INEEL/EXT-04-02242

GEN IV Nuclear Energy Systems

Initial Requirements For Gas-Cooled Fast Reactor (GFR) System Design, Performance, And Safety Analysis Models

Thomas Y. C. Wei
Kevan D. Weaver

August 2004

Idaho National Engineering and Environmental Laboratory
Bechtel BWXT Idaho, LLC

INEEL/EXT-04-002242

GEN IV Nuclear Energy Systems
Initial Requirements For Gas-Cooled Fast Reactor (GFR) System Design, Performance, And Safety Analysis Models

Thomas Y. C. Wei – ANL
Kevan D. Weaver - INEEL

August 2004

Idaho National Engineering and Environmental Laboratory

Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Assistant Secretary
Under DOE Idaho Operations Office
Contract DE-AC07-99ID13727

Table of Contents

1. Introduction **3**
2. Design Options for the GFR **4**
2.1 Reference Design *4*
2.2 Optional Designs *4*
2.3 GFR Fuel *5*
3. Design Configuration Modeling Requirement **7**
3.1 Direct Cycle Plant Design Configuration *7*
3.2 Indirect Cycle Plant Design Configuration *13*
4. Plant Design Modeling Requirements **15**
4.1 Core Neutronic Models *15*
4.1.1 Fuel Models 15
4.1.2 Modeling Data and Experimental Needs 16
4.2 Core Engineering Design Models *16*
4.3 Primary System Thermal-Hydraulic Models *17*
4.4 Balance of Plant Models *17*
5. Plant Performance Modeling Requirements **18**
5.1 Plant Operation Modeling *18*
5.2 Plant Control Modeling *20*
5.3 Modeling Data and Experimental Needs *20*
6. Plant Safety Modeling Requirements **20**
6.1 Depressurization BDBA Primary System Model *21*
6.2 Depressurization BDBA (Guard) Containment System Model *22*
6.3 Reactivity Control and Transient Model *23*
6.4 Control Rod Ejection Modeling *24*
6.5 Local Flow Blockage Modeling *24*
6.6 Modeling Data and Experimental Needs *24*
7. Modeling and Code Development Costs **25**
8. References **25**

1. Introduction

The gas-cooled fast reactor (GFR) was chosen as one of the Generation IV nuclear reactor systems to be developed based on its excellent potential for sustainability through reduction of the volume and radio toxicity of both its own fuel and other spent nuclear fuel, and for extending/utilizing uranium resources orders of magnitude beyond what the current open fuel cycle can realize. In addition, energy conversion at high thermal efficiency is possible with the current designs being considered, thus increasing the economic benefit of the GFR. However, research and development challenges include the ability to use passive decay heat removal systems during accident conditions, survivability of fuels and in-core materials under extreme temperatures and radiation, and economical and efficient fuel cycle processes. Nevertheless, the GFR was chosen as one of only six Generation IV systems to be pursued based on its ability to meet the Generation IV goals in sustainability, economics, safety and reliability, proliferation resistance and physical protection.

GFR work (together with the CEA-France/ANL-US I-NERI project) has reached the stage where the effort will now focus on the characterization of point designs [1]. Exploratory studies have been performed on a broad range of fuel forms and types, core configurations, coolant types, and primary system/BOP concepts. Major goals and criteria specifically formulated for the GFR, which guided the major focus of the exploratory effort on innovative concepts, have resulted in:

1. Reactor core design concepts which meet the goal of sustainability (conversion ratio=1.0) with low proliferation risk (no external blankets), and homogeneous recycle of minor actinides.
2. Passive decay heat removal concepts in combination with active systems.
3. A GFR concept that makes maximum use of high temperature VHTR technology (direct cycle, cogeneration capability) to minimize R&D costs and development time.

With the completion of exploratory studies, the project is now in a position to begin identifying analysis modeling needs, which has focused on a limited number of design options identified by the exploratory assessments. In addition, specific design issues that have been identified by the work-to-date, and which need to be evaluated, are included. Analysis models will have to be developed for the resolution of these issues. An umbrella plant duty cycle has been defined and modeling phenomena outlined here. The process for defining analysis modeling requirements begins with this step, and future requirements of this duty cycle and set of design issues will further refine the set of modeling requirements.

2. Design Options for the GFR

2.1 Reference Design

The reference GFR system features a fast-spectrum, helium-cooled reactor and closed fuel cycle (see Figure 2.1). This was chosen as the reference design due to its close relationship with the VHTR, and thus its ability to utilize as much VHTR material and balance-of-plant technology as possible. Like thermal-spectrum helium-cooled reactors such as the Gas- Turbine Modular Helium Reactor (GT-MHR) and the Pebble Bed Modular Reactor (PBMR), the high outlet temperature of the helium coolant makes it possible to deliver electricity, hydrogen or process heat with high conversion efficiency. The GFR reference design uses a direct-cycle helium turbine for electricity (42% efficiency at 850°C), and process heat for thermochemical production of hydrogen.

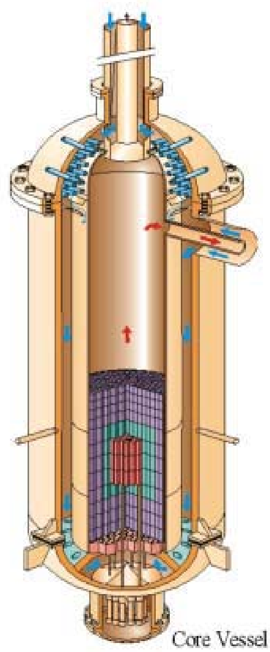


Figure 2.1. Possible GFR vessel and core configuration for block/plate core.

2.2 Optional Designs

The alternate design is also a helium-cooled system, but utilizes an indirect Brayton cycle for power conversion. The secondary system of the alternate design utilizes supercritical CO_2 (S-CO_2) at 550°C and 20 MPa (see Figure 2.2). This allows for more modest outlet temperatures in the primary circuit (~ 600-650°C), reducing the strict fuel, fuel matrix, and material requirements as compared to the direct cycle, while maintaining high thermal efficiency (~ 42%).

The optional design is a S-CO_2 cooled (550°C outlet and 20 MPa), direct Brayton cycle system. The main advantage of the optional design is the modest outlet temperature in the primary circuit, while maintaining high thermal efficiency (~ 45%). Again, the modest outlet temperature (comparable to sodium-cooled reactors) reduces the requirements on fuel, fuel matrix/cladding, and materials, and even allows for the use of more standard metal alloys within the core. This has the potential of significantly reducing the fuel matrix/cladding development costs as compared to the reference design, and reducing the overall capital costs due to the small size of the turbo machinery and other system components. The power conversion cycle is equivalent to that shown in Figure 2.2, where the IHX would be replaced by the reactor and reactor pressure vessel.

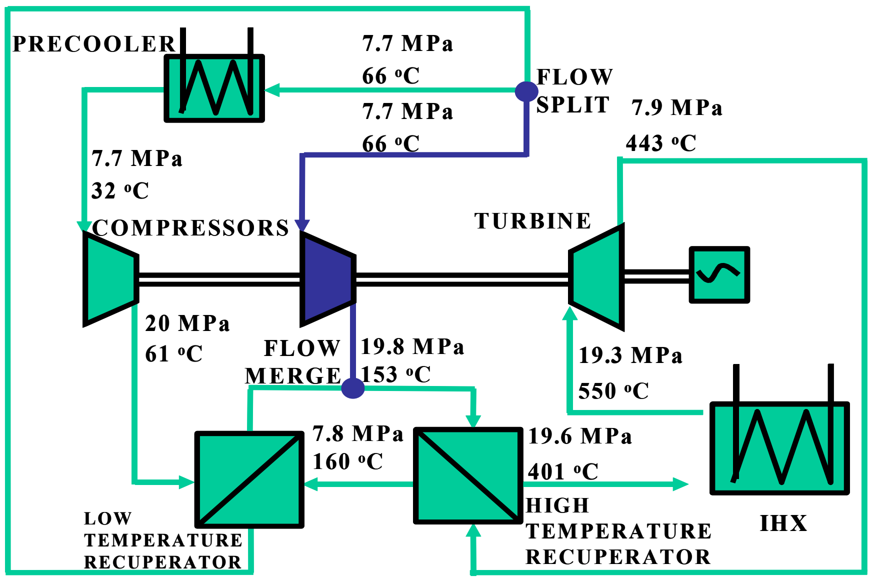


Figure 2.2. Schematic of the S-CO_2 recompression cycle.

2.3 GFR Fuel

The safety system design will be affected by the choice of primary coolant, whether a direct or indirect power conversion cycle is used, and the core geometry (i.e., block, plate, pebble, etc.). The trade-off between high conductivity and high temperature capabilities has led to the choice of ceramics, including refractory ceramics. The reference fuel matrix for the Generation IV GFR is a cercer dispersion fuel, based on a balance between conductivity and high temperature capability. Figures 2.3 and 2.4 are graphical representations of the fuel types being considered.

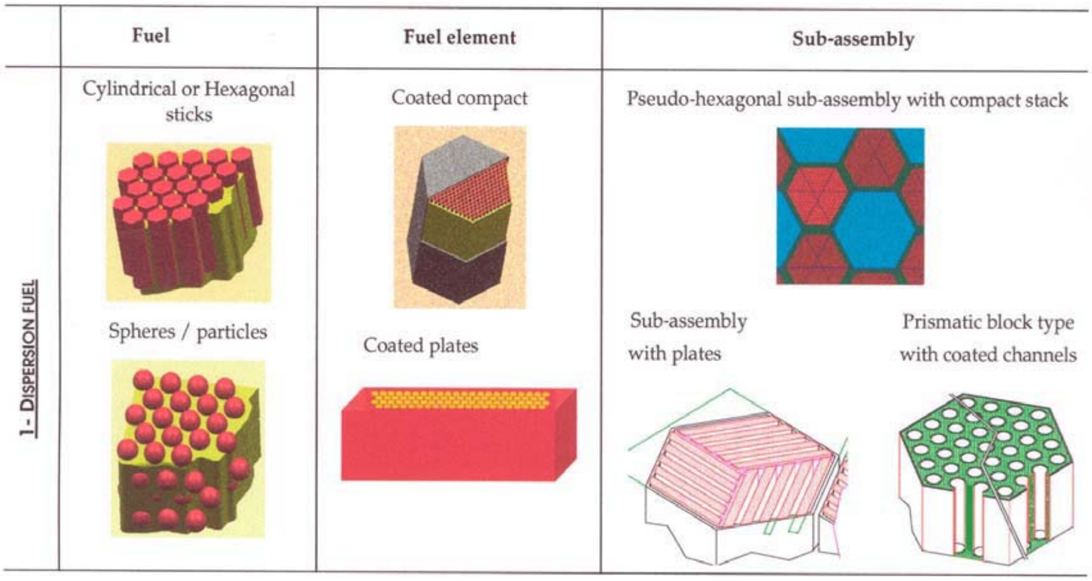


Figure 2.3. Dispersion fuel.

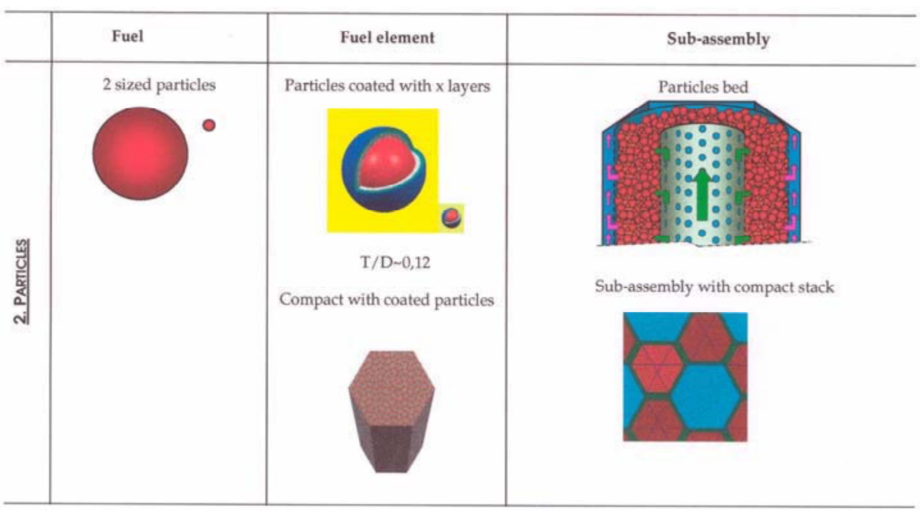


Figure 2.4. Particle fuel.

Current fuel designs are based on dispersion fuels (either as fibers or particles) in an inert plate/block type matrix, with option to use particle fuel in an inert pebble matrix, or solid solution fuel clad in a refractory ceramic (e.g., SiC/SiC composites). The reference fuels chosen for the GFR are UN and UC for their high heavy metal density, high conductivity, and minimal impact on neutron spectrum (although limited irradiation data exists). The matrix materials are dependent on the coolant and operating temperatures, and can be classified into three categories: ceramic (for high temperatures), refractory metal (for modest to high temperatures), and metal (for modest temperatures). As the fuels are of ceramic composition, the resulting fuel forms can be classified into two categories: cercer and cermet. The fuel fibers, or "sticks", would be extruded into the matrix, where the matrix would have a "honeycomb" appearance. The particles may be coated, but, unlike the thermal spectrum gas reactor fuel, will most likely have one coating to maximize the heavy metal content within the matrix.

It is important to note that fuel development, including fabrication and irradiation performance, is a key viability issue for the GFR, and cannot be separated from the safety design and performance of the GFR. Fuel mechanical and thermal properties are needed from beginning to end-of-life of the reactor to support the safety case, and will have a significant impact on the safety system design work. In addition, fuel development will include the viability of using minor actinide bearing material, which will have further affects on the performance of the GFR.

3. Design Configuration Modeling Requirement

The different design options and combinations have been filtered down to:

1. Fuel choice:
   - Dispersed fuel in plates.
   - SiC clad pellets in pins.
   - Actinide compounds are carbide in the design studies but nitride remains a possible candidate.
2. Unit size:
   - 600 MWt (modular concept) and 2400 MWt (economics of scale)
3. Power density:
   - 100 MW/m3
4. Passive decay heat removal approach:
   - Natural convection with double containment, but creative work on possible alternative options is not excluded.
5. Direct cycle, helium cooled balance-of-plant and indirect cycle primary helium cooled with secondary supercritical-CO_2 (S-CO_2) BOP.

Modeling tools will have to be provided for this set of design options and safety approach. Details are given in subsequent sections regarding modeling requirements for each of these general areas.

3.1 Direct Cycle Plant Design Configuration

The reference 600 MWth configuration is the cercer based core at 100 MW/m^3 power density with the challenging 70/30 dispersed fuel (i.e., 70% fuel – 30% matrix), using a direct cycle, helium cooled plant. The 2400 MWth direct cycle option is a scale-up of this plant.

A summary of the design configuration is given here.

1. Pre-stressed concrete cavity housing the entire primary circuit.
2. Leak tight cavity as second barrier with dry air (P=2MPa) and helium natural convection to remove the decay heat passively.
3. DHR loops:
   - 3 loops 100%, 3x20MW (diversified)
4. Secondary side:
   - Water system at 0.3 MPa; designed to withstand 7 MPa primary pressures.
5. Passive check valves to avoid core bypass in normal operation.
6. Each DHR loop provided with 2 blowers (2x100%) with secure electrical supply, but not upgraded.

Figure 3.1 shows the plant equipment layout while Table 3.1 provides the steady state operating conditions. For the direct cycle plant option, this is the configuration and the steady state conditions that the models will be required to analyze. Details of the DHR removal loops, core data, fuel handling system, and control drive mechanisms are shown in Figure 3.2 through Figure 3.4. This leads to the following overall assembly shown in Figure 3.5.

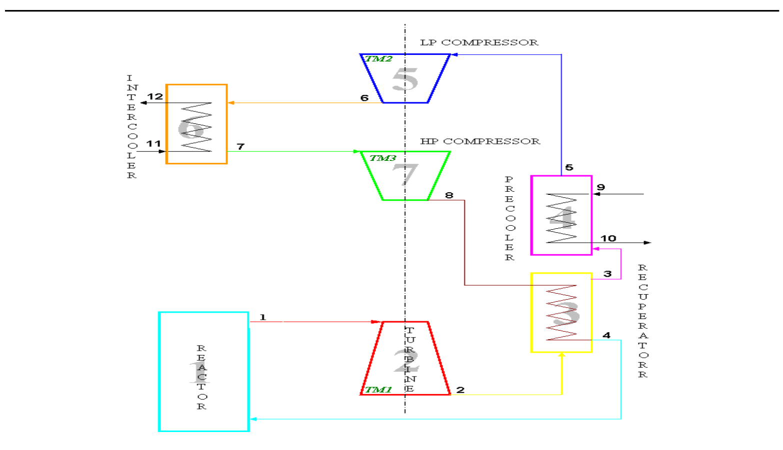


Figure 3.1. Plant Equipment Layout

- Adapted from 600MWt GT-MHR
- Direct cycle
- Single shaft

Table 3.1. Full Power Operating Conditions

	GA Point Design
Reactor	
Core Power, MW(t)	600
Core Inlet/Outlet Temperatures, $^oC/^oC$	490/850
Core Upper Plenum Inlet Pressures, MPa	7.07
Helium Mass Flow Rate, kg/s	320
Turbomachinery	
Turbine Mass Flow Rate, kg/s	320
Turbine Inlet/Outlet Temperatures, $^oC/^oC$	850/510
Turbine Inlet/Outlet Pressures, MPa/MPa	7.02/2.65
Compressor Inlet/Outlet Temperatures, $^oC/^oC$	33/112
Compressor Inlet/Outlet Pressures, MPa/MPa	2.60/7.24
Compressor Overall Pressure Ratio	2.82
Recupeator	
Mass Flow Rate, kg/s	322
Hot Side Inlet/Outlet Temperatures, $^oC/^oC$	510/131
Cold Side Inlet/Outlet Temperatures, $^oC/^oC$	112/490
Precooler	
Mass Flow Rate, kg/s	322
Inlet/Outlet Temperatures, $^oC/^oC$	131/33
Intercooler	
Mass Flow Rate, kg/s	322
Inlet/Outlet Temperatures, $^oC/^oC$	112/33

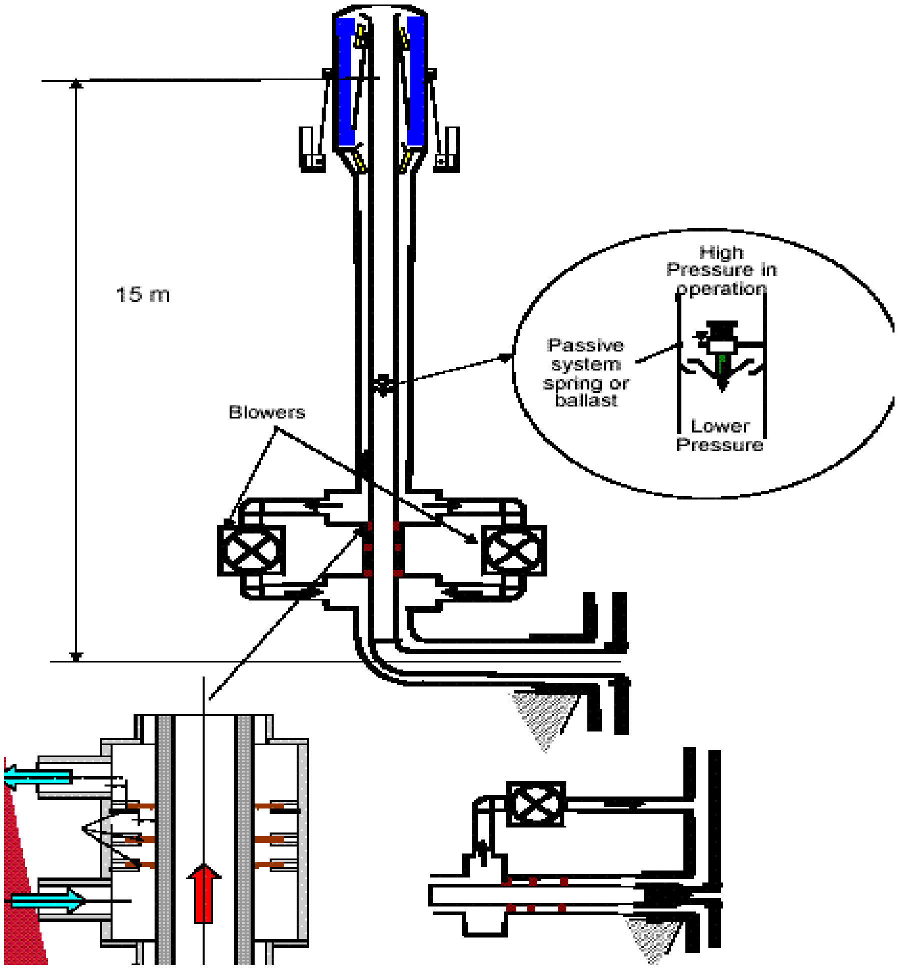


Figure 3.2. DHR Loop Preliminary Design

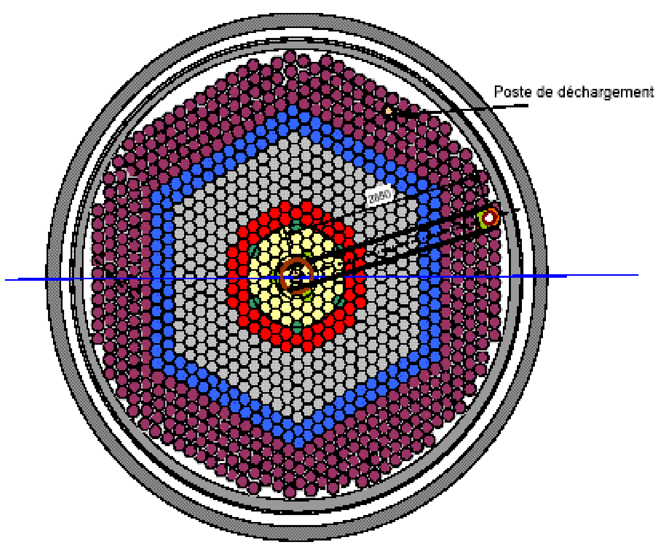


Fig 3.3. Vessel Radial Cross Section

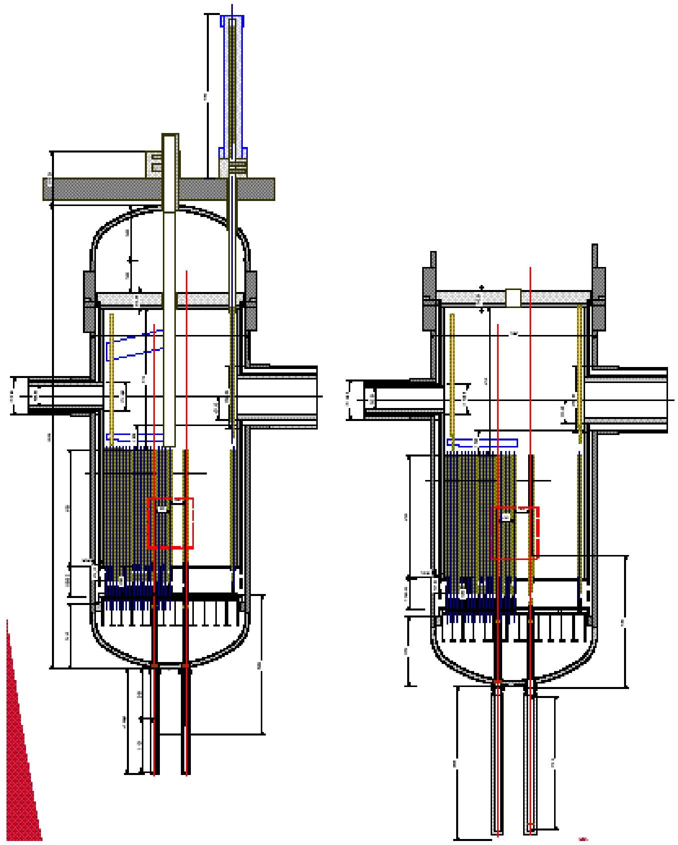


Figure 3.4. Vessel Elevation View

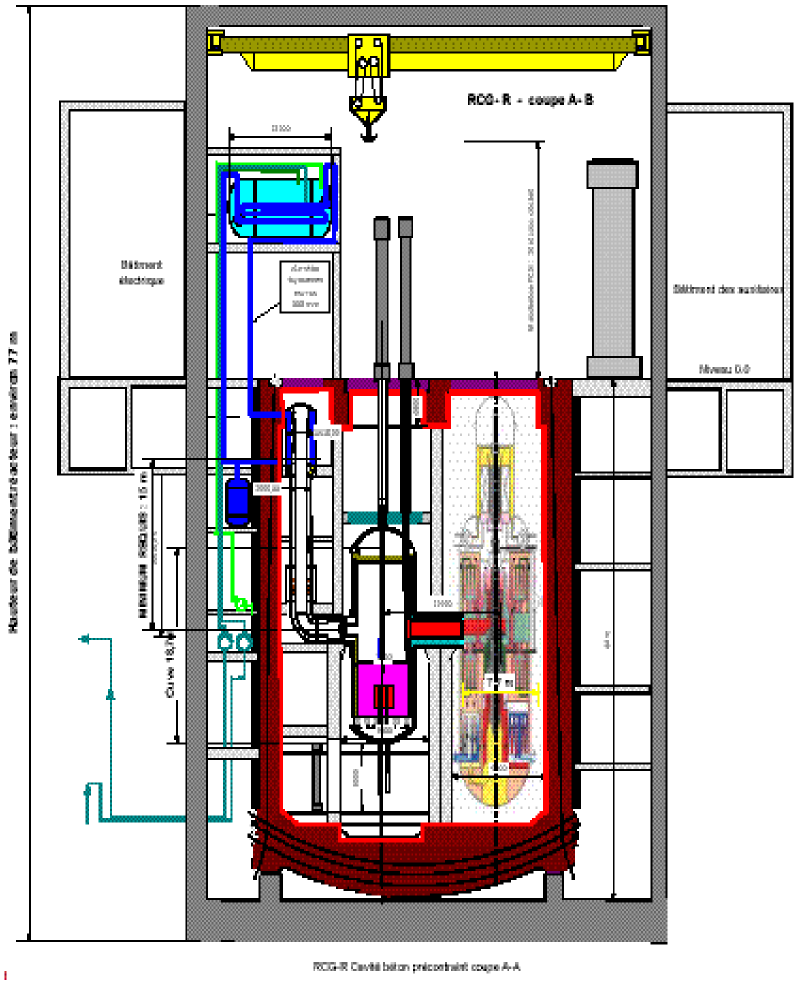


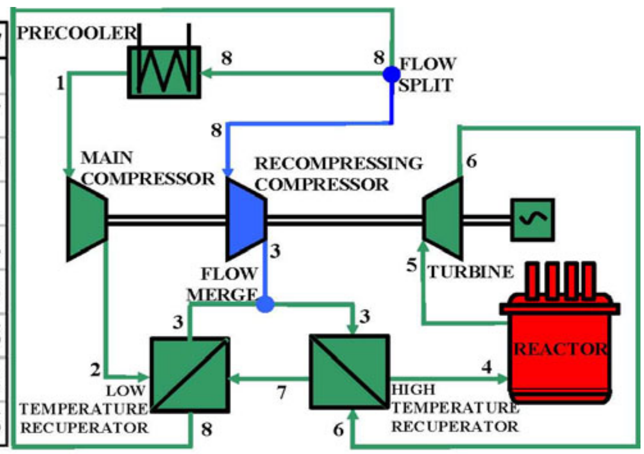
Figure 3.5. Containment Building Elevation Views

3.2 Indirect Cycle Plant Design Configuration

A number of indirect cycle design options and combinations have been explored. The design configurations summarized here are a collection of these discussions and should be regarded as only the starting point for the point design characterization effort on the 2400 MWt unit indirect cycle plant option. The He/He option has been dropped, and thus the focus will be on the reduced temperature primary He/S-CO_2 secondary option. The design configuration provided here is for the He/He option that will have to be modified for the primary He/S-CO_2 secondary option. In particular, the in-vessel IHX will be replaced by an ex-vessel IHX. The suggested starting point has a secondary side S-CO_2 temperature of 550^{O}C at the IHX outlet. This design would have a core outlet temperature of 650^{O}C and a cycle efficiency of 41%. The optimized design S-CO_2 secondary side state points for the IHX outlet temperature of 500^{O}C are shown in Table 3.2.

Table 3.2. Optimized Design S-CO_2 Statepoints for T=550^{o}C

Point	P	T	Enthalpy
	(MPa)	(oC)	(kj/kg)
1	7.692	32	306.7
2	20.000	61.1	327.3
3	19.989	158	536.1
4	19.958	396.5	846.4
5	19.828	550	1035.3
6	7.901	440.3	914.5
7	7.814	168.3	604.2
8	7.705	69.6	478.6



For conservative compressor (89%) and turbine (90%) efficiencies
thermal/net efficiency =45.3%/ 41.0%

Figure 3.6 to Figure 3.7 shows the plant layout for the He/He indirect cycle option. This layout will have to be modified to accommodate the selected He/S-CO_2 indirect cycle option. Figure 3.6 shows the primary system elevation cross-section for the He/He indirect cycle option that will be modified when the He/He IHX is replaced with the He/S-CO_2 IHX. Figure 3.7 shows additional details of the circulator layout. The guard containment (GC) elevation view for the 5 bar pressure case is presented in Figure 3.6. The guard containment plane view is shown in Figure 3.7.

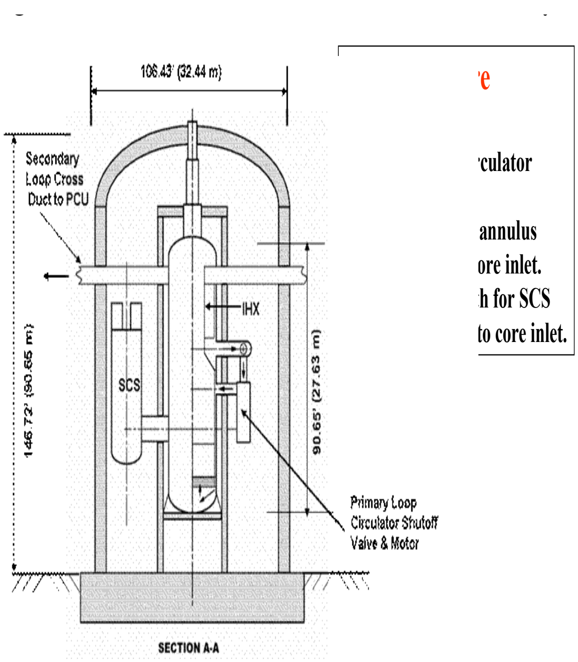


Figure 3.6. GC Elevation View for He/He Indirect Cycle

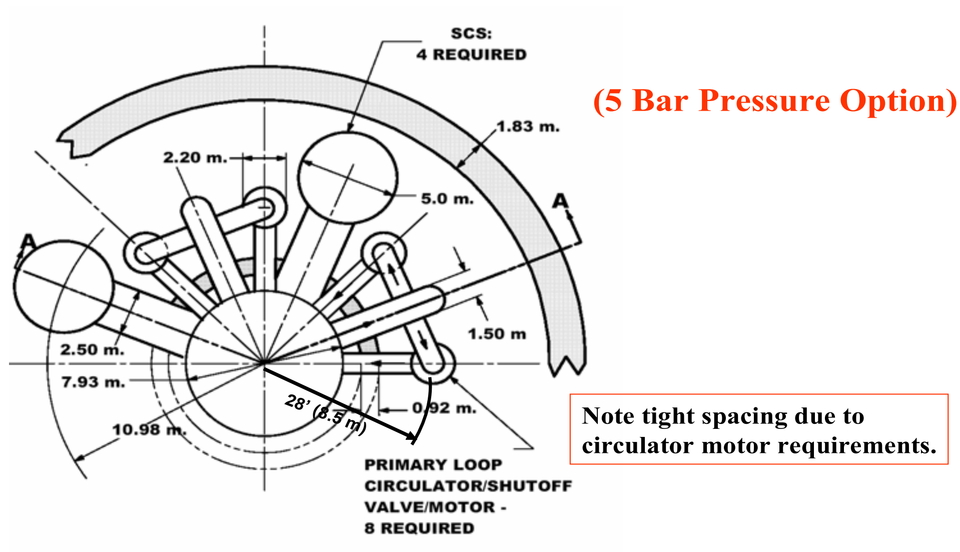


Figure 3.7. GC Layout View for He/He Indirect Cycle

4. Plant Design Modeling Requirements

The design and safety analysis for the GFR will require major adaptations or further development of calculation tools to accommodate the innovative features in the field of core design (new fuel and sub-assembly forms), fuel composition (homogeneous recycling of major actinides with a robust on-site integrated cycle), implementation of safety devices, and the necessity of demonstrating the safe behavior of the system under all operational conditions [2]. This will call for adaptation of neutronics, thermo-aero-mechanics, systems, and safety analysis computer codes. Furthermore, qualification work (e.g., benchmarking against critical experiments and sub-assembly mock-up testing) will also have to be considered. The adoption of a Core Melt Exclusion Strategy should rule out the need for severe accident models, but future work may reconsider this strategy based on design evolution.

4.1 Core Neutronic Models

The current neutronic modeling effort utilizes standard codes, such as MCNP and the REBUS suite. The Monte Carlo codes (e.g., MCNP) are based on “exact” 3-D geometry, and continuous treatment of energy, space, and angle for the problem. This leads to the reduction of systematic computation errors, but requires a large number of particle histories to reduce the overall error of the results. The result is a hi-fidelity model that requires long run times. Deterministic codes, on the other hand, can greatly reduce the computational time, but cannot treat the geometry explicitly.

Specific modeling requires codes that can perform analysis of reactivity initiated accidents, e.g., control rod ejection, and also includes fuel performance from beginning to end-of-life, reactivity limited burnup, and reflector performance. Analysis of reactivity coefficients is also key, which would include void, expansion, Doppler, etc., and will require reliable models and data.

Current models involve calculations performed by both stochastic and deterministic codes, but also have other challenges that need to be addressed. These challenges are discussed below.

4.1.1 Fuel Models

While the functional area of “fuels” falls under the purview of AFCI, fuel modeling will be required for all reactor types under Gen IV, including the GFR. In addition, the fuel will play an integral role in the safety case for the GFR, as fuel development and modeling are as high a priority as the system design work.

During the first stage, candidate fuels will have to be evaluated with current or slightly modified fuel codes, integrating the properties of new materials to be used, and the specific irradiation conditions for the GFR. As the options for the fuel are refined, modeling will have to progress, and may generate specific needs in terms of out-of-pile or in-pile experiments. The decision to develop new codes, or to adapt existing codes, will depend on the degree of innovation of the selected reference fuel options. Code development will be required not only to describe the fuel behavior, but also the global behavior of the fuel subassemblies and their configurations in the core. In that area, fuel subassemblies may also present innovative configurations, and the question of their thermal-mechanical and aerodynamic behavior will be a key issue. Code qualification will require benchmarks involving instrumented subassembly mock-ups to be tested in representative helium flows. Reactor and vessel thermal-hydraulic features will also have to be addressed. Given the early stage of the design, the top level thermal-hydraulic modeling issues which would be part of the data and experimental needs for future tasks would be:

1. Core pressure loss correlations that include orificing, inlet/outlet and other forms losses due to supporting structure.
2. Outlet plenum jetting and mixing
3. Check valve leakage and stratification in the various plena and ducts.

4.1.2 Modeling Data and Experimental Needs

Data needed to obtain an accurate analysis mainly includes:

1. Neutron cross section data, particularly data in the unresolved resonance regions of the minor actinides.
2. Other cross section data that would be important for analysis purposes, such as scattering cross section for the candidate reflector materials.
3. Temperature dependent cross section data.

In order to obtain the above data, experiments will need to be performed. This includes:

1. Cross section experiments for minor actinides and other materials.
2. Critical experiments to measure material worth's, particularly those of the reflector.

4.2 Core Engineering Design Models

Gas-cooled reactors have engineering design issues in the core, some of which are unique to the coolant genre, and others that are common across a broad range of reactor types. Some of these issues are: thermal stress, fluid flow instability caused by the temperature dependent viscosity, and vibration. The objective of the design models is to develop the confidence that the proposed high coolant temperature for the GFR, high temperature gradients, and high coolant gas velocity does not lead to unacceptable mechanical/thermal consequences in the core. Modeling needs are described below:

1. There is a need for thermal-stress models to establish the temperature and stress fields for the different fuel plate designs and pin configurations. Large temperature gradients, and the potential for thermal shock, need to be factored into the development of finite element models for the novel geometries with these temperature fields.
2. There is a need for parallel channel thermal-hydraulic models for the various fuel assemblies, regions of the in-core heat sinks, and the control rod locations. This will be required for various power to flow conditions. Flow instability modeling will need to be developed.
3. There is a need for gas dynamic models to define the fluid forcing terms for the mechanical structure in the core region, and also evaluation of the possible acoustic forcing functions. With these forcing functions, models are required to evaluate the novel fuel assembly configurations and conditions to assess the vibration potential of each of the designs. Coupled neutronic models will need to assess the core power/reactivity stability under these conditions. Structural mechanics design tools will be needed for additional core restraints, if necessary, to mitigate possible core power reactivity fluctuations.

4.3 Primary System Thermal-Hydraulic Models

Table 4.1 shows the major components and regions of the primary system. Single phase models in this area are required to develop the confidence that the proposed high coolant temperature, high temperature gradients, and the coolant gas velocity of the GFR does not lead to unacceptable mechanical/thermal consequences for the primary internals, the vessel, or the downstream balance of plant. The adequacy of cooling certain regions of the primary systems, shielding, insulation, and the internals could be treated by one-dimensional modeling. The boundary between the hot gas and the cold gas will require attention in the modeling.

However, for the more complex geometries of the plena, computational fluid dynamics (CFD) models to analyze flow stagnation and starvation regions would be needed. In particular, streaking or striping in the core outlet plenum, and stratification in the core inlet plenum should be evaluated. Streaking, if transported through the cross vessels, would affect the design of the gas turbines. Examination of vibration potential would require gas-dynamics, and possibly acoustics modeling. Jetting, leading to local wastage of the high temperature insulation, will require models for design evaluation. Potential leakage through the check valves into the shutdown vessel, from the primary vessel during normal operation, may impact the natural convection startup and transition during accident situations. Design models would be needed to assess these issues.

Table 4.1. Primary System Components

Reactor Internals
Neutron Control
Reactor Vessel
Cross Vessel
Downcomer
Primary Circulator (indirect cycle)
Core Support Arrangement
Shutdown Heat Exchanger
Shutdown Circulator
Shutdown Vessel

4.4 Balance of Plant Models

The model requirements for the GFR BOP design are essentially those of the NGNP/VHTR. Table 4.2 shows the components for which models will be required.

Table 4.2. BOP Components

Turbomachinery
Recuperator
Precooler
PCS Component Supports and Ducts
Intercooler
Generator

5. Plant Performance Modeling Requirements

The plant duty cycle describes the type of operation and the plant operational transients that should be considered in evaluating and analyzing the structural design of the systems and components of the primary systems and BOP for the plant. The plant will operate as a base-load plant, but will be capable of part-load operations during its 60-year design life. The plant will not be operated as a traditional load-following plant, i.e., it need not respond directly to the utility system demands. However, the plant will be capable of plant loading and unloading from TBD to 100% of rated power. These load changes will take place in a continuous ramp power change of less than TBD% per minute. In addition, the plant will be designed to accommodate TBD% step load changes.

5.1 Plant Operation Modeling

Table 5.1 represents the anticipated operation modes. A listing of the related operating duty cycle events that need to be factored into the modeling of the design are provided in Table 5.2. Plant performance models will be required to encompass these operating conditions.

Table 5.1. Reactor Operating Modes

Mode	Definition
Refueling Shutdown Condition	When the reactor is at refueling TBD% Δ k/k subcritical and primary coolant T_{avg} is less than TBD°C
Cold Shutdown Condition	When the reactor is at cold shutdown, TBD Δk/k subcritical and primary coolant T_{avg} is less than TBDC
Hot Shutdown Condition	When the reactor is subcritical by TBD Δk/k and T_{avg} is greater than TBD°C
Hot Standby Condition	The reactor is considered to be in a hot standby condition if the average temperature of the primary coolant (T_{avg}) is greater than TBD°C and any of the control rods are withdrawn and the neutron flux power range instrumentation indicates less than TBD% of the rated power
Reactor Critical	The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than TBD% of rated power
Power Operation Condition	When the reactor is critical and the neutron flux power range instrumentation indicates greater than TBD% of rated power

Table 5.2. Duty Cycle Operational Events

1. System Heatup Helium Fill • Heatup to Refueling Temperature
2. Cooldown from Refueling Temperature • Helium Letdown System Cooldown
3. Startup from Refueling Temperature
4. Startup from Hot Standby Condition
5. Shutdown to Refueling Temperature
6. Shutdown to Hot Standby Conditions
7. Loading and Unloading
8. Steady State Temperature Variations
9. Steady State Flow Induced Vibrations
10. Loop Out of Service (N-1 Operation)
11. Stepload Increase or Decrease of TBD% Full Power
12. Turbine Inlet Valve Testing

The duty cycle events are based on the duty cycles for HTGRs, ALMRs, and existing LWR plants. The selected events are representative of conditions which are considered likely to occur during plant operation, and which are sufficiently severe or frequent to be of possible significance to the cyclic behavior of plant components. The events described are based on conservative assumptions; they are meant primarily for use in component stress analysis, and do not necessarily represent actual plant operation. The transient analysis of these events, when used as a base for component structural design, will provide confidence that the component is appropriate for its application over the design life of the plant. Modeling will be required for these events.

5.2 Plant Control Modeling

Models are required to explore methods of power control to meet changing load requirements. Gas reactors have traditionally relied on coolant inventory control to meet different load patterns, as bypass control is regarded as an inefficient method of meeting different load patterns. However, inventory control will require a complex storage system. Models will be needed to:

1. Develop other load change control algorithms.
2. Evaluate the potential for power adjustments without control rod motions.
3. Perform quasistatic analysis to explore the adequacy of these innovative load charge algorithms.
4. Perform dynamic analyses to assess the transient performance of the innovative algorithms with the performance of integrated system calculations.

This will be needed to establish that the plant control systems, and malfunctions of the plant control, do not act in opposition with the inherent response and feedback features of the core and plant. Startup and shutdown issues will also need to be addressed.

5.3 Modeling Data and Experimental Needs

The capability of current codes to describe the normal, and off-normal plant control system transient behavior of the GFR will need to be evaluated. Depending on the effort to be invested and the necessary flexibility for concepts, which may co-exist under various options for a certain time, the opportunity to develop a new code will need to be considered. Whatever the outcome, given the early stage of the design, the top-level modeling issues that would be part of the data and experimental need for future tasks, would center on the reactivity feedback, and the interaction with the design of the core support/restraint mechanical structure.

6. Plant Safety Modeling Requirements

The major modeling requirements in the area of safety are dictated by depressurized decay heat accidents. Consensus has been reached on the safety approach to the depressurized decay heat accidents. It is an alternative based on a well-balanced combination of both active and passive means, termed as a "semi-passive" approach. The guard/proximate containment will still be utilized, but it will be sized for 5 bar backup pressure with an initial pressure of 1 atm. The 5 bar back-up pressure, plus whatever natural convection is available at this pressure, will be utilized to significantly reduce the blower power of the active DHR system (sized to remove 2-3% decay power). The objective is to have such low power requirements that power supplies, such as batteries without the need for startup, can be utilized. This 5 bar back-up pressure should be sufficient to support natural convection removal of 0.5% decay heat which occurs at ~24 hours. The need for active systems/power supplies after the initial 24 hours will not be required.

Furthermore, since decay heat will decrease from 2-3% to 0.5% in this time period, credit can be taken in probability space for loss of active systems during the first 24 hours. The safety approach then becomes a probabilistic risk analysis. Work will therefore require PRA as well as deterministic models. The 5 bar guard containment should be significantly cheaper, and could be either of metal or concrete. Refueling will be carried out at 5 bar, and actions should be taken to rule out double containment bypass. Modeling requirements for this approach would cover both station blackout and refueling incidents.

While the class of depressurized decay heat accidents is a major driver for the modeling requirements, there are other off-normal events that are also of significance. Table 6.1 shows the list of umbrella events. These range from reactivity upsets to secondary side upsets that are also typical of the current LWR fleet within-design-basis. Specific to gas reactor technology are the air ingress events, while local fault propagation is specific to fast reactor designs. Seismic events for these designs have reactivity implications, and a similar reactivity modeling requirement would also hold for the ATWS events.

Table 6.1. Umbrella Events

1. Reactivity Upsets • Control rod ejection • Water Ingress
2. Depressurization (Decrease of Inventory) • Small Leak • Large Leak
3. Increase of Inventory
4. Flow Upsets • Loss of load/turbine trip (direct cycle) • Seized shaft • Turbine deblading (direct cycle)
5. Secondary side upsets
6. Air Ingress
7. Local Faults
8. Seismic Events
9. Station Blackout
10. ATWS
11. Station Blackout + Leak
12. Fuel Handling Incident

6.1 Depressurization BDBA Primary System Model

Work to achieve target GFR core power densities, which result in economic fuel cycle costs, indicates that this range makes it difficult to completely remove core decay heat through passive conduction and radiation to the vessel walls and beyond, at depressurized conditions with loss of power. Models will be required to resolve those technical issues which face the alternative passive approach of reliance on natural convection and/or heavy gas accumulator injection for passive decay heat removal. These models will provide the in-depth analysis capability required to resolve these issues, and include:

1. The transition to, and startup of, natural convection with a period of heavy gas injection, helium depressurization, and discharge through the break with eventual air ingress is an uncertainty that will need to be modeled. A combination of scoping-type models, CFD, one-dimensional integrated system models, literature review with data from experiments, and possible experiments to fill in the data gaps, will need to be utilized to perform the assessments. Delay in setting up quasi-steady natural convection conditions would lead to higher core fuel temperatures. Models of gas mixing will need to be provided for the buoyancy forces.
2. In tandem with experiments, a combination of one-dimensional integrated system models and CFD models are needed to investigate the efficiency of heavy gas injection[1] through the use of accumulators. Distribution and number of injection ports should be modeled in order to assess the feasibility of the scheme. Modeling of a coupled cold/cryogenic system to the primary system should be included. Models for initiating the injection, and two-phase flow issues, should be provided.
3. While it appears that the concept of core internal heat sinks alone would be insufficient to remove initial decay heat (~1-2%), and losses to the vessel through radiation and conduction are insignificant for the target power density, it may be that a combination of mitigation mechanisms that include (a) extended flow coastdown by the single shaft turbo-compressor, with some limited auxiliary forced convection cooling, (b) minimal heavy gas accumulator injection, (c) the in-core heat sinks, plus (d) vessel wall losses could be sufficient to augment the natural convection, and passively cool the core to ~0.5% decay heat. Initially, a set of standalone models for each of these phenomena would be needed to optimize each separately. Then, a set of integrated system models would be needed to evaluate the feasibility of such a combined approach using various combinations and parameters. A scoping reliability model would also be needed to decide on an optimal combination. This implies that if containment leak-tightness were lost in a few days, this combination of alternatives would still suffice to resolve long-term decay heat removal.

6.2 Depressurization BDBA (Guard) Containment System Model

For a depressurization accident period that is longer than a few days, it may be necessary to assure containment leak-tightness, and to provide passive core decay heat removal modes in addition to natural convection of the gas mixture. Containment models are required to provide the technical basis for longer term models when electric power is concurrently unavailable. These include:

1. Discharge of primary system helium, and eventually gas mixture, into the containment atmosphere could lead to jetting, plumes and stratification with a hot layer at the top of the containment. Reliance on natural convection for core decay heat removal requires a significant containment back pressure (~3-4 atmos for CO_2, and ~15 atm for He to remove 1-2% decay heat) to produce the required mass flow densities. Furthermore, containment integrity has to be maintained for 1 to 2 weeks, if natural convection is the

[1] Heavy gas injection is currently being given lower priority.

only passive means of core decay heat removal (~0.01% decay heat removal at 1 atm back pressure). It is important to maintain leaktight containment integrity during this time period. Scoping models, experiments, and a set of CFD models will be needed to assure that the temperature and pressure conditions of the liner, concrete, and penetration seals during this period will not lead to loss of containment integrity. Models will be needed to investigate various system layouts to promote mixing.

2. Preserving containment integrity is but one necessity. In addition, there is a need to transfer the heat out of containment into the ambient atmosphere for an aboveground building. Models will be required of in-containment and ex-containment heat exchangers, and the need for isolation valves. The sizing and location of gas to water heat exchangers and cooling towers, and related system design questions such as heat losses at normal operations, will need to be simulated with scoping-type models to resolve feasibility. Modeling of equipment failure at these extreme conditions will be required. Models for heat transfer through the containment structure, such as that in AP-600, will be needed.
3. An integrated model involving all these separate phenomena will be required, once the standalone models provide design results and decisions. The integrated system model should encompass feedback and interactive effects, and confirm the feasibility of the design.
4. Ultimately, a water dump to flood the core could be used. Such a concept could be based on the SBWR gravity driven cooling system. The SBWR dry well/wet well design would serve as an example of a secondary containment to provide a backpressure, and simultaneously provide the water dump inventory. The dry well concept would require a Passive Containment Cooling Concept (PCCS) model similar for the SBWR. A scoping reliability model would also be needed. Calculations could be performed with standard LWR tools, such as RELAP-5, to determine the accident conditions under which such an approach would be successful. The classical LWR issues of rewet, steam binding, and counter-current flooding would be analyzed. The need for boration would be assessed. Condensation heat removal in the secondary containment would be performed either with TRAC, or with a RELAP-5/CFD combination.

6.3 Reactivity Control and Transient Model

Historically, anticipated transients without scram (ATWS) have played a large role in the safety analysis and licensing discussions with regulatory authorities in the development of the safety case for fast reactors. This has been largely due to the large fissile inventory of fast reactors, as opposed to that in a thermal reactor, and the potential for energetic consequences of melt driven recriticality accident scenarios. With the advent of the ANL IFR concept, it was shown that with proper selection of materials and appropriate design, the neutronic characteristics of fast reactors with pin geometries would lead to benign upsets. Modeling would be needed to translate the IFR experience with pin cores to the innovative fuel element and core restraint design being considered for the GFR (blocks/plate). This would include:

1. Providing models to derive the core expansion/movement coefficients for each type of fuel, reactivity control element, core restraint, and support configuration. These models would be utilized to evaluate the various mechanical designs and materials (structural and fuel) that produce the optimal core expansion/movement coefficients. Other design

possibilities that could be considered are equivalents to the LMR GEM, and control rod driveline expansion.

2. A coupled neutronics/thermal-hydraulic transient system model with ATWS drivers to factor-in individual time constant dependencies would be needed. This would enable a comprehensive study of which passive design mechanisms for reactivity shutdown should be feasible. Modeling options should also include a capability for reactivity changes due to a seismic driver or to vibrations.
3. Models would be needed for possible designs of a backup passive reactivity shutdown system, which could be utilized if the primary rod structure system fails to insert. In addition to neutronics and mechanics models, an accompanying reliability model would also be needed.

6.4 Control Rod Ejection Modeling

With a fairly high coolant pressure (of 10^3 psi), it becomes necessary to consider control rod ejection events in the transient envelope. Models are needed to ascertain whether control rod housing failure will lead to an unacceptable depressurization accident, in conjunction with an unacceptable driving reactivity insertion accident. Given the current positive void coefficient, modeling of design options to minimize the reactivity insertion rate would be needed. Limiting control rod ejection velocities should be the objective. Thermal-hydraulic flow and temperature models are required to carry out the evaluation, mechanical design models are required to ascertain structural integrity, and three dimensional neutronics kinetics modeling should be carried out to provide energy deposition rates in the neighboring fuel element. This should be used in assessing the feasibility of the designs for the different fuel element options.

6.5 Local Flow Blockage Modeling

Historically, due to the high power densities and the ducted subassemblies of fast reactors, local flow blockages were considered important contributors to risk. Models are required to ascertain that the proposed fuel element designs are not susceptible to local fault propagation. A combination of scoping thermal-hydraulic calculations and expert judgment modeling is needed to assess the different fuel element designs. Fuel performance modeling, in particular fission gas behavior and fuel element failure mechanics, will also be required to complete the assessment. Neutronics modeling will be required to characterize the reactivity perturbation and flux shape changes.

6.6 Modeling Data and Experimental Needs

Given the early stage of the design, one of the major issues will be to model the depressurization event with scram, while taking into account all the modes of heat transfer (conduction, convection, radiation and storage), and the action of the specific safety devices. For off-normal conditions, appropriate codes will have to be adapted to helium, and validated against measurements in existing test loops. The top level issues which would be part of the data and experimental needs for future tasks are:

1. Investigation of the start up/transition to natural convection in a low-pressure loop, and the ultimate stability of that decay heat removal mode.
2. Guard containment loading due to thermal jets and plumes during the depressurization event.

7. Modeling and Code Development Costs

Because there are many open questions about the final design, a specific cost estimate is not possible. However, the Gen IV Roadmap estimates a cost of ~$100M for reactor systems work, ~$50M for balance of plant work, ~$150M for GFR safety work, and ~$120M for design and evaluation work, all through 2020. However, these estimates may change based on continuing updates of available data and requirements.

8. References

[1] J. Rouault, T.Y.C. Wei, “Selection of the Concept for Point Design”, I-NERI Project #2001-002-F Report GFR 021, May 2004.

[2] Generation IV Nuclear Energy Systems - Gas-Cooled Fast Reactor (GFR) R&D Program Plan (Draft Rev. 5.1, January 2004).