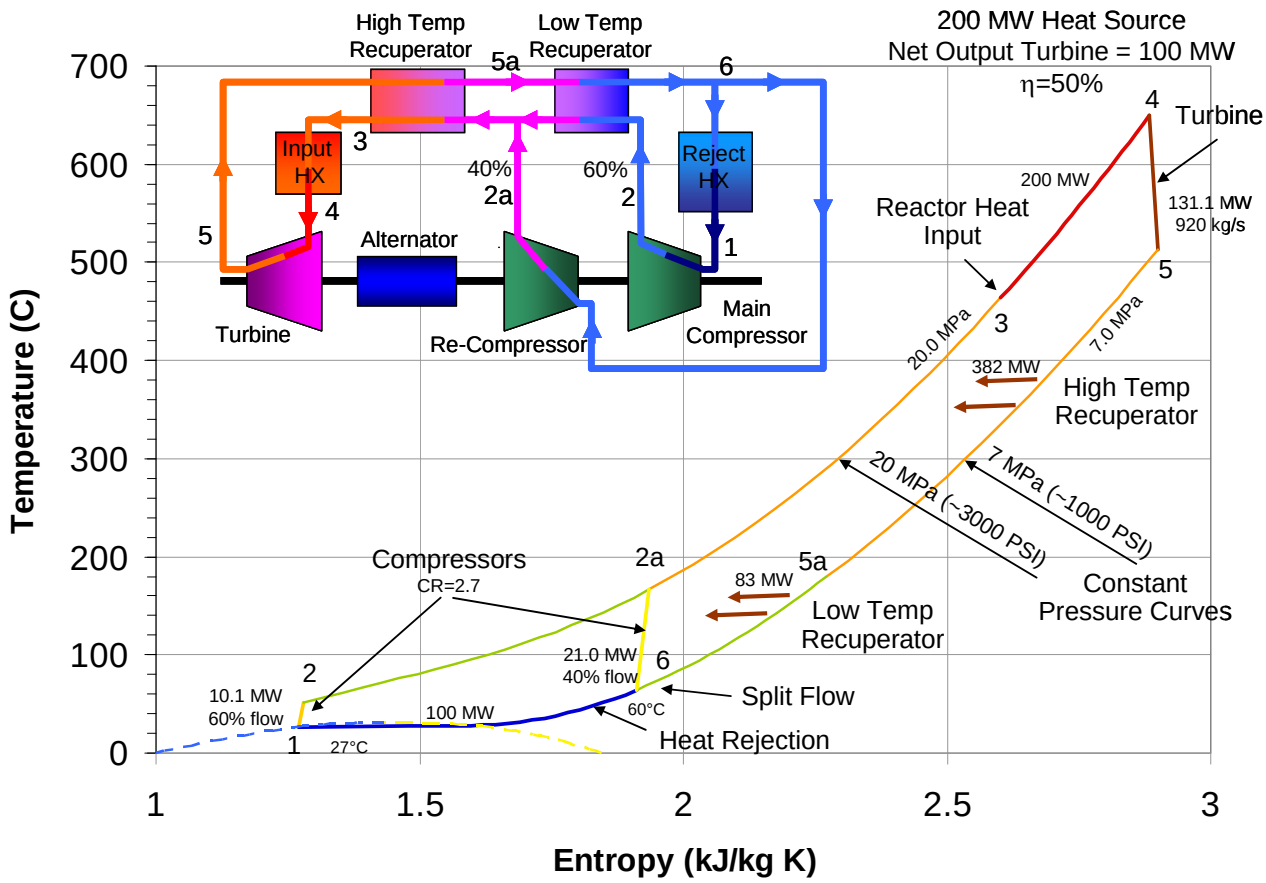

Figure 2. Flow Schematic and T-S Diagram for the Split-Flow S-CO_2 Cycle.

For the analysis shown in Figure 2, the thermal efficiency is found to be ~50%. This efficiency does not include windage or electrical power conversion losses, heat losses in the piping and other components, or other second order inefficiencies. The analysis was performed for an output power of 100 MW, heat rejection temperature of 20°C, low pressure value of 7.0 MPa (1030 psia), compressor efficiencies of 85%, turbine efficiency of 93%, pressure ratio of 2.7, total fractional pressure drop of 5%, heat exchanger effectiveness of 97%, and reactor coolant exit temperature of 650°C. The reactor input power is 200 MWth. The reactor inlet temperature is found to be 477°C and the coolant mass flow rate is ~920 kg/s.

Note that ~465 MW of thermal power is transferred from the hot side of the recuperators to the cold side. In order to use printed-circuit heat exchanger (PCHE) technology and a low pressure drop, the volume of each of the two PCHE recuperators and the heat rejection PCHE will be on the order of 10 m^3 each. For a stainless-steel PCHE, this equates to a mass of about 80,000 kg or 80 metric tons (MT). Using recent cost estimates for stainless-steel PCHEs, the cost per unit would be 8 to 17 million dollars ($M).

Radial turbine and compressor sizes are important parameters when costing out these components. Although detailed modeling is usually undertaken in the final hardware design, early turbine modeling can utilize the fluid velocity at the tips of the turbine blades. Compressor modeling can utilize a method known as "similarity" to initially size these components. For the main compressor unit, the radial compressor wheel is estimated to be 0.26 m (~10.2 inches) in diameter and operates at ~300 Hz. For a separate turbine unit operating the main compressor, the turbine wheel is estimated to be 0.22 m (~8.7 inches) in diameter. For the recompression compressor unit, the radial compressor wheel is estimated to be 0.22 m (~8.7 inches) in diameter and operates at ~500 Hz. For a separate turbine unit operating the re-compressor, the turbine wheel for is estimated to be 0.13 m (5.1 inches) in diameter. For a separate power generating turbine unit operating at 60 Hz, the turbine wheel is estimated to be 1.05 m (41.3 inches) in diameter with a blade height of about 5 cm (2 inches).

The piping or ducting size in the system is dependent on the pressure drop that is acceptable and the lengths of pipes between each component. As for the heat exchanger, smaller components can be made if higher pressure drops and corresponding efficiency losses are acceptable. Allowing for a fractional pressure drop of 0.1% in the piping, the piping diameter from the reactor to the turbine is about 1.6 m. Increasing the fractional pressure drop to 1% for the piping, decreases the overall efficiency by ~3%, and decreases the reactor to turbine piping diameter to 1.0 m.

More work is required to optimize the component geometry and integrate components in order to develop a power conversion system that can be built and operated efficiently. Integration of the power conversion system with the reactor pressure vessel will also be an important consideration.

4. Reactor Core Conceptual Design and Plant Layout

The SC-GFR concept is a gas-cooled RSR concept where the overall size and output power level is commensurate with modular factory construction of the pressure vessel, reactor, and power conversion system, and with the overall capital cost maintained at a level of less than $5,000 per kilowatt of electrical power output. The plan is that the facility would be built and the reactor and power conversion system would then be shipped to the facility and installed. The licensing process would be similar to that proposed for current power reactors being considered. With this consideration, one of the main focus areas of the SC-GFR concept is to keep the reactor and pressure vessel as small as reasonably possible, while still allowing for a reasonable power level and operating history.

4.1 Reactor Fuel and Core Description

A 200 MWth reactor system was chosen as a reasonable reactor power output for an RSR type system, although additional work will be presented in this report for a reactor power of 400 MWth. In order to determine the core size, enrichment, fuel pin diameter, pitch, reactor diameter, fuel pin length, and burnup lifetime, a number of objectives had to be established. These objectives are as follows:

- Core power level of 200 MWth;
- Core reactivity burnup life of ~20 years;
- Minimal reactivity change over core lifetime;
- Core pressure drop less than 1% of total reactor power;
- Small reactivity void coefficient;
- Acceptable cladding and peak fuel temperature; and
- Acceptable fuel and cladding burnup.

A number of iterations were performed between thermal hydraulic and burnup analyses in order to converge on an optimum conceptual configuration that incorporates all of the aspects of the objectives. The thermal hydraulic analysis, burnup, and k effective (k_{eff}) analyses will be presented in the following sections of this report. Although more complex and rigorous analyses will follow in future work, the results presented represent a good first estimate of a conceptual core design.

Figure 3 shows a Monte Carlo N-Particle code (MCNP, 2003) neutronics model of the reactor core. The core was modeled in three dimensions with each fuel pin individually specified. The fuel pins are set on a triangular pitch. Each fuel pin is cylindrical with fuel, gap, and cladding specified. The core is cylindrical with a reflector surrounding it. For this conceptual stage of the work, no control rods or other hardware were included in the design.

Figure 4 shows a conceptual illustration of the reactor vessel and core. The reactor vessel will most likely be fabricated from a high-nickel content stainless steel, to reduce corrosion over its lifetime. Other lower-cost steels may be considered in the future if it can be shown that the corrosion rate is slow for the inlet coolant temperature, or if liners can be used with thermal breaks to reduce the temperature of the vessel. Other vessel configurations can also be considered, including a liner within a pre-stressed concrete structure.

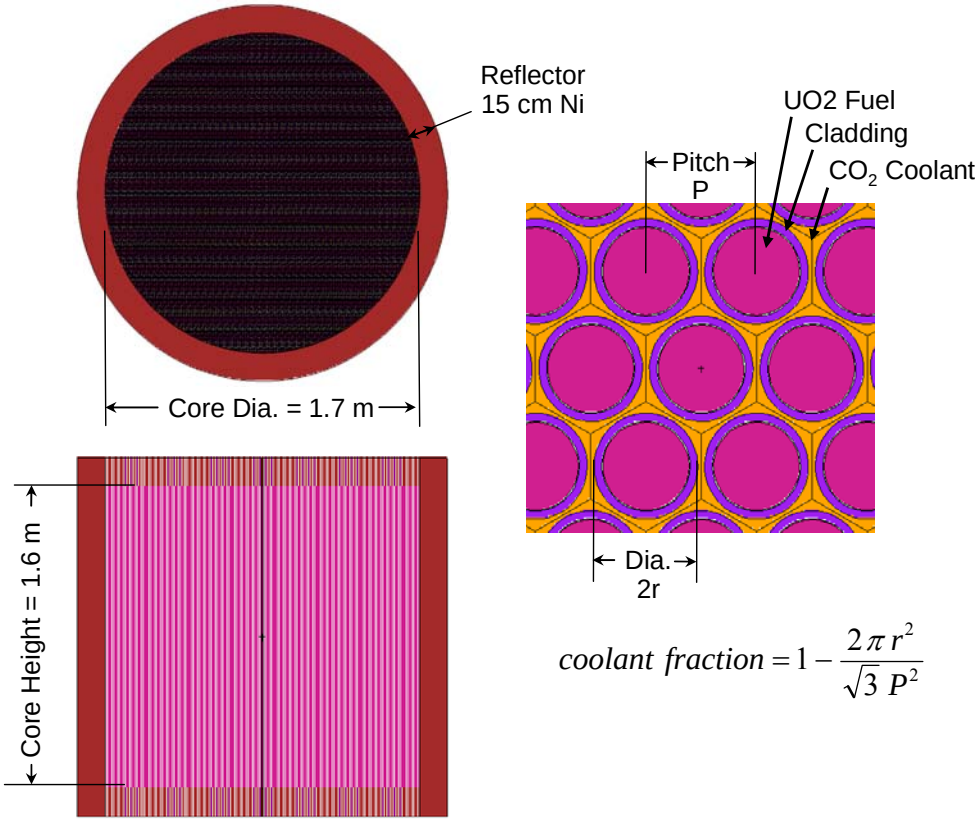

Figure 3. MCNP Model of the Reactor Core.

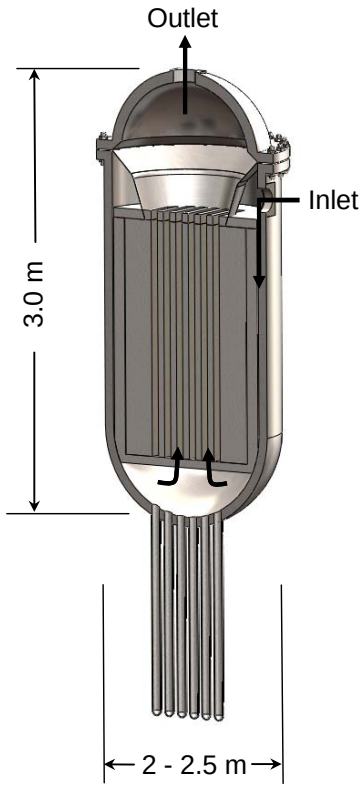

Figure 4. Conceptual Illustration of the Reactor Pressure Vessel and Core.

The vessel size is currently configured to be about 2 to 2.5 m in diameter and about 3 m in height. The vessel wall thickness will be on the order of 10 cm or greater. As the vessel is currently configured, the coolant enters the side of the vessel near the upper bulkhead, travels down the downcomer along the vessel wall to the bottom plenum of the vessel, and then moves upward through the core. The fuel pins maintain the active fuel region and a plenum region for fission gas retention. In order to maintain the correct mixed mean temperature at the core exit, flow redistribution within the core will be required. This can be performed by orifices at the inlet plenum to the core, or by adjusting the pitch in the core from the inner rows to the outer rows. Future work will be required in this area. The flow exits the pressure vessel though the top of the upper bulkhead. Other configurations, including hot pipe exiting in the cold pipe inlet, can be considered.

The reactor will be required to have some type of control rod configuration. No significant work has been performed to determine the best approach for this concept. The current configuration has the control rods entering from the bottom of the core and through the lower bulkhead of the pressure vessel. However, the control rods could just as well be configured from above since the coolant exit temperatures are not extreme and are below the Curie point temperature for most magnetic and ferromagnetic materials.

Pressure vessel embrittlement due to radiation damage and corrosion effects will play a major role in determining the vessel's material, wall thickness, lifetime, and working pressure. Additional neutron moderating and absorbing materials will be required to be placed outside of the reactor core reflector and within the vessel to decrease the fluence of the fast neutrons on the vessel wall. Future work is required in this topical area.

Tables 2 through 4 present the power plant parameters, reactor fuel parameters, and reactor core parameters, respectively. For a split-flow S-CO_2 cycle, the thermal efficiency will probably be between 40 and 50% for conditions that are given in Table 2. The baseline concept assumes a reactor coolant pressure of 20 MPa (~3000 psia) and a reactor outlet temperature of 650°C. The fuel proposed in this concept is UO_2 enriched to 12%. The choice for the enrichment will be discussed in a later section of the report. UO_2 was chosen somewhat arbitrarily. The AGR systems use UO_2 and it is expected that it should be compatible with CO_2. Other fuel options can be considered in the future, including bonded metal fuels. However, it is expected that UO_2 will be the fuel of choice due to a number of considerations including operating experience, compatibility with the cladding and coolant, and performance reliability.

The cladding proposed in the conceptual design is a high-nickel content stainless steel, such as a 316-type material. The nickel is required to ensure corrosion resistance at CO_2 temperatures up to 650°C. The cladding will most likely be the weak link in the lifetime burnup of the core. The AGR systems burn their fuel to about 24,000 MWD/MTU. The current concept, 200 MWth for 20 years, has a fuel burnup of 71,000 MWD/MTU. Current LWR technology allows for ~60,000 MWD/MTU. A fission gas plenum will be required in the upper portion of the fuel pin. This plenum will be an extension of the cladding. During the lifetime of the fuel pin, the cladding will be in compression due to the high pressure coolant. For this concept, the fission gas plenum height was chosen to be 1 m. Additional work is required to determine the expected fuel and cladding performance over the desired burnup lifetime due to corrosion and neutron damage.

Table 2. Power Plant Parameters for the Baseline S-CO_2 Concept.

Parameter	Value
Reactor Power (MW)	200
Cycle Efficiency	50%
Output Power (MW)	100
Coolant Pressure in Reactor (MPa)	20.0
Coolant Pressure Before Compression (MPa)	7.0
Reactor Inlet Temperature (°C)	450
Reactor Outlet Temperature (°C)	650
Coolant Mass Flow Rate (kg/s)	920
Reactor Power Density (kW/l core)	55.1
Specific Power (kW/kg HM or MW/MTU)	10.0
Reactor Core Pressure Drop (MPa)	<0.3
Proposed Reactor Lifetime (yr)	20
Average Fuel Burnup (MWD/MT)	71,000

Table 3. Reactor Fuel Parameters.

Parameter	Value
Fuel and Density	UO_2 10.3 g/cc
Cladding	High Ni Stainless Steel
Enrichment (%)	12.0
Overall Pin Diameter (cm)	0.75 and 1.20
Fuel (UO_2) Diameter (cm)	0.622 and 1.072
Gap Thickness (cm)	0.008
Cladding Thickness (cm)	0.056
Active Fuel Length (cm)	160
Fuel Pin Gas Plenum Length (cm)	100
Coolant Fraction	0.2 and 0.3
Pitch – Triangular (cm)	0.799 and 0.854

Table 4. Reactor Core Parameters.

Parameter	Value
Active Core Diameter (cm)	170
Active Core Height (cm)	160
Reflector Material	Nickel (Ni)
Reflector Thickness (cm)	15
Active Core Volume (m^3)	3.63
Total Mass of Fuel Meat (kg)	20,600

The core pressure drop is a function of the core size, coolant flow rate, the fuel pin diameter, pitch, and pin length. The design objective is to maintain the pumping power though the core to a value less than 1% of the total core power, which is achievable for a pressure drop less than 0.3 MPa (44 psi). Two different fuel pin diameters and coolant fractions were analyzed for the concept: a 0.75 cm pin diameter with a coolant fraction of 0.2 and a 1.20 cm pin diameter with a coolant fraction of 0.3. For the same size reactor core, the fuel mass is almost equal. The impact on fuel temperatures and pressure drop through the core for different fuel sizes and coolant fractions will be discussed in a later section of this report. Other configurations may also be shown to be acceptable.

The nickel reflector material and thickness were chosen somewhat arbitrarily. A high Z material with good scattering and coolant compatibility properties is desired. Materials with moderating properties were found to increase k_{eff}, but have deleterious effect on the burnup reactivity changes. Other materials may be found to work adequately as a reflector material. The reflector was modeled as a solid unit in the MCNP model, but can be made into pins or a solid with coolant channels. The reflector will be located in the downcomer section of the coolant.

The reactor radius, height, and enrichment were chosen to minimize the change in reactivity over the core lifetime. For a 200 MWth power level, a small core is conceivable with an enrichment greater than 12%. However, burnable poisons and/or a higher worth reactivity control system would be required to maintain the reactor critical throughout the lifetime of the core. The core configuration in this concept allows for significant conversion of the U-238 to Pu-239 with only a small reactivity change over the core lifetime. This requires a somewhat larger core, but allows the fuel cycle to be sustainable. The first core loading would contain 12% enriched uranium fuel. Subsequent loadings would have larger quantities of the recycled Pu. The first core loading, 12% enriched and 20,600 kg, would cost approximately $150M. This cost would be a significant portion of the initial capital investment in the plant at $1,500/kW electric. However, assuming that the fuel would last 20 years, this capital investment would equate to a cost ~1.5 cents per kW-hr electric, which is not that much greater than for LWR fuel at ~1 cent per kW-hr electric. Subsequent cores using the reprocessed, recycled fuel would cost significantly less, on the order of ~$30M, since only reprocessing and fuel make-up costs would be required.

4.2 Facility and Plant Layout

A conceptual plant layout is shown in Figures 5 to 7 for different possible S-CO_2 system configurations. These layout designs are being modeled in three dimensions using the SolidWorks engineering design code. Currently the designs have only the reactor vessel, piping and power conversion system layout. Although not completely to scale, the relative sizes have been factored into the design. The pipe sizing and layout and the integration of components will be important future considerations in developing workable and realistic plant design. Future work will incorporate conceptual ideas for the building and containment structure, auxiliary systems, and more realistic sizing information.

The reactor vessel will most likely be located in a below-grade vault that will provide shielding and auxiliary cooling for the vessel. The turbine/compressor unit and recuperator will be at ground level. The heat-rejection heat exchanger may be at ground level or above ground level, depending on the height requirements to ensure natural convection flow capabilities for decay heat cooling with the compressor not operating.

Other auxiliary systems will include, for example, a CO_2 make-up, recovery, and purification system; emergency core cooling system; cooling water system; and containment ventilation system.

Figure 5 shows a configuration for a split-flow S-CO_2 cycle with a combined turbine, compressor, and generator on the same shaft. Two PCHE recuperators are required along with the PCHE heat rejection system. For a 100 MW electric unit, and the pressures and temperatures specified previously, the high-temperature PCHE recuperator would be about 10 m^3 in size, the low-temperature PCHE recuperator about 9 m^3, and the heat-rejection PCHE about 7 m^3. The sizes of the compressors, turbine, and generator have not yet been identified for this concept. However, they will be relatively small compared to a water-Rankine cycle due to the pressure/density of the working fluid and their rotational speeds.

Figure 6 shows a variation on the power conversion cycle by separating the power generating turbine/generator unit from the rest of the system. This approach adds complexity to the system but allows for power to be generated at 60 Hz. The compressor unit would maintain its own turbine and motor/generator for starting the system and maintaining energy efficient operation.

Many other schemes are conceivable to optimize the system performance or allow for other considerations. For example, splitting the flow a second time in the high-temperature recuperator region could increase the efficiency by another few percentage points. Figure 7 shows a scheme devised by Muto and Kato (2007) to allow for the reactor coolant pressure to be significantly reduced. By placing a power-generating turbine after the high-temperature recuperator, but prior to the reactor, the pressure in the reactor vessel can be reduced from ~20 MPa (3000 psia) to ~13 MPa (2000 psia) with only a small loss (~1%) in efficiency. The ultimate power conversion scheme that is used will depend on research conducted over the next several years on S-CO_2 test systems that would be scalable to 100 MW electric. Until further experimental work is performed on these types of scalable test units, optimizing a system for efficiency, cost, reliability, and complexity is difficult and speculative.

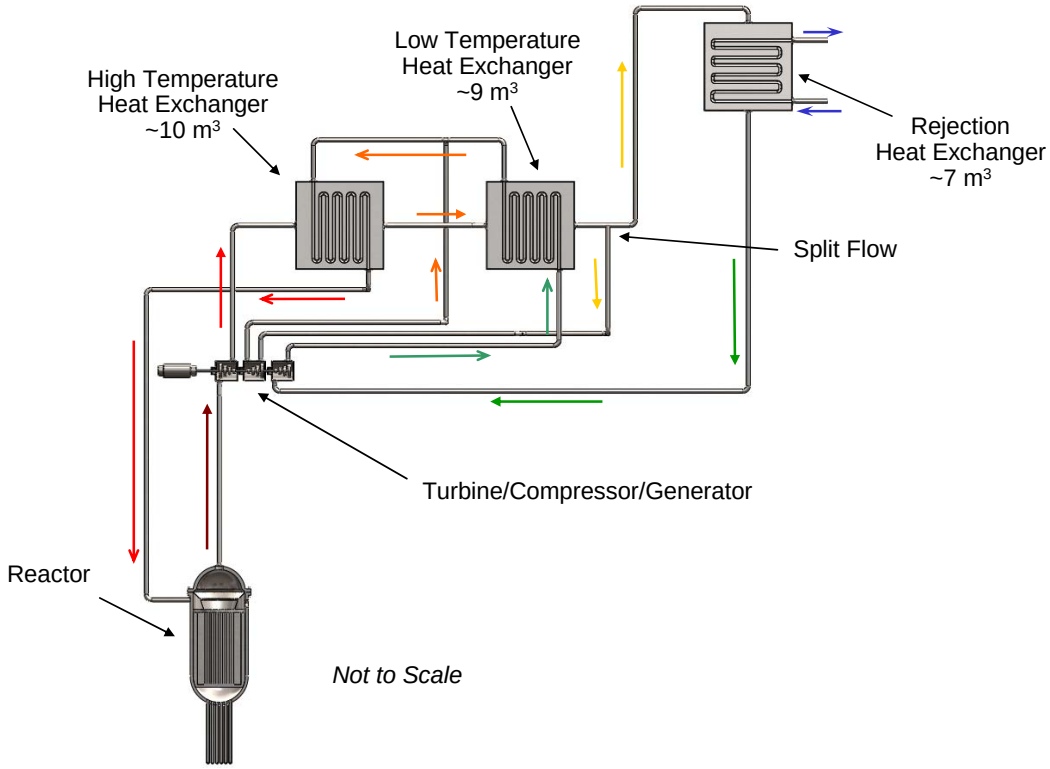

Figure 5. Conceptual Plant Layout for a S-CO_2 Power Conversion System With a Combined Turbine/Compressor/Generator Unit.

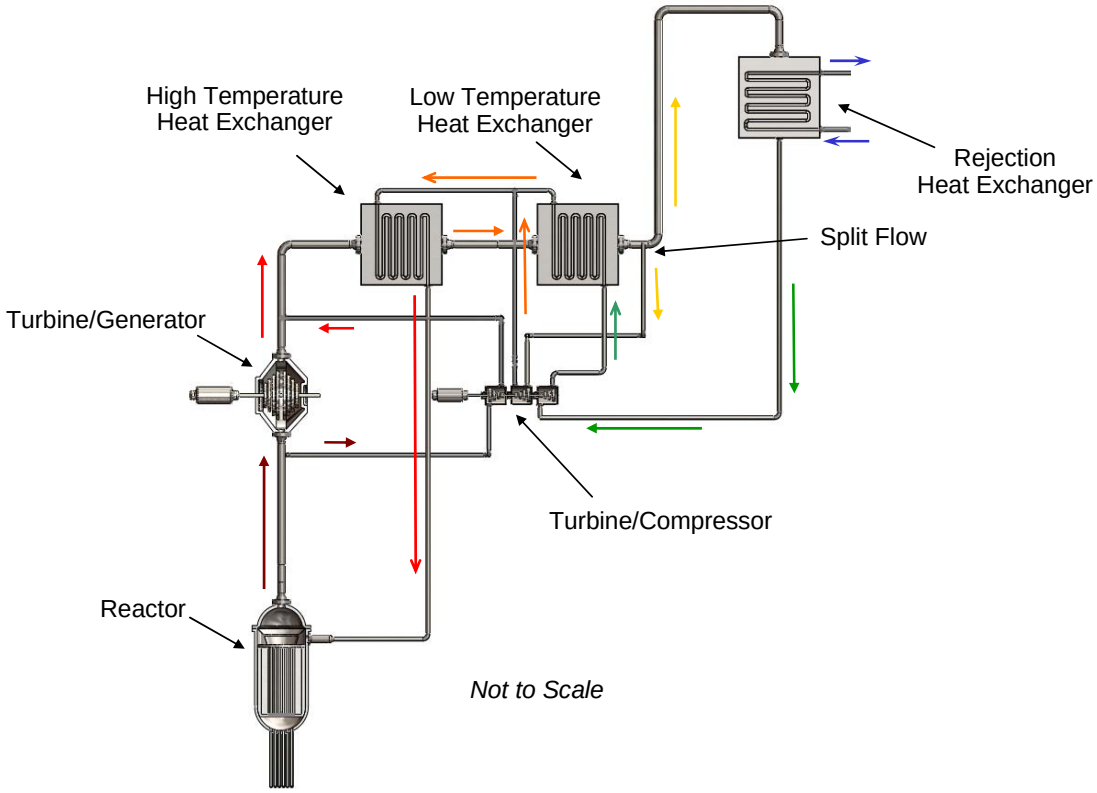

Figure 6. Conceptual Plant Layout for a S-CO_2 Power Conversion System With a Separate Turbine/Generator Unit and Combined Turbine/Compressor Unit.

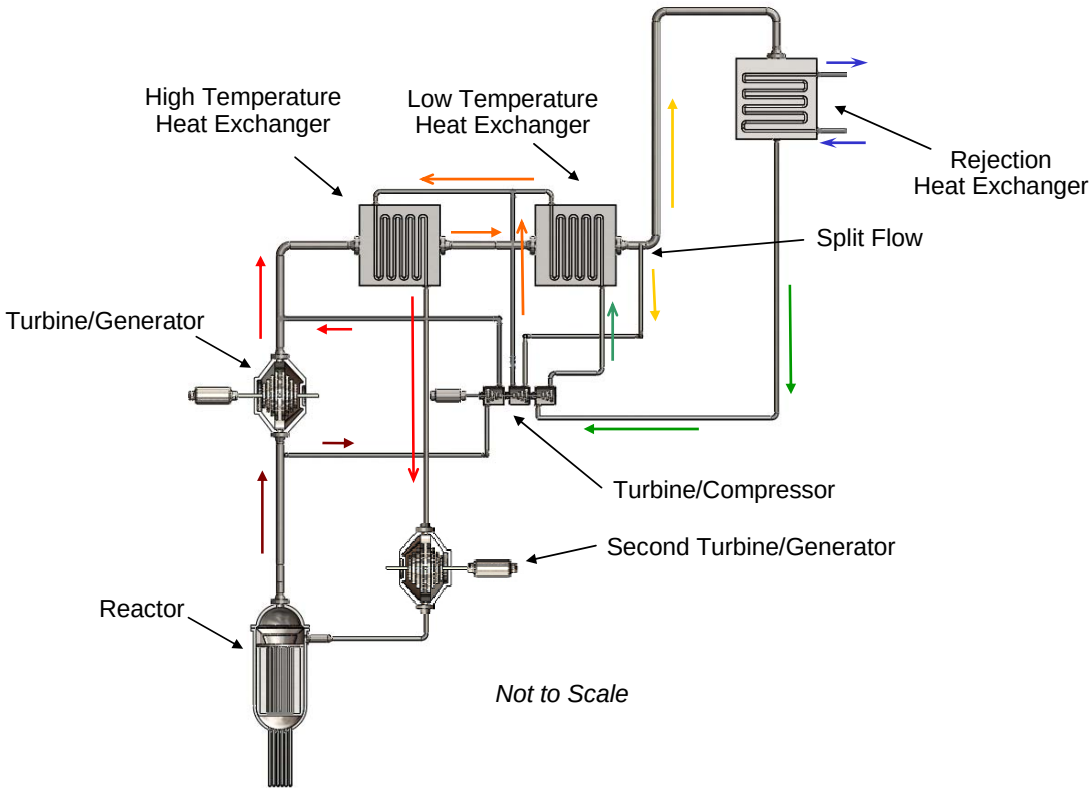

Figure 7. Conceptual Plant Layout for a S-CO_2 Power Conversion System With Two Separate Turbine/Generator Units and a Combined Turbine/Compressor Unit.

Figure 8 shows the same power conversion scheme as Figure 5 but with additional equipment that will be required for the system, including a pressurizer/accumulator and an emergency core cooling system. A pressurizer/accumulator will be required to maintain the system volume and pressure during startup, transients, and shutdown. It would most likely be placed in the cold leg of the system. More work is required to identify the performance features and volume requirements of the pressurizer/accumulator, as well as a CO_2 make-up, recovery, and purification system, which may be part of the unit. Also shown in the figure is an emergency core cooling system that would be required in the event of a LOCA. This system would probably be located near the inlet and outlet of the reactor vessel and may require both active and passive systems. Again, more work is required to establish the performance features of the system.

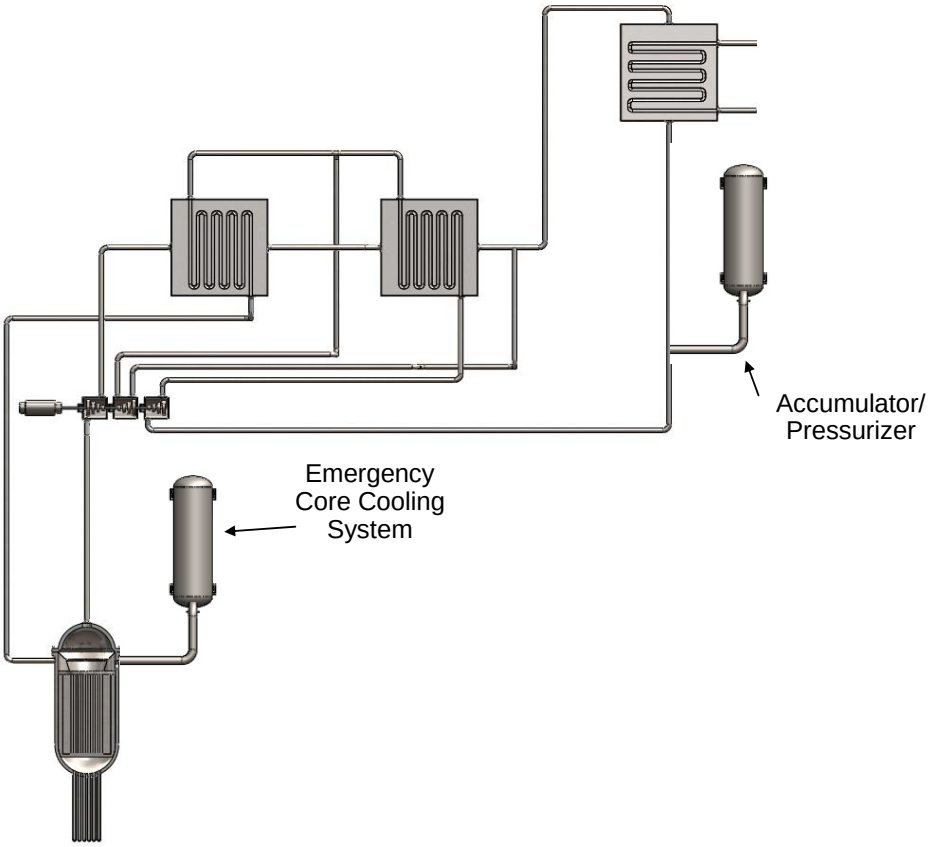

Figure 8. Conceptual Plant Layout for a S-CO_2 Power Conversion System Showing a Pressurizer/Accumulator and an Emergency Core Cooling System.

5. Heat Transfer and Thermal Hydraulic Analyses

Scoping analyses were performed using a simple, steady-state, Single-Channel Flow Analysis (SCFA) Mathcad code to parametrically consider the effects on the maximum fuel temperature, clad temperature, average pressure drop through the core, and resulting pumping power requirements as a function of the reactor power level, pin diameter, and coolant fraction. SCFA performs a one-dimensional (radial) heat transfer analysis on the fuel and cladding at the axial centerline of the core using the thermal conductivities of the fuel and cladding, along with a lumped parameter single-channel flow analysis using the coolant thermophysical properties. User input variables include the reactor power level, power density, active core height, total pin height including gas plenum, inlet and outlet coolant temperatures, and radial and axial peaking factors. The code calculates the core radius and total mass flow rate. The cladding thickness and gap thickness are fixed, and parametric analysis is performed using the pin diameter and coolant fraction for a triangular pitch. The Nusselt number, heat transfer coefficient, and friction factor are calculated parametrically to generate the fuel centerline temperature at the center of the reactor core, clad temperature at the center of the core, and average channel pressure drop. The pumping power is calculated for one channel and multiplied by the total number of fuel pins in the core to determine the total core pumping power. By analyzing the resulting plots, the appropriate pin diameter and coolant fraction can be determined such that the maximum fuel temperature, cladding temperature, pressure drop, and pumping power are maintained below prescribed limits. For the UO_2 fuel and stainless-steel cladding, the maximum operating temperatures are assumed to be less than 2200°C and 800°C, respectively. It is desired that the pumping power be less than 1% of the total core power.

Analyses are presented for a cladding thickness of 0.056 cm, gap of 0.008 cm, fuel pin active height of 1.6 m, fuel pin plenum of 1.0 m, core radius of 0.85 m, CO_2 pressure of 20 MPa, core inlet temperature of 450°C, and core outlet temperature of 650°C. Core radial and axial peaking factors were both set to 1.4, allowing for an overall peaking factor of 2.0. Both 200 MWth and 400 MWth are presented to determine the potential upper bound on reactor power.

Figures 9 through 12 show the 200 MWth parametric analyses for the core centerline fuel temperature, cladding temperature, average core pressure drop, and pumping power fraction. A smaller coolant fraction (cf) is desirable to allow for a higher core loading density, hence a smaller core. A smaller cf results in lower fuel and cladding temperatures as seen in Figures 9 and 10. However, a smaller cf results in a larger pressure drop and pumping power as seen in Figures 11 and 12. Likewise, a smaller diameter fuel pin results in lower fuel and cladding temperatures but also a larger pressure drop and pumping power.

Two points were selected as a possible range of desirable conditions: a 0.2 cf with a 0.75 cm diameter pin, and a 0.3 cf with a 1.2 cm diameter pin. Both of these values result in approximately the same fuel loading. The 0.2 cf case has a maximum fuel centerline temperature of 880°C, cladding temperature of 600°C, and pumping power fraction of 0.008 (0.8%). The 0.3 cf case has a maximum fuel temperature of 1440°C, cladding temperature of 690°C, and pumping power fraction of 0.001 (0.1%). Both of these cases are acceptable for 200 MWth, with the 0.2 cf case having a slightly greater margin in the fuel temperature and larger pressure drop.

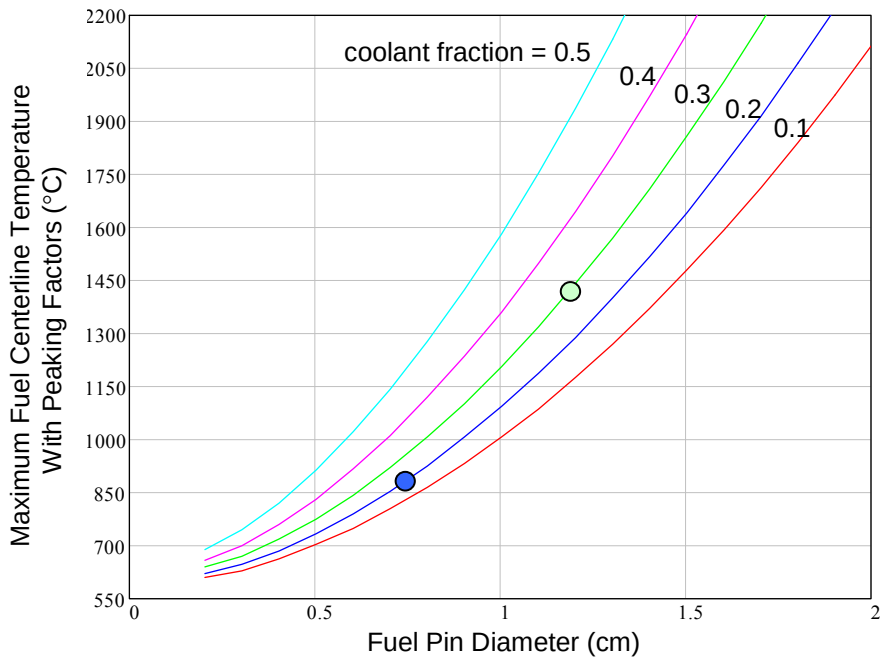

Figure 9. Fuel Centerline Temperature at the Center of the Core as a Function of the Fuel Pin Diameter and the Coolant Fraction for 200 MWth.

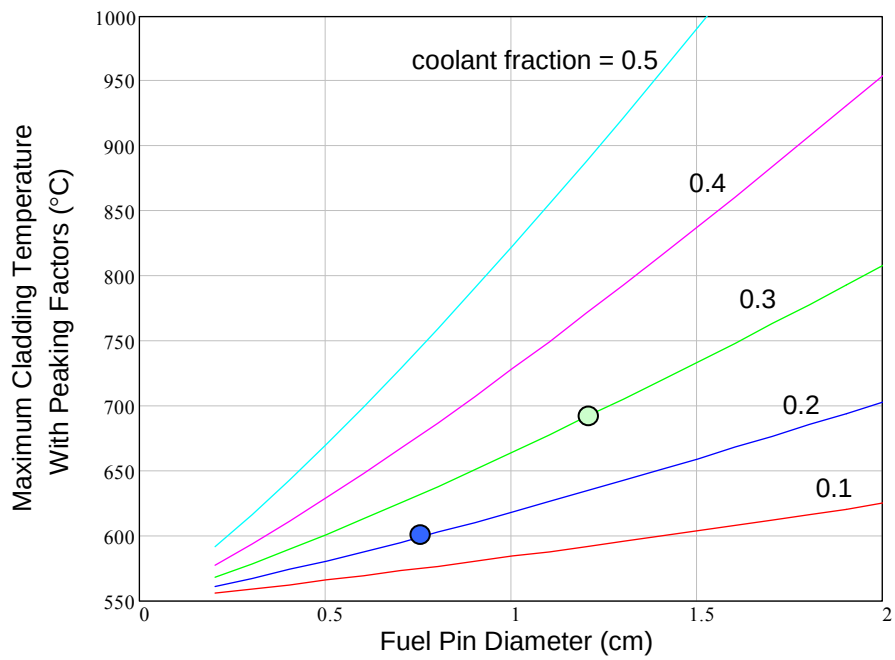

Figure 10. Cladding Temperature at the Center of the Core as a Function of the Fuel Pin Diameter and the Coolant Fraction for 200 MWth.

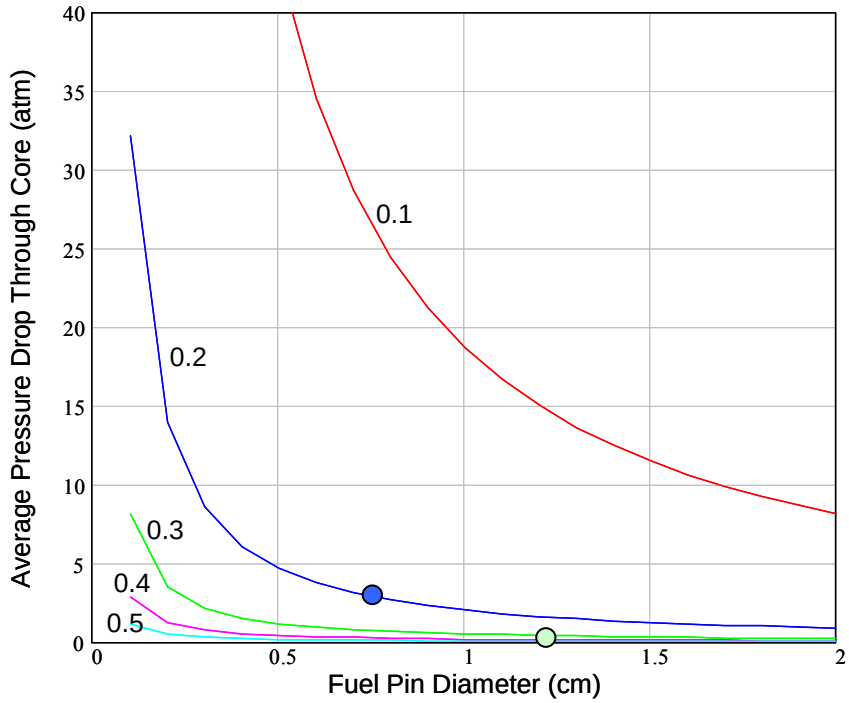

Figure 11. Average Core Pressure Drop as a Function of the Fuel Pin Diameter and the Coolant Fraction for 200 MWth.

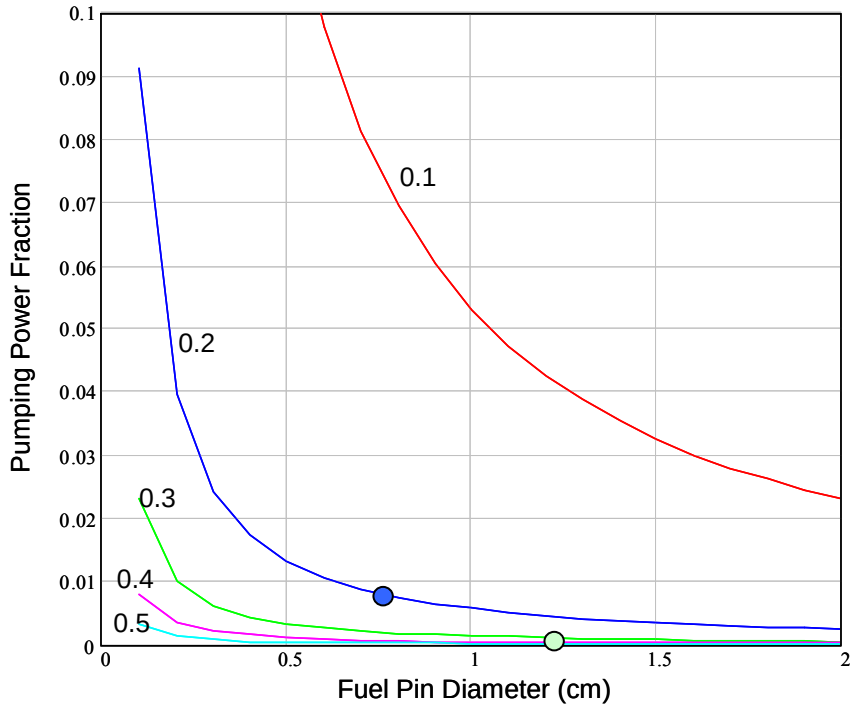

Figure 12. Pumping Power Fraction as a Function of the Fuel Pin Diameter and the Coolant Fraction for 200 MWth.

Figures 13 through 16 show the 400 MWth parametric analyses for the core centerline fuel temperature, cladding temperature, average core pressure drop, and pumping power fraction. Again, a smaller cf results in lower fuel and cladding temperatures as seen in Figures 13 and 14. However, a smaller cf results in a larger pressure drop and pumping power as seen in Figures 15 and 16. Likewise, a smaller diameter fuel pin results in lower fuel and cladding temperatures but a larger pressure drop and pumping power.

The same two points were selected as a possible range of desirable conditions: a 0.2 cf with a 0.75 cm diameter pin, and a 0.3 cf with a 1.2 cm diameter pin. The 0.2 cf case has a maximum fuel temperature of 1190°C, cladding temperature of 610°C, and pumping power fraction of 0.028 (2.8%). The 0.3 cf case has a maximum fuel temperature of 2220°C, cladding temperature of 725°C, and pumping power fraction of 0.004 (0.4%). These cases are more interesting to evaluate. The 0.2 cf is unacceptable because the pumping power fraction exceeds the desired value of 0.01 (1%) by a factor of 2.8. However, the 0.3 cf case exceeds the maximum fuel and cladding temperatures. A compromise condition of 0.25 cf and a pin diameter of 1.1 cm would allow for a 400 MWth core, for which the pumping power and fuel and clad temperatures remain below the acceptable limits.

In conclusion, the thermal hydraulic scoping analysis shows that an acceptable fuel pin and coolant fraction can be found for the reactor operating at both 200 MWth and 400 MWth with a core radius of 0.85 m, core height of 1.6 m, pin plenum height of 1.0 m, CO_2 pressure of 20 MPa, core inlet temperature of 450°C, core outlet temperature of 650°C, and core radial and axial peaking factors of 1.4.

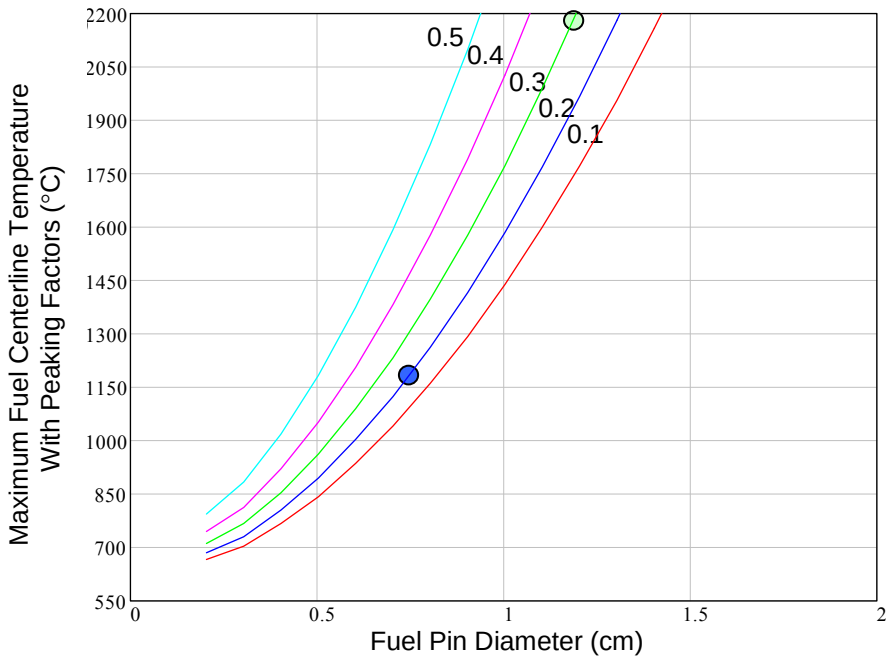

Figure 13. Fuel Centerline Temperature at the Center of the Core as a Function of the Fuel Pin Diameter and the Coolant Fraction for 400 MWth.

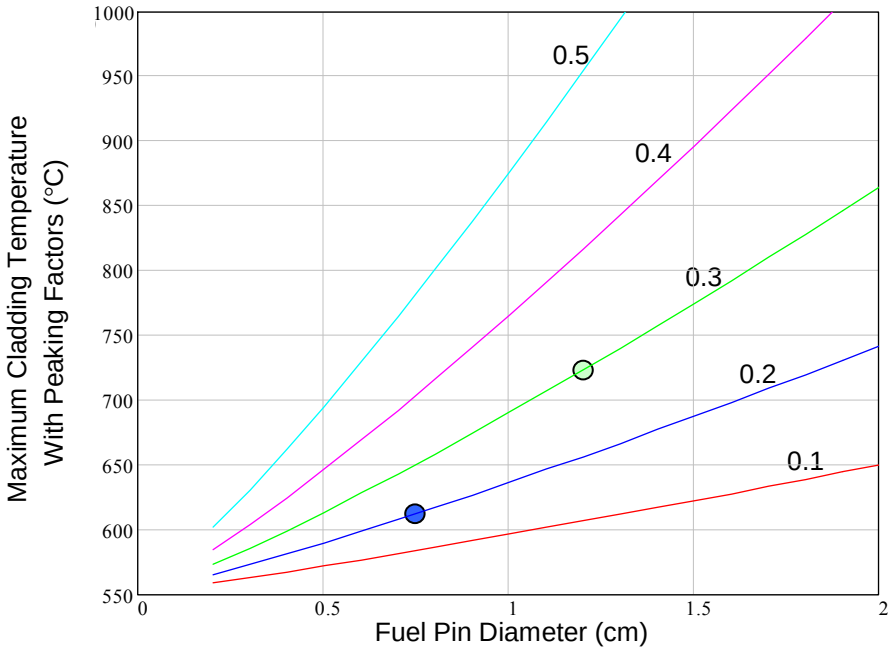

Figure 14. Cladding Temperature at the Center of the Core as a Function of the Fuel Pin Diameter and the Coolant Fraction for 400 MWth.

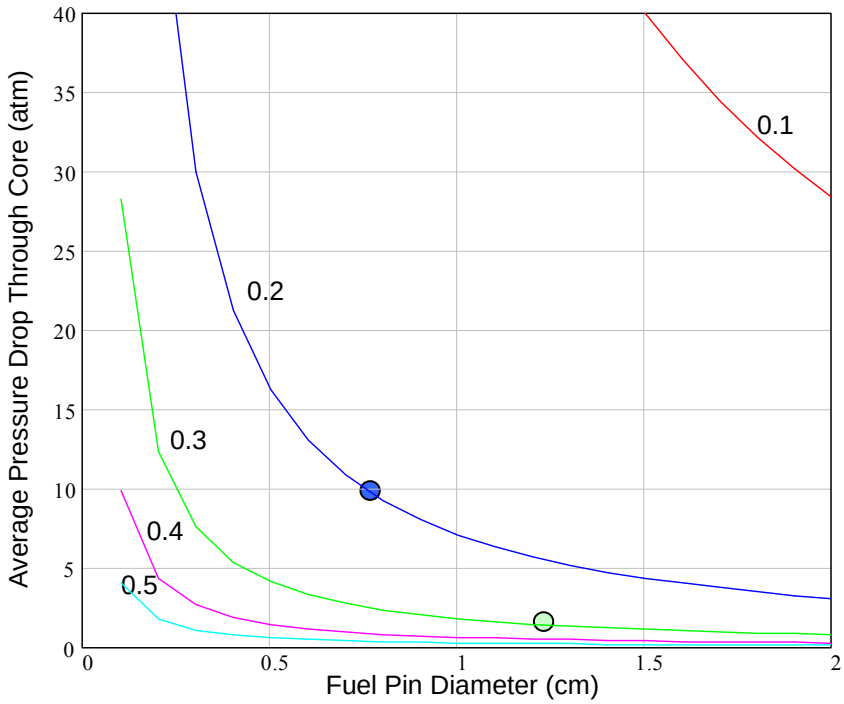

Figure 15. Average Core Pressure Drop as a Function of the Fuel Pin Diameter and the Coolant Fraction for 400 MWth.

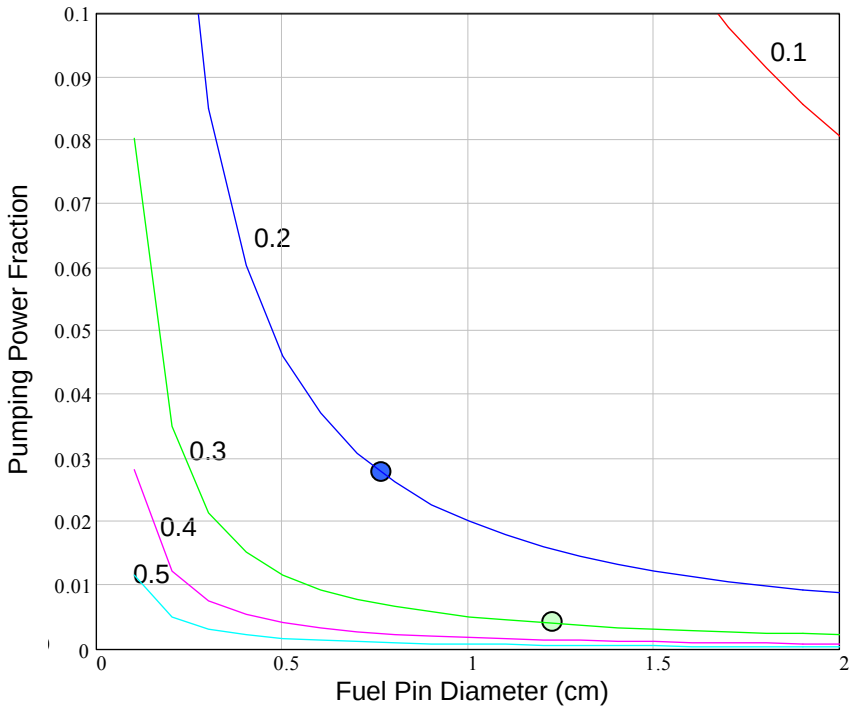

Figure 16. Pumping Power Fraction as a Function of the Fuel Pin Diameter and the Coolant Fraction for 400 MWth.

6. Neutronic Analysis

Neutronic analyses were performed using the Monte Carlo N-Particle code Version 5 (MCNP, 2003) and the ENDF/B-VI cross sections included in the distribution. K effective (k_{eff}), burnup, and void reactivity worth calculations were performed using the three-dimensional model of the core shown in Figure 3 and the baseline parameters in Tables 3 and 4. The calculations were performed with the fuel cross sections at 1200 K. Scoping k_{eff} and burnup analyses were iterated with the flow analysis to determine the appropriate fuel-pin size, coolant fraction, enrichment, reactor size, and reflector thickness. The goal was to determine if a reasonably-sized reactor could be configured that would have only a small change in reactivity over the desired 20-year operating life as the U-235 is consumed and the fissile Pu-239 is produced from the neutron absorption of U-238.

6.1 Burnup Analyses

Burnup analyses were performed using the burnup code BURNCAL (Parma, 2002). BURNCAL uses MCNP to perform the neutronic analysis and calculates the fuel inventory, including the fission product, activation product, and transuranic inventory, using the MCNP tally results. Calculations are performed for a time-at-power history defined in an input file. It is important to note that the results presented represent scoping calculations performed using a single reactor zone. Thus the inventory calculated over the time history represents the core average values and not the true three-dimensional representation. Future calculations will include zoned core configurations to more accurately represent the core inventory.

In order to determine the appropriate enrichment range for a long-life reactor core, initial burnup calculations were performed for an infinite reactor system. To perform this analysis, a fuel pin was modeled with coolant in a hexagonal geometry. Specular reflector boundary conditions were identified in the MCNP model that allow for the simulation of an infinite reactor. UO_2 fuel was used with stainless-steel cladding. CO_2 coolant at a pressure of 20 MPa and 0.2 cf was modeled. The results are shown in Figure 17 for an operating history of seventeen years and a power density of 50 MWth/MTU. Calculations were performed for enrichments of 5%, 10%, 12.5%, and 20%. The results show that a long-term burnup is achievable for an infinite reactor with an enrichment of greater than 5%. An infinite reactor with an enrichment of 5% cannot be made critical. At an enrichment of 12.5%, the infinite reactor value of the multiplication constant stays almost constant over the entire seventeen years of burnup. The initial value of k-infinity is about 1.25, which allows for some leakage margin in a finite reactor configuration. The question is whether a finite reactor can be configured with a reasonable size and for ~12.5% enriched fuel. A core could be configured with a higher enrichment value, up to 20%. However, many dollars of negative reactivity would be required in the form of burnable poisons, control rods, or other removable poisons to maintain a critical condition over the operating history.

Figure 18 shows the burnup results for a 200 MWth, 12.5% enriched UO_2 fuel pin, 0.75 cm in diameter, 0.2 cf, core radius of 0.75 m, active fuel height of 1.4 m, and a Ni reflector 25 cm in thickness. The core was sized to have an initial k_{eff} value of greater than one. 250,000 particle histories were run for each time point resulting in a statistical uncertainty in k_{eff} of ~0.001 (~$0.12 of reactivity). The error bars are included within each data point.

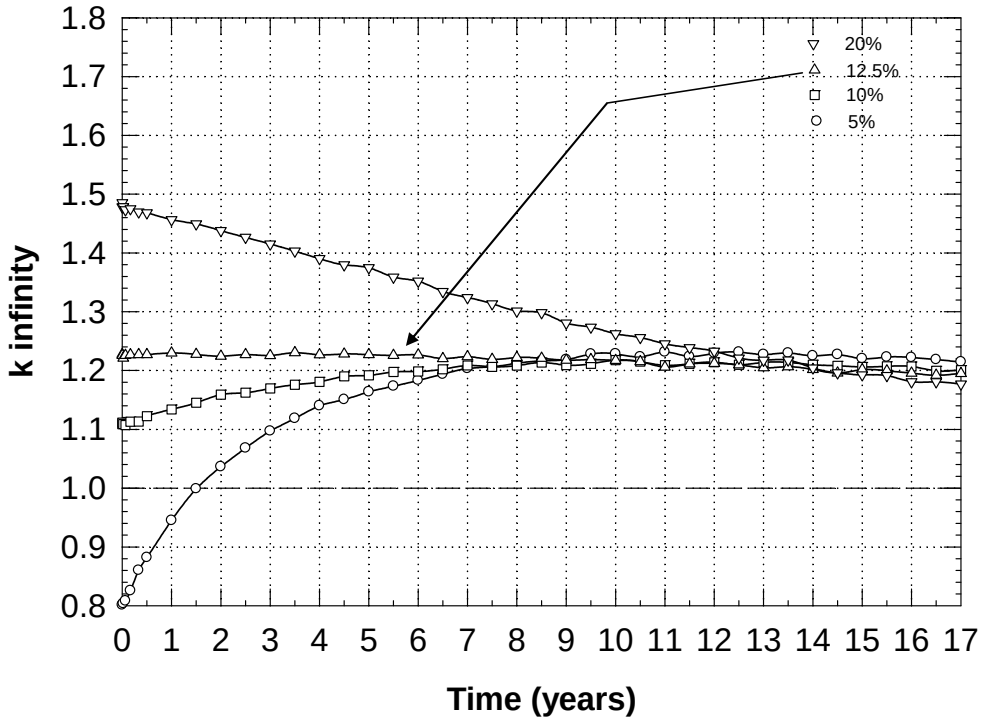

Figure 17. K-infinity as a Function of Operating History for a Power Density of 50 MW/MT, 20% CO_2 Coolant, and Different Enrichments of UO_2.

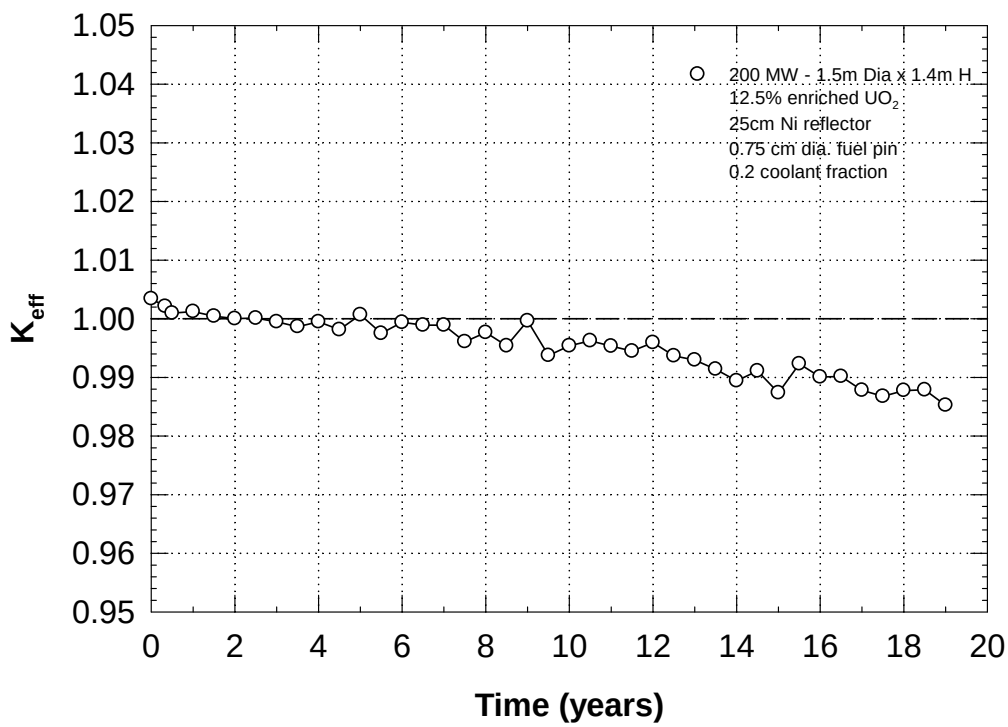

Figure 18. k_{eff} as a Function of Operating History at 200 MW and 12.5% Enrichment.

The results show that k_{eff} decreases somewhat (by ~0.02 or ~$2.50) over the operating history, indicating that there is insufficient conversion of the fertile U-238 to Pu-239 to maintain a critical condition. Lowering the enrichment slightly and increasing the core size to increase k_{eff} may allow for a more constant k_{eff} versus power history.

Figure 19 shows the burnup results for a 200 MWth, 12.0% enriched core, a decrease in the enrichment by 0.5%. The core size was increased to a core radius of 0.85 m, active fuel height of 1.6 m, and a Ni reflector of 15 cm in thickness, to allow for an initial k_{eff} value greater than one. The results now show that k_{eff} increases slightly (by ~0.008 or ~$1.00) over the operating history and appears to plateau near the end of life. Figure 20 shows the fuel constituent inventory over the same operating history. Over the 20-year history, at a power density of 10 MW/MT (200 MWth), the U-235 density decreases from about 1 g/cc to 0.5 g/cc. The U-238 is consumed as it is converted to Pu-239. At the end of 20 years, the Pu-239 is ~0.5 g/cc. For this reactor type, the optimum conversion is about 0.8 atoms of Pu-239 per atom of U-235 consumed at the beginning of life. A one-to-one conversion is not required since Pu-239 has a high value of η (neutrons emitted per absorption) compared to U-235.

Since the power density for this core is about 10 MW/MTU and the k-infinity calculations were performed at 50 MWD/MTU, a higher power density (20 MW/MTU or 400 MWth) was also analyzed to determine if it was a feasible scenario. Figure 21 shows the results of this analysis using the same conditions as for Figure 19 but at a power level of 400 MWth. The results show that k_{eff} increases over the first 10 years of operation, to a value of about 1.01, and then decreases over the next 10 years to a value close to the starting value. This indicates that a 400 MWth – 20-year cycle could be feasible from a reactivity point of view.

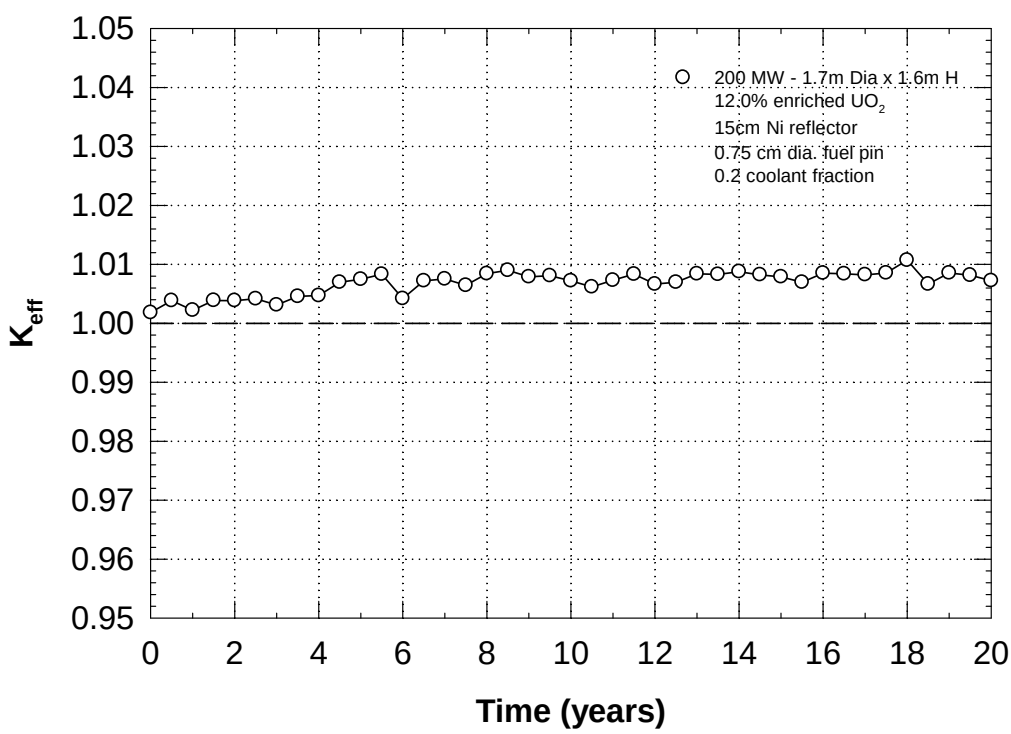

Figure 19. k_{eff} as a Function of Operating History at 200 MW and 12.0% Enrichment.

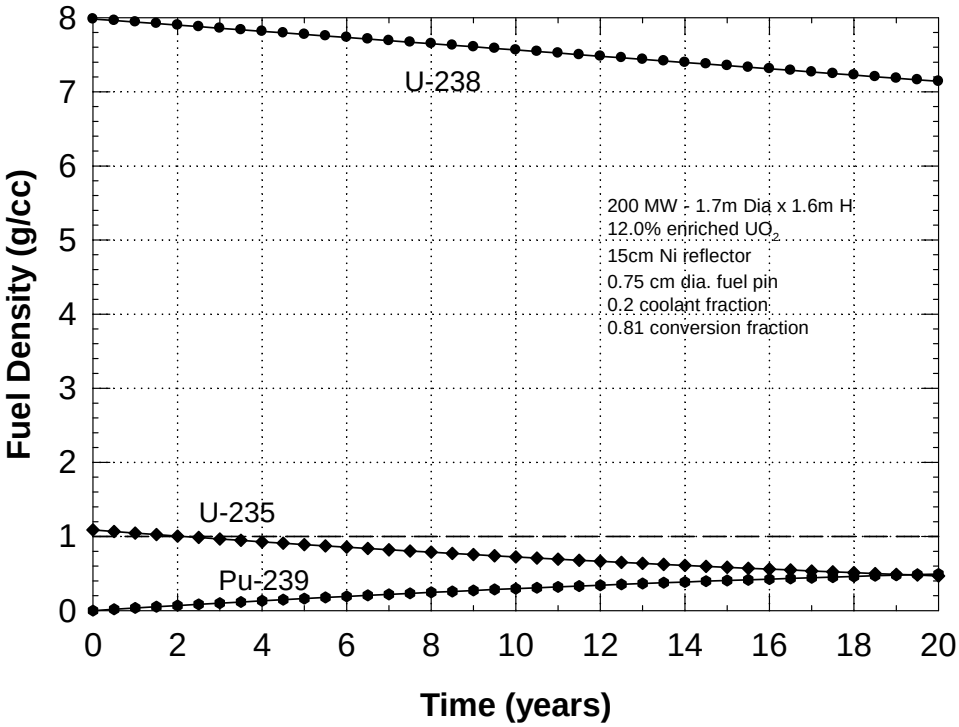

Figure 20. Fuel Constituent Density as a Function of Operating History at 200 MW and 12.0% Enrichment.

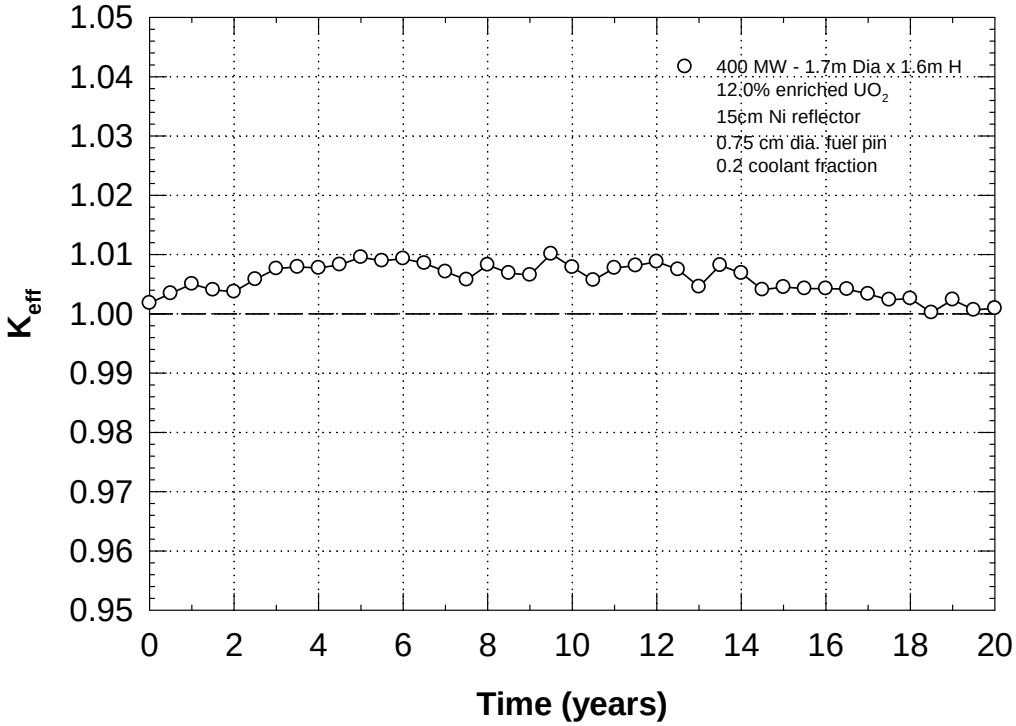

Figure 21. k_{eff} as a Function of Operating History at 400 MW and 12.0% Enrichment.

Other burnup analyses were performed using the same basic reactor geometry but with a pin diameter of 1.20 cm and a 0.3 cf. This configuration has approximately the same fuel density as for the 0.75 cm pin diameter and 0.2 cf. The results are not included in this report but are found to be very similar to those presented in Figures 19 to 21.

The results of the burnup analyses show that a long-life core can be made feasible with an initial core loading of 12% enriched UO_2. The reactivity change over the lifetime can be maintained within ~$1.00 of reactivity. Further analyses are required to more accurately maintain the three-dimensional inventory by zone loading the core.

6.2 Delayed Neutron Fraction and Void Worth

The burnup calculations were performed with the reactor temperature at conditions representing full power operation. Additional MCNP calculations were performed at cold, room temperature conditions to determine the startup reactivity requirements. The cold reactor k_{eff} was found to be larger than the hot condition by $\Delta k/k = 0.00956 \pm 0.0015$, or about +$1.20 of reactivity. Further, the neutron lifetime of the reactor was found to be 41.4 μs. Future work will include an in-depth analysis to determine the temperature coefficient of reactivity for the fuel and coolant.

MCNP calculations were also performed to determine the delayed neutron fraction (β) and the void reactivity worth of the core over the core lifetime. The results are shown in Table 5. The values for β are based on the U-235, U-238, and Pu-238 values given in Lewins (1978) for fast fission. The results were determined by multiplying each fissile constituent β by the normalized fraction of fissions for that constituent. The β results do not include the effectiveness factor, which would increase the values somewhat, due to the importance of the delayed neutrons at the center region of the core. Future work will include a determination of the effectiveness factor for the core. A beginning of life (BOL) value for β of 0.008 is calculated. This value is greater than the U-235 value due to a relatively large number of U-238 fissions that occur in a fast reactor. This value decreases over the lifetime of the core due to the increased inventory of Pu-239. The end of life (EOL) value for β of 0.0062 for 200 MWth and 0.0052 for 400 MWth is calculated. These values are significantly smaller than for the BOL β value.

Void reactivity worth was calculated by changing the density of the CO_2 coolant in the reactor from a value at 20 MPa (3000 psia) to a value at 0.1 MPa (14.7 psia), a factor of 200. The results are shown in Table 5 for the 0.75 cm pin 0.2 cf case, and for the 1.20 cm pin 0.3 cf case. Calculations were performed for BOL and EOL for 200 MWth and 400 MWth. Although there is significant uncertainty in the results (±~$0.25), they indicate that the voiding is positive but small – less than +$1.00. Since the only way that voiding could occur in the reactor core is by a depressurization of the system, this reactivity effect is considered to be manageable by inserting the control and safety rods. It would be expected to take several minutes to depressurize the core in the event of a small pipe break. Smaller reactor configurations have been shown to have negative void reactivity worth, due to the higher neutron leakage from the core. Future work will include more accurate analysis of the void worth. Further accident analysis will provide more information on the effects of having a positive void worth on the safety of the system.

Table 5. Delayed Neutron Fraction and Void Worth Calculated Using MCNP.

From Lewins – Fast Fission - β(U-235) = 0.0066
β(U-238) = 0.0161
β(Pu-239) = 0.00212

Estimates for the delayed neutron fraction β
β BOL = 0.0080
β EOL (200 MW @ 20 yrs) = 0.0062
β EOL (400 MW @ 20 yrs) = 0.0052

<u>Case 1 - 0.75 cm OD, cf = 0.2</u>

Δk/k (BOL) = 0.00185 ± 0.0016	ρ = +\$0.23 ± \$0.20
Δk/k (200MW@20yrs) = 0.0020 ± 0.0015	ρ = +\$0.32 ± \$0.24
Δk/k (400MW@20yrs) = 0.0015 ± 0.0013	ρ = +\$0.29 ± \$0.25

<u>Case 2 - 1.20 cm OD, cf = 0.3</u>

Δk/k (BOL) = 0.00056 ± 0.0015	ρ = +\$0.07 ± \$0.19
Δk/k (200MW@20yrs) = 0.0057 ± 0.0015	ρ = +\$0.92 ± \$0.24
Δk/k (400MW@20yrs) = 0.0040 ± 0.0014	ρ = +\$0.76 ± \$0.27

7. Natural Circulation Flow and Decay Heat Removal

One of the most difficult design and safety issues for a gas reactor is the loss-of-flow accident, which is an anticipated event. A loss-of-flow condition can occur due to a loss of power to the recirculating compressor or compressors, a fault such as a seizure in the compressor shaft bearings, or some other fault that disrupts the compressor operation. It is assumed that for this anticipated event, the plant protection system would shut down the reactor down by inserting control and safety rods. The pump coast-down and inertial flow of the coolant gas may allow for the removal of a significant fraction of the heat generated during the ~80-second reactor shutdown period. However, the decay heat generated due to the fission product decay must be continually removed to keep the reactor fuel and cladding from overheating and melting. Cooling the core can be achieved by a separate active auxiliary decay heat removal system or by removal of heat from the pressure vessel by conduction through the core structure. For a thermal gas reactor, the moderating material may allow for significant heat capacity and heat removal by conduction through the core radially to the boundary; however, the operating power density of the core must be limited such that the decay power can be removed without overheating the fuel. For a gas fast reactor, such as the SC-GFR concept, the moderating materials are non-existent and the structural materials are insufficient for conductive heat transfer to the boundary.

Decay heat removal using natural circulation flow is achievable in water reactor systems, since a relatively large pressure head can be developed for a small change in temperature from the cold leg to the hot leg. It will be shown in this section of the report that natural circulation flow can also be established in an S-CO_2 system, if the system remains pressurized.

SNL has been operating two S-CO_2 loops since 2008. Occasionally, very large flow rates have been observed well after termination of the experiment, even after all heaters and turbomachinery were shutdown. Because there were large temperature gradients still remaining in these loops, the flow has been attributed to a combination of condensation effects and natural circulation. These early observations prompted SNL to examine the natural circulation capability of S-CO_2 by examining the Grashof number (Gr), which is a dimensionless number providing the buoyancy-to-viscous force ratio,

$$Gr = \frac{g\,\beta\,\rho^2\,Lc^3\,\Delta T}{\mu^2}$$

where g is the gravitational constant, β is the coefficient of thermal expansion, ρ is the density, Lc is the characteristic length, ΔT is the temperature difference (wall to bulk), and μ is the viscosity. A comparison of Gr for CO_2 and water is illustrated in Table 6, for g = 9.81 m/s^2, Lc = 1 m, and ΔT = 10°C. The table shows that at room temperature (300 K) and 7.69 MPa, Gr is 5.9e14 for CO_2 and 7.5e10 for water. These values were calculated at temperatures and pressures near the operating conditions for the CO_2 in the waste heat-rejection system of the cycle. For reference, these conditions are near the critical point of CO_2 that occurs at 7.37 MPa and 304.1 K. This four order of magnitude increase in Gr for CO_2 compared to water provides clear evidence of the tremendous potential for natural circulation flow and decay heat removal in S-CO_2-cooled reactor systems. This would apply to both direct and indirect cycles.

Table 6. Comparison of the Grashof Number for S-CO_2 and Water.

Parameter	Units	CO_2	Water
Bulk Temperature	K	300	300
Pressure	MPa	7.69	7.69
β	1/K	0.039	0.00037
ρ	kg/m^3	275.6	996.7
μ	Pa-s	2.21e-5	6.94e-4
Gr		5.9e14	7.5e10

Since decay heat removal in GFRs has been a performance-limiting component in the reactor design, the use of a passive mechanism, such as natural circulation, could greatly improve the reliability and performance of S-CO_2-cooled GFRs. To evaluate these capabilities further, SNL has begun to develop a series of tools and validation experiments that can be used to define the limits, constraints, and characteristics for S-CO_2 natural circulation flow.

The evaluation of natural circulation in S-CO_2 flow in power systems is being implemented at SNL in three tasks. First, a simple lumped-parameter Excel spreadsheet model has been developed and is being used for scoping calculations. Second, a computational code effort has begun that will be used to support a more detailed design of the S-CO_2 power conversion loop. Third, an experimental effort is being planned that will be used to verify and validate the code development effort. The experimental program requires making minor modifications to the existing SNL S-CO_2 research loop. The testing and modifications are scheduled for the second half of fiscal year 2011.

The first task has already been implemented. A simple natural circulation model that can be used to guide a number of scoping calculations has been developed. This steady-state lumped-parameter supercritical natural circulation flow analysis simulator is implemented using Excel Visual Basic and the NIST REFPROP (NIST, 2007) database, and is based on a published report presented in the S-CO_2 power cycle symposium by Milone (2009). The results of this model were compared to the published S-CO_2 natural circulation experiments and show similar trends and reasonable agreement. The lumped-parameter code was also extended to include counter-flow heat exchangers. The results indicate that the S-CO_2 Brayton cycle loop can be used as an effective way to remove decay heat, provided that the heat source and heat sink elevations are properly located to take advantage of the gravity pressure head and that the pressure remains near the critical pressure.

The second task includes the development of a more sophisticated natural circulation model using computational fluid dynamics models. This code is based on an existing SNL Fire Code, C3D (Greiner, et al., 2004, Chalasani, et al., 2010). C3D was developed for fire modeling, where buoyancy effects (due to density changes) of factors of three to four are common. The code uses multidimensional flow with natural circulation models. The code has been modified to use enthalpy-based solutions using the NIST REFPROP (NIST, 2007) equation-of-state models

for S-CO_2 or other fluids. The modified code is named C3D-SC. Previous experience has shown that supercritical fluid codes require an enthalpy-based solution model when operating near the critical point. The enthalpy-based solution is required to account for the large enthalpy changes that can occur near the pseudo-critical point of a fluid, along with large density and viscosity changes that occur. Future efforts will use the code to perform dynamic simulations on the reactor and power conversion system, including transitioning from forced-flow circulation to natural circulation, as well as natural circulation modeling in the reactor, recuperators, and heat exchanger. This code will also be one of the primary tools used to model the dynamic behavior of natural circulation in the SNL S-CO_2 natural circulation experiments planned in the second phase of this project.

The third task will be to modify the existing S-CO_2 research loop at SNL to provide a natural circulation loop that can be used to validate the steady-state and transient behavior of supercritical natural circulation at 20-50 kWth. The transition from forced-flow cooling to natural circulation can also be explored since the loop has a high-speed compressor installed. The tests will measure the actual flow rate, fluid densities, and temperatures as a function of time at various initial fill densities, pressures, heat load, and degree of subcooling in the heat rejection system. The tests results will be used to validate the natural circulation codes and tools.

7.1 Lumped-Parameter Natural Circulation Analyses

The lumped-parameter model assumes a vertical hot and cold leg, and balances the frictional pressure drop through the loop against the pressure head created by the density difference due to the temperature difference between the hot and cold legs. Because of its simplicity, the model generally over-estimates the flow rate because it neglects other real pressure drop effects. Figure 22 provides four images that show the results and some of the major characteristics of natural circulation in S-CO_2. Image D illustrates the natural circulation test loop using a 500 W heater to produce natural circulation in a 76" tall loop with 0.45" internal diameter stainless-steel tubing (Milone, 2009). Image A shows the calculated temperature difference between the hot leg and the cold leg as a function of cold leg temperature for various pressures within the loop (Milone, 2009). It is important to note that near the pseudo critical point the hot-to-cold leg temperature difference nearly vanishes for all pressures shown. This occurs because, near the pseudo-critical point, the heat capacity of the fluid has a large spike (up to a factor of 30 increase) due to the heat-of-vaporization-like effects that occur near the critical point. These effects reproduce heat of vaporization/condensation-like properties within a single-phase fluid. Image B shows the SNL lumped-parameter results at 1100 psia. The results agree with the Milone (2009) model. Image C presents actual measured data for this same configuration. The simple lumped-parameter model clearly shows the same trends, but the experimental ΔT's are larger than those that the simple model predicts. These results are provided to show that the simple lumped-parameter model basically captures the correct behavior of natural circulation for S-CO_2, but due to its simplicity, such as ignoring horizontal sections of piping and turns, and finite heater and gas chiller lengths, it does not capture all the characteristics of the geometry within the loop. A more sophisticated computational flow dynamics model used by Milone (2009) does agree well with the data however.

The proposed SC-GFR design assumes that the supercritical CO_2 Brayton cycle system is directly connected to the reactor, as illustrated in Figure 23. There is an elevation difference

between the reactor, the turbomachinery, recuperators, and the ultimate waste heat sink. This elevation difference can be optimized into the design to provide natural circulation flow with the turbomachinery not operating. At this time, the exact layout of the power conversion system and the choice of the type of S-CO_2 power cycle are still being investigated. The most likely configuration would require an inventory control volume, pressurizer, or accumulator to keep the complete loop within a specified pressure range during shutdown.

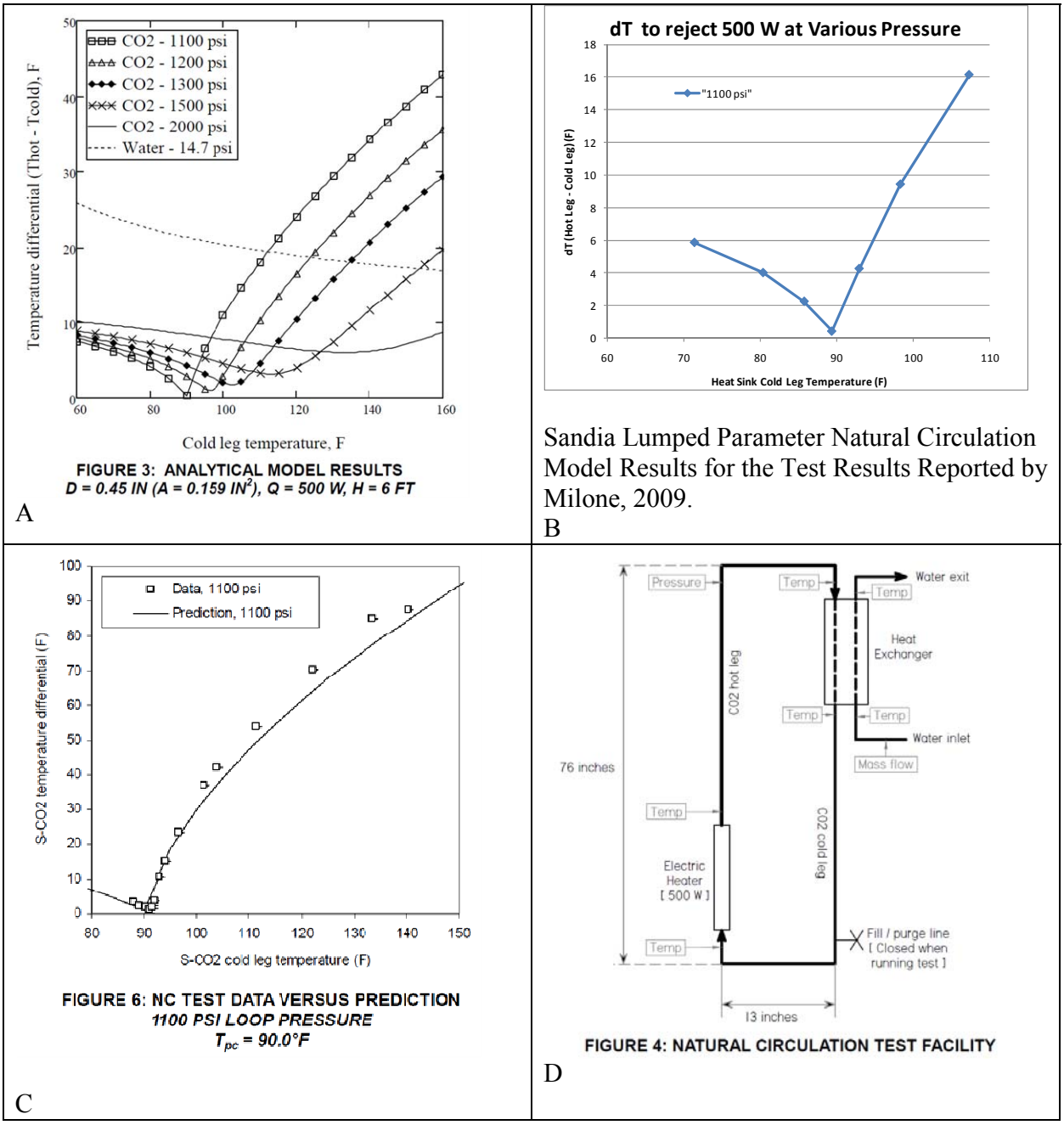

Figure 22. Natural Circulation Results From Milone Compared With the SNL Lumped-Parameter Model.

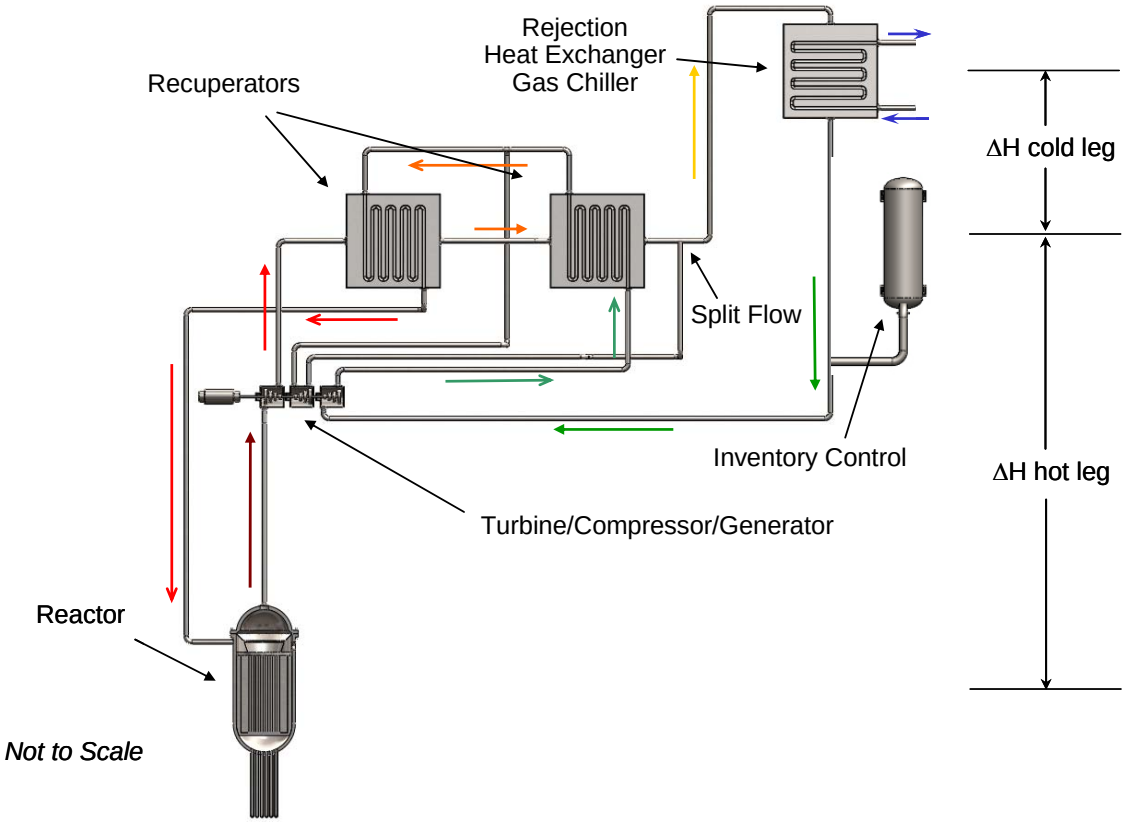

Figure 23. Natural Circulation Flow With Hot and Cold Legs.

The SNL lumped-parameter natural circulation model can include recuperators in the analysis. The recuperator could potentially disrupt the driving force for natural circulation because it transfers a significant amount of heat from one leg to the other, and the temperature differentials for natural circulation flow are small. A schematic of the model is illustrated in Figure 24. The recuperator is located at an elevation above the reactor but below the ultimate waste heat rejection system. In the model presented, the recuperator is 10 m above the reactor, and the heat-rejection heat exchanger or gas chiller is 20 m above the recuperator. The ducting is sized for full-power operation, with a mass flow rate of 1460 kg/s, and is about 0.6 m in diameter. The heat source was nominally set at 2.8 MWth, which is 1.4 % of the full thermal power in a 200 MWth reactor. For this analysis, the pressure was 7.58 MPa and the cold leg temperature was 315K (42°C). The natural circulation flow rate for this problem is about 122 kg/s or about 8.3% of the full power flow. The recuperator does a good job of keeping the hot side of the loop hot while the cold side remains cool, but there is still sufficient density difference to drive natural circulation. In this example, the peak exit temperature from the reactor is 559K while the cold leg fluid temperature exiting the gas chiller is 315K. This model clearly shows that the natural circulation is quite effective, even when there is a large gas-to-gas recuperator between the reactor and the gas chiller. Additional results show that effective flow can be maintained for different heat rejection temperatures and different elevations.

Other configurations can allow for much lower elevations if the recuperator is bypassed on one leg. However, to have this design feature, active valves would be required to initiate the bypass when the turbomachinery was not operating or the compressor pressure dropped below a prescribed level.

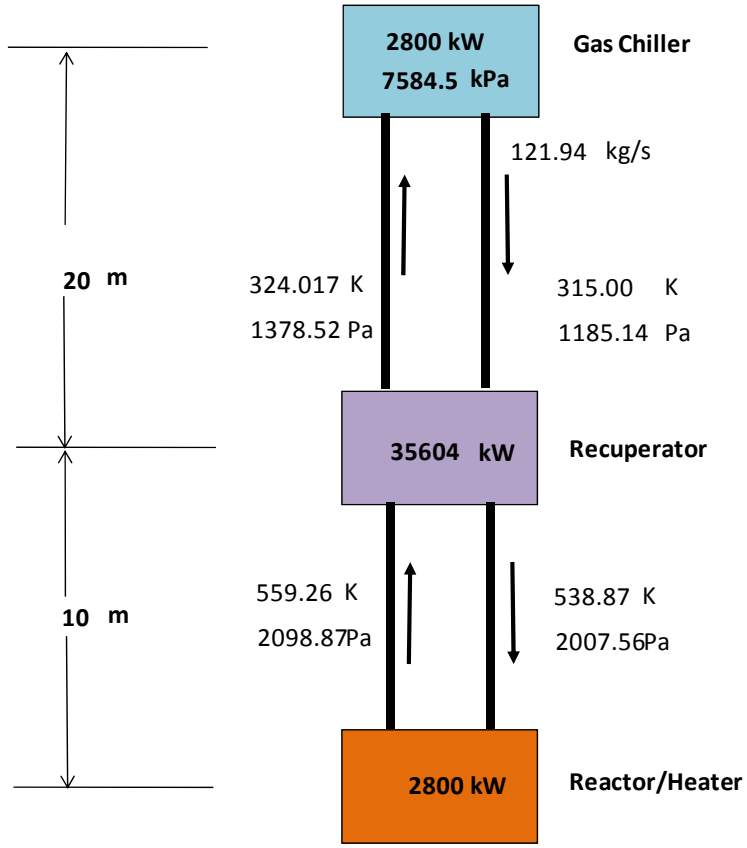

Figure 24. Lumped-Parameter Model of Natural Circulation Flow Through the Heat Rejection Heat Exchanger, Recuperator, and Reactor.

7.2 Three-Dimensional CD3-SC Natural Circulation Analyses

The three-dimensional fluid dynamics code C3D-SC was applied to reactor concepts similar to the lumped-parameter model to test its ability to model complex reactor systems. The full three dimensional model will allow investigation of both dimensional and transient effects. More complex models will be developed and tested in the future. For the current study with limited resources, only a simple reactor/heat exchanger operating in independent loops was demonstrated. Figure 25 shows the three dimensional model that was analyzed. The secondary loop connects the heat exchanger by 0.6 m OD piping to the gas chiller which is located 10 m above the intermediate heat exchanger. The cutaway view shows the mesh/grid structure, reactor internals, and simple flat-plate intermediate heat exchanger. In this model, the intermediate heat exchanger is situated 10 m above the bottom of the reactor core, and the final waste heat exchanger is 20 m above the core. There are two independent gas loops with CO_2 at different pressures. The main reactor loop is at 20 MPa, and the heat sink loop is at 7 MPa, which is near the critical point of the CO_2. Both loops are driven only by natural circulation and do not exchange any fluid. The results to-date show that two independent CO_2 flow loops can be energy-coupled in this way and driven entirely natural circulation. It is important to note that the actual loop circulates all the fluid through one loop, which is currently being implemented in the code.

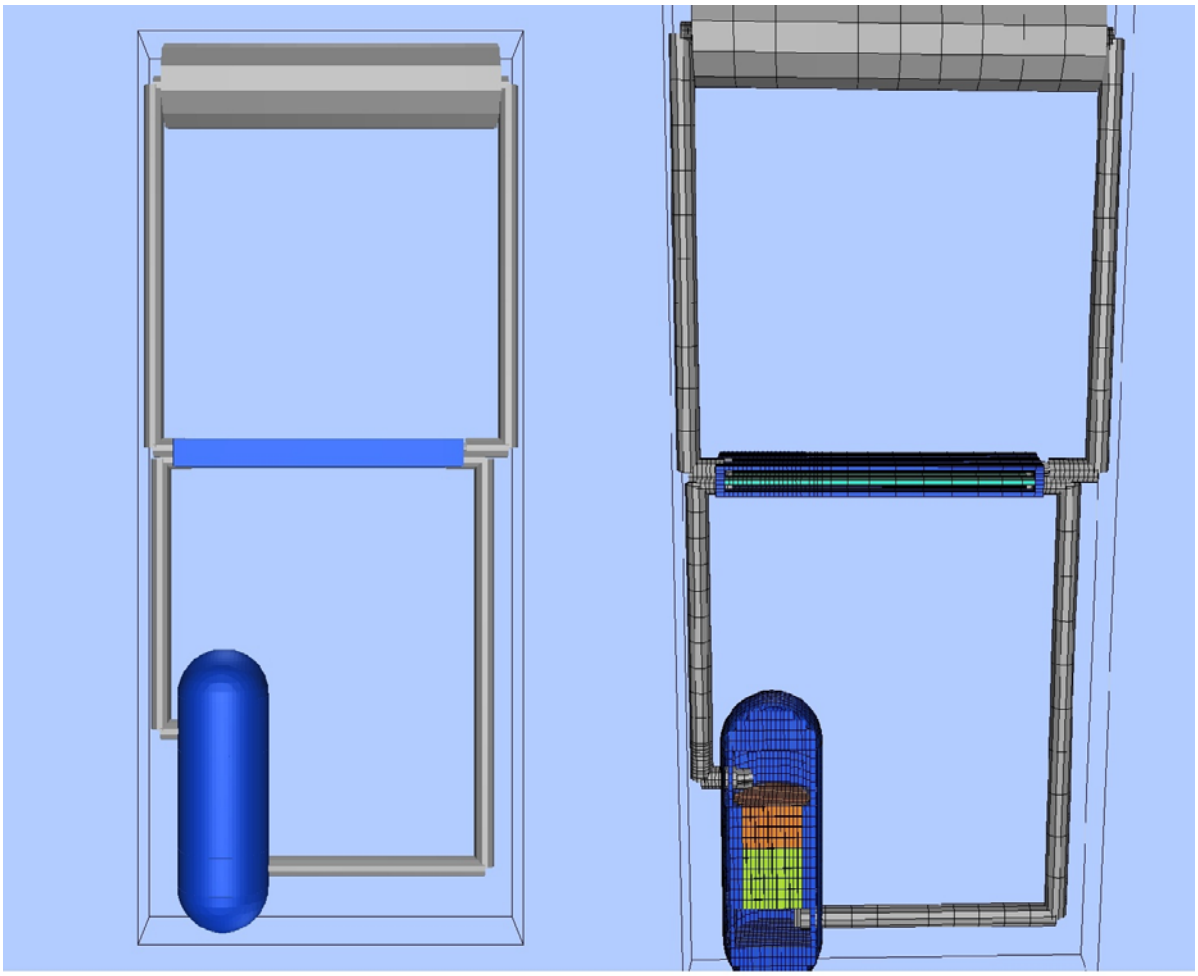

Figure 25. CD3-SC Model for a Reactor, Recuperator, and Heat Exchanger.

Future efforts will include developing better models of recuperators, heat exchangers, turbomachinery, and other necessary components found in the S-CO_2 power conversion system. Future analyses will include different geometries, component layout, and elevation, including the volume control pressurizer/accumulator and both steady-state and transient behavior of the system. Once a baseline configuration is developed and analyzed, more complex accident analyses can be investigated for anticipated transient events, anticipated transients without scram (ATWS), and LOCAs.

7.3 Natural Circulation Test Plans With the SNL S-CO_2 Research Loop

Although the scoping calculations show that the SC-GFR can very likely use natural circulation mechanisms to passively remove decay heat from the reactor, more experimental validation work needs to be performed. SNL plans to perform a series of experiments in the test S-CO_2 compression loop to verify the capability of natural circulation and to validate the codes and models. The testing and modifications are scheduled for the second half of fiscal year 2011. A major difference between these proposed experiments and those previously performed by Milone

(2009) is the availability of a flow meter within the loop and the heating and cooling capabilities of the loop (20-50 kWth).

A schematic of the SNL S-CO_2 compression loop is illustrated in Figure 26, and an engineering drawing is shown in Figure 27. A photo of the actual loop is provided in Figure 28. The loop has two heat exchangers. In Figures 27 and 28, one is located in the lower portion of the loop (large light blue tank) and the other in the upper portion (top-hat-shaped heat exchanger). The intended modifications include elevating the upper "top hat" shaped heat exchanger by about 3 m by inserting extension sections of 1 ½" schedule 160 piping. The only other modification required is to re-plumb the water circuits so that the heated section uses the lower heat exchanger and the cooling section the upper heat exchanger. The tests will measure the actual flow rate (using a highly accurate Coriolis flow meter), fluid densities, and temperatures as a function of time for various initial fill densities, pressures, heat load, and degree of subcooling in the heat rejection system. The tests results can then be used to validate the natural circulation codes and tools. Currently, the simple lumped-parameter model predicts that 20 kW of heating will produce a flow rate of about 0.2 kg/s and a hot-to-cold leg ΔT of 63K, with a loop ΔP of 4 kPa.

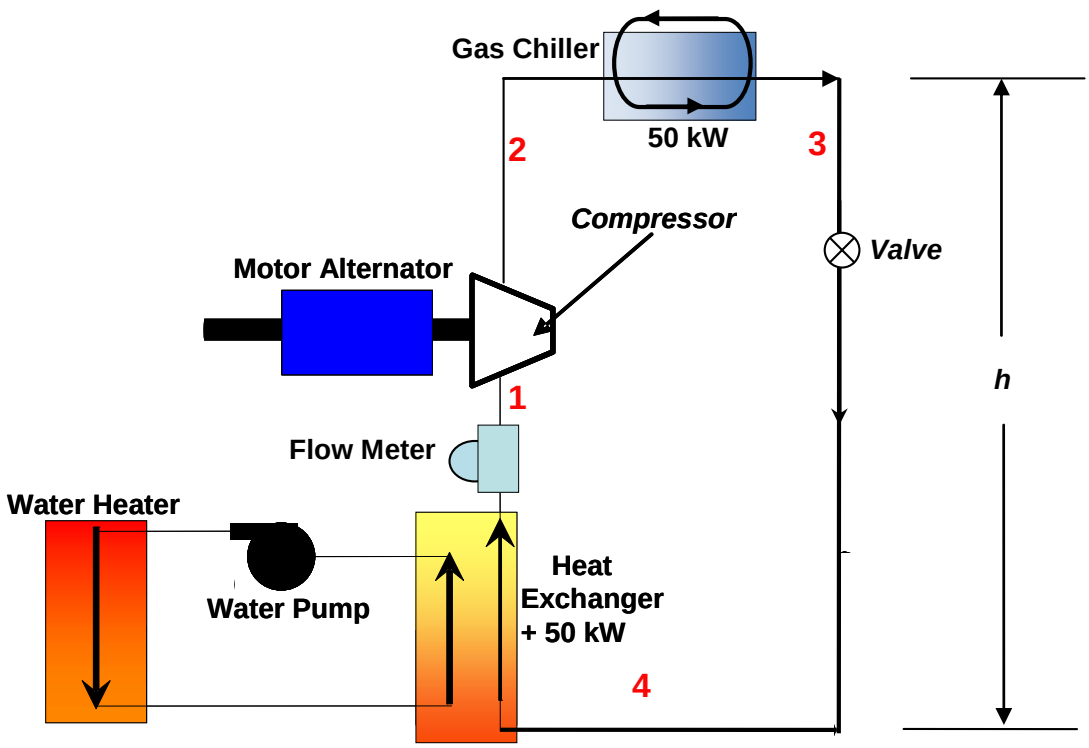

Figure 26. Schematic of the SNL S-CO_2 Compression Loop Modified for Natural Circulation Flow Tests.

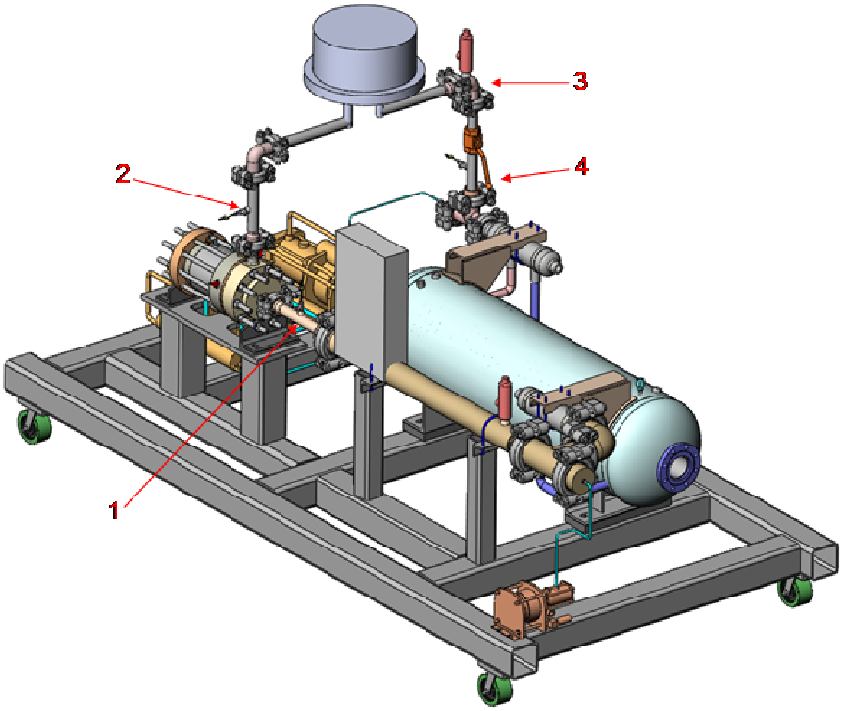

Figure 27. Engineering Drawing of the S-CO_2 Compression Loop With the Addition of a Head-Addition Heat Exchanger.

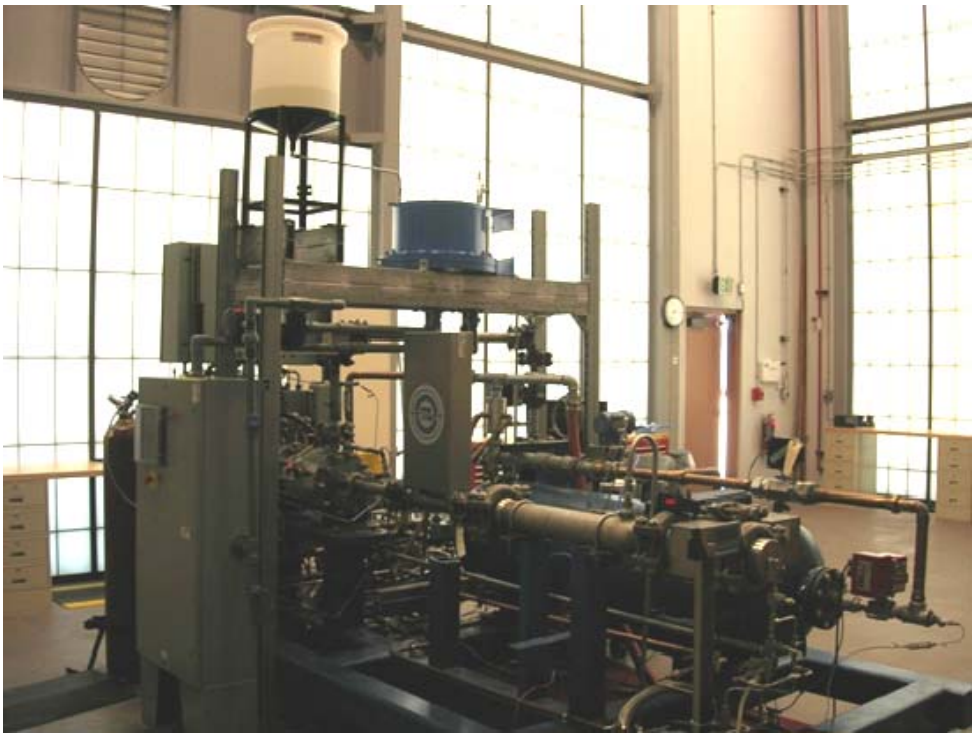

Figure 28. Photo of the SNL S-CO_2 Compression Loop.

8. Future Research

This report has presented the SC-GFR concept and scoping analyses to determine the feasibility of such a design. The results show that the concept has some promising aspects, especially when applied to a small system on the order of 200 to 400 MWth. Of course, a significant amount of research and analysis remains to determine if such a concept could be built and operated as anticipated. The next step in developing the concept is to progress further into the design and safety aspects of the system. The following paragraphs list topical areas that need to be further considered in determining if this transformational concept warrants a more detailed effort.

Reactor Configuration, Design, and Analyses

The reactor core design, pressure vessel design, and control rod design and configuration require ongoing study. The core layout, grid structure, and integration of the control rods and reflector need further development. Continued burnup calculations are required to more accurately determine the three-dimensional inventory in the core. Further MCNP calculations are required to analyze the reactivity void worth and determine the power coefficient from start-up to full power operations and shutdown. Core life reactivity control using the control rods or other mechanism requires further design and analyses. The pressure vessel and upper and lower bulkheads require additional analyses to determine the diameter, thickness, material type, neutron damage, inlet and outlet configuration, downcomer, and control rod integration to the vessel. Corrosion in the core, vessel, and piping requires ongoing study and research for CO_2 at these operating temperatures.

Fuel and Cladding Design

The fuel and cladding design requires further study to determine the most appropriate configuration for a medium-temperature, high-pressure configuration. Cladding lifetime due to neutron damage requires continued development to determine the most appropriate material, thickness, and burnup limitations. The fission gas plenum design and size require further analysis. The fuel type requires ongoing research to determine what is the most appropriate for the reactor and fuel cycle. The pin size, pitch, and integration into the core require more definition. Clad surface roughening or other heat transfer enhancements need further research and analyses. Corrosion issues for the cladding also require ongoing study and research for CO_2 at these operating temperatures.

Plant Layout and Integration

A SolidWorks plant layout needs ongoing development to help visualize the size and other features of the plant. Component sizing and integration require further design innovation. Further study is required to incorporate the advancing PCHE technology into an S-CO_2 power conversion system using a cost effective and feasible approach. Determining interfaces to auxiliary systems and determining core refueling options and pressure vessel replacement, require continued development.

Thermodynamic Cycle Analysis and Optimization

As the S-CO_2 loop experiments progress, and more sophisticated and complex hardware is developed and integrated, further thermodynamic and cycle analysis and optimization are required. Both simple and complex S-CO_2 cycle analyses need to be performed to ultimately determine the trade off of complexity to efficiency and viability in plant design. It may

ultimately be more cost effective to maintain as simple a power conversion system as possible, even though a lower efficiency is found. Trade-off studies comparing efficiency, complexity, and viability with cost are needed to help establish the payoff potential for more complicated systems. Further analyses must be completed to determine the penalties associated with lowering the pressure in the reactor vessel by lowering the pressure ratio or including a turbine-generator unit before to the reactor vessel inlet.

Thermal Hydraulics Analysis at Full Power

Scoping analyses have shown the range for the pin size and pitch. Further, more sophisticated analysis is required to determine the optimized configuration in relation to the maximum fuel temperature, cladding temperature, and core pressure drop. Analyses are necessary to determine if the core should have ducted assemblies to better control the flow in the core. Additional study is also essential in determining the most appropriate mechanism for flow distribution in the core, either by flow orifices in the lower grid structure or a varied pitch configuration.

Natural Circulation Flow Modeling

Natural circulation flow analyses require further modeling development of the code CD3-SC to determine both steady-state and transient flow conditions and optimization for decay heat removal without the compressor operating. Continued investigation to determine the flow through the power conversion system and optimization of the heat exchanger elevation is necessary. Analyses are required to determine the potential for an auxiliary cooling system design. Validation work should continue on the SNL S-CO_2 flow loops to develop validation experiments and results.

Auxiliary Systems

Analyses and development of auxiliary systems need ongoing study. Development of a volume control system using a pressurizer/accumulator, as well as the gas makeup and cleanup system is required. An auxiliary decay cooling system requires further development. The features for an emergency core cooling system need study and development. The pressure vessel vault design and core refueling system must be developed. The seismic and operating requirements for the containment structure and ventilation system need to be established and incorporated into the design effort.

Accident Analysis

Development of the safety case for accident analysis must begin. Once the CD3-SC code is finalized with a baseline configuration, anticipated transients, ATWS, and LOCAs can be evaluated from a plant thermal hydraulics perspective. Other accident analyses codes need to be identified and modified for the SC-GFR concept.

Economic Analysis

An economic analysis is essential as the cycle, plant, and reactor are optimized to determine if this type of reactor concept and fuel cycle would be economically viable. This report has identified that the initial core loading, the PCHE recuperators, and the heat rejection system could be significant drivers in the plant's capital cost. The use of stainless steels to minimize corrosion will also have a significant impact in the plant's capital cost. More rigorous corrosion testing needs to be performed to determine where low cost materials can be used in the plant.

9. Conclusions

The SC-GFR reactor concept was developed to determine the feasibility of an RSR type concept using S-CO_2 as the working fluid in a direct cycle fast reactor. Scoping analyses were performed for a 200 to 400 MWth reactor and an S-CO_2 Brayton cycle. Although a significant amount of work is still required, this type of reactor concept maintains some potentially significant advantages over ideal gas-cooled systems and liquid metal-cooled systems. The analyses presented in this report show that a relatively small long-life reactor core could be developed that maintains decay heat removal by natural circulation. The concept is based largely on the AGR commercial power plants operated in the UK and other GFR concepts. This work was performed as part of the Advance Reactor Concepts Working Group – Transformational Reactor Concepts.

10. References

Angelino, G., "Carbon Dioxide Condensation Cycles for Power Production," ASME 68-GT-23, Am. Soc. Mech. Eng., 1968.

Angelino, G., "Real Gas Effects in Carbon Dioxide Cycles," ASME 69-GT-103, Am. Soc. Mech. Eng., 1969.

ANS – "World List of Nuclear Power Plants," Nuclear News, March 2010.

Chalasani, N.R., Greiner, M., and Suo-Anttila, A., "Validation of Container Analysis Fire Environment (CAFE) Code for Memorial Tunnel Fire Ventilation Test Program," PVP2010-26075, Proceedings of the ASME 2010 Pressure Vessels and Piping Division / K-PVP Conference, Bellevue, Washington, July 18-22, 2010.

Dumaz, P., Allegre, P., Bassi, C., Cadiou, T., Conti, A., Garnier, J. C., Malo, J. Y., and Tosello, A., "Gas-Cooled Fast Reactors – Status of CEA Preliminary Design Studies,", Nuclear Engineering and Design, Vol. 237 p. 1618, 2007.

GIF – Generation IV International Forum, "A Technology Roadmap for Generation IV Nuclear Energy Systems," U.S. DOE Nuclear Energy Research and Advisory Committee, GIF-002-00, December 2002.

Greiner, M. and Suo-Anttila, A., "Validation of the ISIS Computer Code for Simulating Large Pool Fires Under a Varity of Wind Conditions," ASME J. Pressure Vessel Technology, Vol. 126, pp. 360-368, 2004.

Handwerk, C. S., "Optimized Core Design of a Supercritical Carbon Dioxide-Cooled Fast Reactor," Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, June 2007.

Kato, Y., Nitawki, T., and Muto, Y., "Medium Temperature Carbon Dioxide Gas Turbine Reactor," Nuclear Engineering and Design, Vol. 230, p. 195, 2004.

Lewins, J., Nuclear Reactor Kinetics and Control, Pergamon Press, New York, NY, 1978.

Milone, D. V., "Computational Model and Experimental Data on the Natural Circulation of Supercritical Carbon Dioxide," Proceedings of S-CO_2 Power Cycle Symposium, RPI, Troy, NY, April 29-30, 2009.

Muto, Y. and Kato, Y., "Optimal Cycle Scheme of Direct Cycle Supercritical CO_2 Gas Turbine for Nuclear Power Generation Systems," Int. Conf. Power Engineering, Hangzhou, China, October 23-27, 2007.

MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, Los Alamos, NM, April 2003.

NIST Standard Reference Database 23 – NIST Reference Thermodynamics and Transport Properties – REFPROP, Version 8.0, April 2007.

Newton, T. D. and Smith, P. J., "Design and Performance Studies for Minor Actinide Target Fuels," ARWIF-2001, Chester, UK, October 22-26, 2001.

Parma, E. J., "BURNCAL: A Nuclear Reactor Burnup Code Using MCNP Tallies," SAND2002-3868, Sandia National Laboratories, Albuquerque, NM, November 2002.

Pope, M. A., "Reactor Physics Design of Supercritical CO_2-Cooled Fast Reactors," Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, September 2004.

Pope, M. A., Lee, J. I., Hejzlar, P., Driscoll, M. J., "Thermal Hydraulic Challenges of Gas Cooled Fast Reactors With Passive Safety Features," Nuclear Engineering and Design, Vol 239, p. 840, 2009.

Shropshire, D. E., "Lessons Learned From GEN I Carbon Dioxide Cooled Reactors," Proceeding of ICONE 12, Arlington, VA, April 25-29, 2004.

Van Rooijen, W. F. G., "Gas-Cooled Fast Reactor: A Historical Overview and Future Outlook," Review Article, Science and Technology of Nuclear Installations, Article ID 965757, 2009.

Weaver, K. D. and Khalil, H. S., "Gas Cooled Fast Reactor," American Nuclear Society Winter Meeting, Washington, DC, November 2002.

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