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GCFR DEVELOPMENT STATUS REPORT

by
PROJECT STAFF

Prepared under
Contract DE-AT03-76SF71023
for the San Francisco Operations Office
Department of Energy

MAY 1980

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GENERAL ATOMIC COMPANY

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ABSTRACT

This report describes the major design features of the gas-cooled fast breeder reactor being developed in the United States principally at General Atomic Company. The report gives the general design strategy and highlights the design features of the reactor core and the nuclear steam supply components. It describes the design results on plant safety and licensing.

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1. GCFR OVERVIEW

1.1. DESIGN STRATEGY

The commercialization strategy for the gas-cooled fast breeder reactor (GCFR) is based upon a three-plant scenario: (1) a demonstration plant, (2) a prototype plant, and (3) a commercial plant. The demonstration plant selected size is a nominal value of 350 MW(e), based upon the desires of the GCFR utility supporters and the current state-of-the-art. A small plant size was selected to minimize capital cost, risk, and licensing problems, but it had to be large enough to provide a meaningful assessment of the concept commercial capability. Minimal component scale-up for the prototype plant was also desired.

The demonstration plant has three main cooling loops and three safety-class auxiliary cooling loops. The number of cooling loops was selected primarily on the basis of safety and licensability. In addition to the main and auxiliary cooling systems, the demonstration plant also has three integral shutdown cooling loops which make use of main loop components.

The circulators for both main and auxiliary cooling loops are driven by electric motors. Main loop electric drive was selected after a detailed comparison with steam turbine drive.

In 1978, General Atomic Company and Kraftwerk Union completed a major study which evaluated the merits of upflow versus downflow core cooling. As a result of this study, upflow core cooling was selected. A major advantage of upflow core cooling is its ability to use natural circulation cooling, which greatly increases the GCFR safety.

Table 1-1 shows the major design parameters of the demonstration plant. These design parameters were selected to meet the conflicting requirements of high reactor performance, low pumping power, and minimum development and capital costs, while taking maximum advantage of the technology developed for the liquid metal fast breeder reactor (LMFBR) and other gas-cooled reactors.

1.2. PLANT LAYOUT

The GCFR demonstration plant contains all the major buildings and facilities required for the reliable operation and control of the reactor and plant during normal and accident conditions. (See Fig. 1-1.) Systems and equipment that must remain functional during and after severe natural phenomena (earthquakes, tornadoes, or floods) are contained in hardened structures. Designed and constructed to conform to applicable regulatory requirements, these hardened buildings are built of either reinforced or prestressed concrete. Additionally, they are supported on a common monolithic foundation base mat of reinforced concrete, designed to meet strict building loading and seismic criteria.

The reactor containment building, located at the center of the plant complex, is a steel-lined, prestressed concrete structure capable of withstanding a postulated depressurization of the reactor. This building houses the prestressed concrete reactor vessel (PCRVR), which contains the reactor core and its associated heat transport systems. The space between the PCRVR and the containment wall is used for steam piping, cable trays, and circulator motor installation and removal. A steel bent structure in this space utilizes intermediate floors to accommodate miscellaneous secondary support systems. Refueling operations are performed at the top of the PCRVR.

TABLE 1-1
GCFR DEMONSTRATION PLANT
MAJOR DESIGN PARAMETERS

Reactor pressure (nominal)	10.5 MPa	(1523 psig)
Maximum cladding temperature	750°C	(1382°F)
Reactor thermal power	1110 MW ^(a)	
Reactor inlet temperature	293°C	(560°F)
Reactor outlet temperature	524°C	(975°F)
Steam temperature at boiler outlet	485°C	(905°F)
Steam pressure at turbine inlet	9.1 MPa	(1450 psig)
Feedwater temperature	171°C	(340°F)

^(a) At 102% of licensed power

(a)

Fig. 1-1. Plant layout: (a) overhead view

(b)

Fig. 1-1. Plant layout: (b) section view

Surrounding this leak-tight reactor containment building is the confinement building. This low leakage building prevents the release of any unfiltered gases to the environment by maintaining a slight negative pressure in the space between it and the containment building.

The reactor service building provides facilities to support fuel handling. It also is hardened and has a controlled atmosphere held at a slight negative pressure. Its seven stories extend from the basement, situated below grade, to the refueling floor, and it contains equipment for handling and storing new and spent fuel. All refueling operations are controlled and monitored from the fuel handling control room located within the reactor service building. Rail car access to the reactor service building is provided at grade level. All incoming new fuel and outgoing spent fuel is handled in the fuel shipping bay, which incorporates facilities and systems for fuel inspection, decontamination, loading, and unloading.

Two more plant buildings are hardened. The control and diesel generator building contains the main control room. This includes the control console, computers and associated equipment, and instrumentation required for reactor and plant operation and maintenance. Adjoining floors above and below the control room are utilized for cable spreading. Adequate spacing provides separation for safety-related power supplies and instrumentation. Three emergency diesel generators, assorted switchgear, and backup battery banks are housed in the subbasement and first floor.

A penetration building, situated between the containment/confinement building and the turbine-generator building, houses safety valving in the steam and feedwater lines. These valves are required for containment isolation and for pressure relief and system interconnections for safe shutdown cooling.

Nonhardened structures compose the balance of the plant. Plant security, health physics, first aid, fire protection, and communication

offices are situated in the administration and access control building. There are also amenities for plant personnel, including lockers, shower and changing room, cafeteria, and rest areas.

The motor control building houses special controls, switchgear, and equipment required to operate the electric motor drives for the helium circulators.

The other major nonhardened building, the turbine-generator building, houses the main turbine generator and associated equipment. The main condenser, feedwater heaters, feedwater storage tank, makeup demineralizer, condensate polishers, deaerating feedwater pumps, and the auxiliary boilers are all placed at optimum locations in the building.

1.3. POWER CONVERSION SYSTEM

The power conversion system of the GCFR demonstration plant employs three basic heat transfer fluid systems along with a conventional steam turbine generator to convert nuclear energy into electricity. (See Fig. 1-2.) All systems and components are designed well within the state-of-the-art to ensure reliable power production. The three heat transfer fluid systems are (1) the primary coolant system using high pressure helium, (2) the steam and feedwater system using high purity water, and (3) the plant heat rejection system using an evaporative cooling tower with a circulating water loop.

The primary coolant system uses motor-driven centrifugal compressors to circulate helium upward through the core where it receives nuclear heat and downward through the steam generators where it transfers heat to the entering feedwater to produce superheated steam for use in the main turbine generator. The primary coolant system and all of its associated components are housed within the PCRV.

Fig. 1-2. GCFR power conversion system

The steam and feedwater system starts with condensate supplied from the main condenser mounted below the main steam turbine. The condenser is equipped with an integral hotwell, which removes oxygen and noncondensibles from the condensed steam and stores condensate to be pumped through a full-flow demineralizer to remove impurities. The feedwater then flows through a series of feedwater heaters, which utilize steam extracted from the main turbine at various bleed points to heat feedwater to a final temperature of 171°C (340°F) before entering the three steam generators. The steam generators are of the once-through type with helical coils. Water flows upward, boils, and becomes superheated steam before exhausting at the top enroute to the main steam turbine generator.

The main turbine generator is a six-flow, tandem-compound, non-reheat, 3600 rpm steam turbine nominally rated at 375 MW(e) with inlet steam at 9.1 MPa (1450 psig) and 485°C (905°F). The electric generator is a totally-enclosed, hydrogen-cooled, three-phase machine delivering electrical energy at 22,000 V to a stepup transformer, which in turn, feeds high voltage power through a switchyard to the transmission lines.

The plant heat rejection system includes a circulating water system, an evaporative cooling tower, a water makeup system, and a blowdown system. The heat to be rejected from the plant comes from the main condenser and various plant service water systems which supply component cooling. In the circulating water system, the circulating water pumps take cool water from the tower basin and discharge it to the main condenser and other component cooling heat exchangers. The heated water returns to the upper part of the cooling tower fill and cools as it cascades through the air flow to the tower basin. Makeup to the circulating water system to replenish water lost due to evaporation, drift, and blowdown is provided by nearby wells. To provide for chemistry control of the circulating

water and to prevent buildup of dissolved solids, a continuous blowdown system with appropriate treatment of effluent to meet environmental standards is provided.

1.4. NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The reactor and its associated reactor coolant circuit components are shown in Fig. 1-3, and typical system parameters are given in Table 1-2. The entire primary coolant system is contained within the PCRV, which is a multicavity pressure vessel reinforced with steel rods and prestressed by a system of longitudinal tendons and circumferential wire wrappings. The reactor core is located in the central cavity; peripheral cavities surrounding the central cavity contain heat exchanger and helium circulation equipment. The peripheral cavities are interconnected with the central cavity by cross ducts. All PCRV interior surfaces are lined with leak-tight steel liners which contain the primary coolant. These, in turn, are lined with thermal barriers to protect the PCRV from the high temperatures of the helium coolant. The limited quantity of heat that passes through the thermal barrier is removed by a liner cooling water system embedded in the PCRV.

The reactor coolant system consists of three main loops, each with an independent steam generator, a horizontally mounted electric-motor-driven circulator, and a gravity closing isolation valve. Three core auxiliary cooling loops are also provided, each having a vertically mounted electrically driven circulator, heat exchanger, and gravity opening isolation valve.

Fig. 1-3. GCFR demonstration plant NSSS

TABLE 1-2
TYPICAL NSSS PARAMETERS

Overall Plant

Thermal power	1090 MW(t)
Net electric power	360 MW(e)
Net plant efficiency	33%
No. of main loops	3

Reactor

Fuel material	(P,U)O ₂
Blanket material	UO ₂
Reactor inlet temperature	298°C (568°F)
Reactor outlet temperature	520°C (975°F)
Helium pressure	10.5 MPa (1520 psi)
Reload interval	Annual

Power Conversion System

Superheater outlet temperature	485°C (905°F)
Superheater outlet pressure	10.7 MPa (1550 psi)
Feedwater temperature	171°C (340°F)
Turbine inlet pressure	10.1 MPa (1450 psi)
Turbine inlet temperature	482°C (900°F)

2. REACTOR CORE DESIGN

2.1. REACTOR CORE CAVITY ARRANGEMENT

The general arrangement of the reactor core cavity is shown in Fig. 2-1. The active core is located in the lower portion of the cavity, supported by a bottom-mounted grid plate in the lower inlet coolant plenum. The grid plate is supported by a skirt attached to a thermal sleeve anchored to the PCRV cavity wall. The grid plate and support skirt are designed to be fabricated from Type 316 stainless steel.

The core cavity closure is provided by a removable plug located at the top of the cavity. The control and shutdown drive mechanisms are located in penetrations provided in the closure plug. Control-rod-guide structures and instrument trees are provided between the closure plug and the top of the reactor core.

Upper plenum, radial, and lower plenum shielding is provided in the core cavity to protect the reactor internals and the PCRV liner from excessive irradiation. The upper and lower plenum shielding has flow-through passages for the helium coolant, which enters the lower plenum and exits in the upper plenum.

2.2. REACTOR CORE

The general configuration of the reactor core is illustrated in Fig. 2-2. The active core region consists of 169 hexagonal assemblies; 150 of these are fuel assemblies, 15 are control assemblies, and 4 are

Fig. 2-1. GCFR configuration with flowthrough shielding

Fig. 2-2. GCFR core, demonstration plant, general configuration

secondary shutdown assemblies. The central core region is surrounded by a radial blanket region of fertile material. The radial blanket consists of 162 hexagonal blanket assemblies arranged in three concentric rows around the active core. Fertile blanket material is also included in the fuel assemblies, above and below the active core (upper and lower axial blankets). The radial blanket is surrounded by 138 hexagonal reflector/shield assemblies arranged in two concentric rows to protect the core restraint.

The core assemblies are axially supported at their lower ends by the bottom-mounted grid support plate. The assemblies are supported laterally by a core restraint mounted on the grid plate, which consists of a cylindrical support barrel and core formers surrounding the core. The core restraint is designed to provide lateral support at two elevations: (1) at the top of the active core and (2) at the top of the core assemblies. This core restraint system has been adopted from the technology developed in the LMFBR program and is designed to minimize reactivity insertion due to core distortions. (See Fig. 2-2.)

The control and shutdown rod assemblies are located on the centers of regions that consist of a central control (or shutdown) assembly surrounded by six fuel assemblies. Removal and replacement of assemblies during refueling is accomplished through the penetrations in the closure plug provided for the drive mechanisms. Each seven-assembly region is serviced by one of the penetrations. In the regions outboard of the control and shutdown assembly locations, additional penetrations are provided in the closure plug to service the blanket and reflector/shield assemblies.

2.3. FUEL ASSEMBLY

The GCFR fuel assembly design is similar to designs employed in LMFBR programs, particularly the fuel rod design. The fuel rod

design employs the same materials and has similar geometry and operating conditions as those being developed in the international LMFBR programs. The unique characteristics incorporated in the GCFR fuel assembly design include the following:

1. Roughened fuel rod cladding to enhance heat transfer to the helium coolant.
2. Pressure-equalized and vented fuel rods, which essentially eliminate any pressure-induced stresses on the cladding from either the primary coolant system pressure or from fission gases generated within the rods.
3. A large fuel rod pitch-to-diameter ratio relative to that commonly employed in the LMFBR designs.

The design of the fuel assembly is shown in Fig. 2-3, and design configuration parameters are given in Table 2-1. The fuel assembly is hexagonally shaped, 201 mm (7.91 in.) across flats, and 500 mm (19.69 in.) long. A cylindrical inlet nozzle, 172 mm (6.77 in.) diameter by 675 mm (26.57 in.) long, is located at the bottom end. The inlet nozzle fits into a mating hole in the core support grid plate. The assembly rests on the conical surface which forms the transition between the cylindrical and hexagonal shapes.

Contained within the assembly is a 2870-mm (112.99-in.) long fuel bundle consisting of 265 fuel rods. The fuel rods are 8 mm (0.315 in.) in diameter and are located on an 11.5-mm (0.453-in.) triangular pitch. The rods are fastened to a grid manifold at their lower ends and are laterally spaced by 12 spacer grids positioned at selected axial locations along the length of the rods. The spacer grids are retained in position by means of six hanger rods. The hanger rods are structural members containing no fuel material.

TABLE 2-1
FUEL ASSEMBLY MECHANICAL DESIGN PARAMETERS

Fuel Assembly

Assembly length	5000 mm	(196.85 in.)
Active core height	1200 mm	(47.24 in.)
No. of fuel rods/assembly	265	
No. of hanger rods	6	
Fuel rod spacer type	Spacer grid	
No. of spacer grids	12	
Duct material	D9	
Duct across flats i.d.	193.0 mm	(7.6 in.)
Duct across flats o.d.	201.0 mm	(7.91 in.)
Duct wall thickness	4.0 mm	(0.16 in.)
Fuel rod pitch	11.5 mm	(0.45 in.)
Rod-to-rod gap	3.5 mm	(0.14 in.)
Rod-to-duct gap	2.87 mm	(0.11 in.)
Rod-to-duct gap	82% of rod-to-gap	
Fuel assembly pitch	208.5 mm	(8.21 in.)
Total estimated assembly weight	590 kg	(1301 lb)

Fuel Rod

Length	2870 mm	(112.99 in.)
Clad material	D9	
Outside diameter	8.00 mm	(0.32 in.)
Outside root diameter	7.74 mm	(0.30 in.)
Average volumetric diameter in rough section	7.815 mm	(0.308 in.)
Inside diameter	6.98 mm	(0.27 in.)
Clad root thickness	0.38 mm	(0.002 in.)
Fuel and axial blanket pellet o.d.	6.84 mm	(0.27 in.)
Fuel and axial blanket pellet-to-clad diametral gap	0.14 mm	(0.005 in.)
Fuel column length	1200 mm	(47.24 in.)
Fuel material	(U,Pu)O ₂	
Fuel material planar smear density	85.5% of theoretical	
Axial blanket column length (each)	600 mm	(23.62 in.)
Axial blanket material	Depleted UO ₂	
Axial blanket material planar smear density	90.0% of theoretical	

Roughness Geometry

Rib shape	Trapezoidal	
Rib height	0.13 mm	(0.005 in.)
Rib width	0.45 mm	(0.018 in.)
Rib pitch	1.56 mm	(0.06 in.)
Length of roughening	1200 mm	(48.03 in.)

Fig. 2-3. Fuel assembly, general arrangement

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A hexagonal flow duct is provided around the rod bundle for channeling the coolant flow through the rod bundle.

Grid plate shielding is provided beneath the rod bundle between the grid manifold and the grid plate to protect the grid plate from excessive irradiation. Additional grid plate shielding is provided within the fuel assembly inlet nozzle adjacent to the grid plate upper surface. Exit end shielding is provided above the rod bundle to prevent excessive irradiation of the instrument trees and control rod guide structures located in the reactor cavity upper plenum.

An exit end nozzle is provided on the top end of the assembly. The exit nozzle is used for handling the assembly and contains a fixed-area replaceable orifice. The orifice is designed to be changed at refueling outages.

Each fuel rod (see Fig. 2-4) consists of a cladding tube, a fuel stack, upper and lower axial blankets, a holddown spring, a fission product trap, and end plugs. The cladding tube is roughened with trapezoidally shaped ribs over the fuel region to enhance heat transfer to the helium coolant. The fuel region consists of a 1200-mm (47.24-in.) long stack of mixed oxide (U,Pu) O_2 cylindrical pellets with a fissile plutonium enrichment of ~20%. The upper and lower axial blankets are located above and below the fuel region. Each axial blanket consists of a 600-mm (23.62-in.) long stack of depleted UO_2 cylindrical pellets.

A pellet holddown spring, located above the upper axial blanket stack, maintains proper positioning of the pellet stacks during handling and shipping prior to irradiation. A special lower end plug vents the fuel rods. The lower end plug contains an axial vent hole to pass the fission gases out of the rod and threads to attach and seal the rod to the grid manifold. A charcoal fission

Fig. 2-4. GCFR core, fuel rod assembly

product trap is assembled as an integral part of the lower end plug. The charcoal trap retains volatile fission products and delays gaseous fission products to permit radioactive decay. Retaining screens are provided at the lower end of the trap to prevent charcoal particles or fuel fragments from passing into the end-plug vent hole.

The rods are screwed into mating holes in the grid manifold where a metal-to-metal seal is made. The grid manifold structure is cross-drilled through the grid ligaments such that there are at least two flow paths from each rod. Redundant flow paths are also provided from the grid manifold to an assembly charcoal trap located in the assembly inlet nozzle below the grid plate shielding for further delay and decay. From the assembly fission product trap, the fission gases pass to two vent connections located at the transition between the hexagonal flow duct and the cylindrical inlet nozzle. The fission gases are swept from the vent connection to the reactor helium purification system for separation and storage.

A schematic of the pressure equalization system (PES) is shown in Fig. 2-5. A suction hole is provided at the PES vent connection such that a small quantity of coolant at core exit conditions sweeps the fission gases diffusing out of the fuel assembly to the helium purification system. As the result of the suction hole being at reactor core exit pressure conditions, the pressure within the fuel rods is maintained at core exit conditions. The only pressure differential acting on the cladding is the core pressure drop of ~ 180 kPa (26 psi). In addition, any cladding breaches will result in coolant flow into the rod, which will limit the release of radioactivity to the coolant.

2.4. RADIAL BLANKET ASSEMBLY

The general arrangement of the radial blanket assembly is shown in Fig. 2-7, and design configuration parameters are given in Table 2-2.

Fig. 2-5. Schematic diagram of GCFR PES

Fig. 2-6. Blanket assembly, general arrangement

Fig. 2-7. Control assembly, general arrangement

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TABLE 2-2
BLANKET ASSEMBLY MECHANICAL DESIGN PARAMETERS

Radial Blanket Assembly

No. of radial blanket assemblies/core	162	
Assembly length	5000 mm	(196.85 in.)
No. of blanket rods/assembly	61	
Blanket rod spacer system	Wire wrap	
Across flats i.d.	193.0 mm	(7.60 in.)
Across flats o.d.	201.0 mm	(7.91 in.)
Duct wall thickness	4.0 mm	(0.16 in.)
Blanket rod pitch	24.1 mm	(0.95 in.)
Total estimated assembly weight	875 kg	(1929 lb)

Blanket Rod

Length	2880 mm	(113.39 in.)
Clad material	D9	
Outside diameter	22.20 mm	(0.87 in.)
Inside diameter	21.20 mm	(0.83 in.)
Wire wrap diameter	1.80 mm	(0.07 in.)
Wire wrap pitch	300.0 mm	(11.81 in.)
Blanket material	Depleted UO ₂	
Blanket material smear density	90.0% of theoretical	
Blanket pellet o.d.	21.05 mm	(0.83 in.)
Blanket material axial length	2400 mm	(94.49 in.)

The blanket assembly design is essentially the same as the fuel assembly, except that the blanket rod bundle consists of a smaller number of larger rods. Other than the rod bundle, the radial blanket assembly is designed to use the same major components as the fuel assembly. These major components include the inlet nozzle, grid plate shielding, assembly charcoal trap, exit shielding, flow duct, and exit nozzle. The fixed-area replaceable orifice would be specific to the blanket assembly and would contain mechanical discrimination characteristics to prevent a blanket orifice from being assembled on a fuel assembly.

The blanket rod bundle consists of 61 rods, 22.2 mm (0.87 in.) o.d. by 2880 mm (113.39 in.) long. Each rod consists of a cladding tube, a stack of depleted UO_2 cylindrical pellets, a pellet holddown spring at the top of the pellet stack, and end plugs. The blanket rods are pressure equalized and vented the same as the fuel rods except that, due to the low fission product yield, a fission product trap is not included in each blanket rod. An assembly trap is, however, included in the inlet nozzle region.

The blanket rods are arranged on a much smaller pitch-to-diameter ratio than the fuel rods. As a result, a wire-wrap spacer system, rather than spacer grids, as used in the fuel assembly design, is used to maintain the rod array.

2.5. CONTROL AND SHUTDOWN ASSEMBLIES

The design of the control assembly is shown in Fig. 2-7. The shutdown assembly design, although not yet decided, is expected to be similar to the control assembly, except for poison material worth. The design configuration parameters of the control and shutdown assemblies are given in Table 2-3.

TABLE 2-3
CONTROL AND SHUTDOWN ASSEMBLY MECHANICAL DESIGN PARAMETERS

Control Assembly

No. of dedicated control assemblies/core	15	
Assembly length	5000 mm	(196.85 in.)
Control rod stroke	1350 mm	(53.15 in.)
No. of control pins/assembly	43	
Control bundle configuration	Circular	
Control pin spacer system	Wire wrap	
Duct material	D9	
Duct across flats i.d.	193.0 mm	(7.60 in.)
Duct across flats o.d.	201.0 mm	(7.91 in.)
Duct wall thickness	4.0 mm	(0.16 in.)
Control rod guide duct i.d.	167.1 mm	(6.58 in.)
Control rod guide duct o.d.	175.1	(6.89 in.)
Control rod guide duct wall thickness	4.0 mm	(0.16 in.)
Control pin shroud i.d.	153.1 mm	(6.03 in.)
Control pin shroud o.d.	161.1 mm	(6.34 in.)
Control pin shroud wall thickness	4.0 mm	(0.16 in.)
Control pin pitch	19.1 mm	(0.75 in.)
Pin-to-pin gap	1.6 mm	(0.06 in.)
Pin-to-shroud gap	1.6 mm	(0.06 in.)

Control Pins

Length	1300 mm	(51.18 in.)
Outside diameter	17.50 mm	(0.69 in.)
Inside diameter	16.50 mm	(0.65 in.)
Clad thickness	0.5 mm	(0.02 in.)
Wire wrap diameter	1.50 mm	(0.06 in.)
Pellet o.d.	16.30 mm	(0.64 in.)
Poison material axial length	1200 mm	(47.24 in.)

Shutdown Assembly

No. of shutdown assemblies/core	4	
Length	5000 mm	(196.85 in.)

The control assembly design has the same external configuration as the fuel and blanket assembly design, and many of the major components are the same or are slight modifications. The flow duct and inlet nozzle structures are the same as used in the fuel and blanket assemblies, and the grid plate shielding and exit nozzle design are slight modifications.

The control rod consists of a bundle of control pins contained within a guide duct having upper and lower end fittings. The composite of the guide duct, control pins, and end fittings is designed to move up and down in the flow duct, but it always remains wholly contained within the flow duct.

The control pin bundle consists of 43 pins, 17.5 mm (0.69 in.) o.d. by 1300 mm (51.18 in.) long. Each pin consists of a cladding tube, a stack of B_4C poison pellets, a pellet holddown spring, and end plugs. The control pins are attached to the upper end fitting and are laterally spaced by wire wrapping. Twelve tie rods are provided at the bundle hexagonal corners. The control pins are vented directly to the helium coolant.

A lower end fitting is provided on the lower end of the guide duct to prevent any of the control pins from moving out of the poison cluster should a failure occur in the poison pin cladding or in the pin attachment to the upper end fitting. A backup stop/absorber is provided in the lower portion of the assembly beneath the poison cluster to prevent the cluster from moving below the active core zone should a failure occur anywhere in the withdrawal shafting in the region between the poison cluster and the primary stop, which comes to rest in the exit nozzle.

2.6. REFLECTOR/SHIELD ASSEMBLY

The reflector/shield design configuration parameters are given in Table 2-4. The reflector/shield has the same external configuration as the balance of the core assemblies and is essentially identical to the radial blanket assembly, except that the reflector/shield assembly has a smaller number of larger rods and does not have a fission product trap in the inlet nozzle or grid-plate shielding.

The reflector/shield rod bundle consists of 19 rods, 40.1 mm (1.58 in.) o.d. by 3180 mm (125.20 in.) long. Each rod consists of a cladding tube, a boronated graphite column, and end plugs. The rods are attached to a grid at their lower end and are laterally spaced by wire wraps. The reflector/shield rods are vented directly to the helium coolant.

2.7. GRID PLATE

The function of the grid plate is to support the core including the fuel blanket and control, shutdown, and shielding assemblies and to maintain the geometry and spatial relationship of these elements within acceptable limits to ensure core cooling and reactivity control. The grid plate consists of a circular disk [5346.7-mm (210.5-in.) diameter by 508-mm (20-in.) thickness] with a hexagon pattern of circular penetrations for mounting and for conducting coolant flow into each assembly. A nonperforated area at the outer edge of the grid plate acts as a stiffening ring. The grid plate is forged in one piece from 316 stainless steel.

A right circular cylinder (core barrel) also made of 316 stainless steel, provides horizontal restraint, and hence, maintains the spatial relationship of all grid-plate-mounted assemblies. This core barrel is welded into the grid plate at the solid peripheral ring.

TABLE 2-4
REFLECTOR/SHIELD ASSEMBLY MECHANICAL DESIGN PARAMETERS

Reflector/Shield Assembly

No. of reflector/shield assemblies	138	
Assembly length	5000 mm	(196.85 in.)
No. of shielding rods/assembly	19	
Shielding rod spacer system	Wire wrap	
Across flats i.d.	193.0 mm	(7.60 in.)
Across flats o.d.	201.0 mm	(7.91 in.)
Duct wall thickness	4.0 mm	(0.16 in.)
Shielding rod pitch	42.9 mm	(1.69 in.)
Total estimated assembly weight	400 kg	(882 lb)

Shielding Rod

Clad material	D9	
Outside diameter	41.10 mm	(1.62 in.)
Inside diameter	40.10 mm	(1.58 in.)
Wire wrap diameter	1.60 mm	(0.06 in.)
Shielding material	Boronated graphite	
Shielding pellet o.d.	40.0 mm	(1.57 in.)

Hexagonal structural frames within the core barrel at two elevations provide lateral restraint for the core. The two frame elevations are (1) just above the active core, and (2) at the upper extremity of the core elements.

2.8. SUPPORT STRUCTURE

The primary function of the support structure is to transfer all deadweight, pressure, and seismic core and grid plate loads to the liner. The support structure also serves the auxiliary functions of separating the core inlet and outlet gas plenums, controlling the core bypass flow, and acting as a thermal sleeve to control heat flow to the PCR. Bypass flow is primarily required for temperature control of the outer circumference of the grid plate and adjacent support structures.

This structure has the approximate shape of a frustrum of an inverted cone. The lower edge is welded to the outer diameter of the grid plate. The upper edge has a flange which rests on a similar PCR mating flange. All core support vertical and horizontal loads are transmitted through this flange connection. Cool inlet gas bypasses the core through a series of small holes in this support structure adjacent to the peripheral ring of the grid plate.

2.9. MATRIX CONNECTION PLATE

The matrix connection plate is a component of the PES. Plate material is 316 stainless steel. This plate lies between the core assemblies and the grid plate. The gas stream from each assembly element flows through this plate and the intrasystem tubing to radiation monitoring stations located outside the PCR.

2.10. UPPER PLENUM SHIELDING

The primary functions of the shielding assemblies, such as the upper plenum, radial and lower plenum assemblies, are to limit (1) radiation-induced damage to the PCRV liner, (2) nuclear heating of the PCRV concrete, and (3) neutron activation of components requiring contact maintenance. The upper plenum shielding has the additional functions of limiting radiation leakage from the core cavity and minimizing pressure drop of the core outlet coolant stream. The configuration chosen to meet these functional requirements and design constraints consists basically of 316 stainless steel shells of various thickness filled with slabs of graphite and boronated graphite.

The cavity closure portion of the upper plenum shielding consists of a 410-mm (16.14-in.) thick disk with perforations for the control rod drives.

The radial portions of the upper plenum shielding are supported from the liner sidewall by struts. Differential thermal expansion is provided by sliding constraint at the lower shield supports. Pressure equilization capability is provided for each shield element to prevent damage in the event of a design basis depressurization accident (DBDA).

2.11. RADIAL SHIELDING

Radial shielding functions are identical with those of the upper plenum shielding regarding limiting radiation-induced damage to the PCRV liner and nuclear heating of the PCRV concrete. Other functions of the upper plenum shielding do not apply to the radial shielding. The radial shielding elements are similar in form and in composition to the beam elements of the upper plenum shielding, except that the radial elements are designed to have minimum clearance between them at operating temperature. The support method for the radial shielding is similar to that of the upper plenum beam elements.

2.12. LOWER PLENUM SHIELDING

In addition to limiting radiation and nuclear heating of the PCRV, the lower plenum shielding provides backup support for the grid plate (and core) in the event of failure of the grid plate support structure. Adequate coolant flow and core reactivity must be maintained during this event. The radial sections of the lower plenum shielding also have provision for coolant flow as in the upper plenum radial section. This lower plenum shield rests on the molten fuel containment system (MFCS) floor.

2.13. MOLTEN FUEL CONTAINMENT SYSTEM (MFCS)

The lower plenum of the core cavity is especially designed to resist the thermal loadings attendant to a whole core melt. This is accomplished by the use of a ceramic crucible with a mixture of steel and heavy metal to act as sacrificial material and for dilution of the melt. Increased cooling around the cavity by the liner cooling system is also provided to remove the heat during the early part of the hypothesized accident.

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3. NUCLEAR STEAM SUPPLY SYSTEM (NSSS) COMPONENTS

3.1. MAIN CIRCULATORS AND DRIVES

The design of the helium circulator system provides for maximum component life and dependable emergency operation. Each of the three main helium circulators is rigidly mounted with closures at the bottom of the PCRV, below the steam generator cavities. They are driven by 13.88-MW (18,600-hp) electric motors horizontally located outside the PCRV. Helium leaving the steam generator enters the circulator and is accelerated by a centrifugal impeller, passes into a series of pipe diffusers, then discharges into a plenum surrounding the circulator. Helium from this plenum then flows through the cold duct to the core inlet plenum.

Water-lubricated journal bearings and controlled leakage seals are used in the main circulators. This type of seal provides for hydrostatic operation, desirable for maximum life. Water is chosen for the lubricant in preference to oil because of its simple shaft sealing systems. Any leakage would not contaminate the primary cooling system. A conventional oil-lubricated thrust bearing has been located in each electric motor.

The use of synchronous electric motors to drive the single-stage radial flow compressors simplifies testing of the circulator and preoperational testing in the reactor by using readily available electric power. A variable frequency (thyristor) controller provides for variable speed operation of the main circulator to provide the required cooling flow between 20% and 100% reactor power. In addition to normal full-load and part-load operation, the main circulators

are capable of performing under emergency conditions. Independently powered pony motors provide circulation for decay heat removal. Figure 3-1 shows a layout of the main circulator and its drive motor.

3.2. STEAM GENERATORS

The steam generators in the GCFR demonstration plant are based upon design features similar to those already operating in the Fort St. Vrain plant near Denver, Colorado, and are expected to provide highly reliable operation. Since helium rather than water is the primary coolant in the GCFR, erosion, corrosion, and denting problems currently being experienced in other steam generator designs are expected to be minimized.

The GCFR steam generators are located in cavities in the PCRV. Hot helium enters at the top of the steam generator, flows downward on the shell side of the tube bundle, gives up heat to water and steam, and exits at the bottom. The water enters through a feed-water tubesheet located at the bottom, flows upward inside the tubes, and exits at the top through a superheater tubesheet. (See Fig. 3-2.)

The steam generator is supported at the top by a flanged connection to the PCRV liner and is provided with a redundant support structure and a lateral seismic restraint at the bottom. A flow distribution device at the inlet ensures uniform helium flow. Helical coiled tubes, which make up the heat transfer section of the steam generator, are threaded through perforated tube support plates which transmit dead-weight and seismic loads out to the supporting shrouds. The heat transfer section of each steam generator, which is 3810 mm (150 in.) in diameter and 8229.6 mm (324 in.) long, is made up of 420 25.4-mm (1-in.) diameter tubes. Expansion loops on both ends of the bundle are provided to accommodate differential thermal expansion between the helical tube bundle and the fixed tube sheets.

Fig. 3-1. GCFR main circulator

Fig. 3-2. GCFR top-supported steam generator, general arrangement

The entire steam generator assembly can accommodate in-service inspection and in-place tube plugging, and has the capability to be removed and reinstalled if required.

3.3. CORE AUXILIARY COOLING SYSTEM (CACS)

In addition to the main cooling loops, the demonstration plant also has three safety-class auxiliary cooling loops. Each auxiliary loop contains a circulator, a heat exchanger, and an isolation valve. Figure 3-3 shows a schematic of the CACS.

3.3.1. Auxiliary Circulator

The auxiliary circulators provide helium circulation within each of the auxiliary cooling loops. Each auxiliary circulator assembly is installed from beneath the PCRV in a vertical orientation and is bolted to a penetration flange within the PCRV. The auxiliary loop isolation valve is installed as a separate assembly prior to insertion of the auxiliary circulator.

The auxiliary circulator employs a centrifugal impeller and is driven by a variable-speed squirrel-cage induction motor, which is located inside the penetration closure in cool and purified high-pressure helium. Oil lubricated bearings are used with shaft-mounted oil pumps. The auxiliary circulator motor drive, rated at 746 kW (1000 hp) and 3600 rpm, is capable of providing ample core cooling during the DBDA.

The auxiliary loop isolation valve prevents backflow through an auxiliary loop when the main cooling loops are operating. The valve is a split butterfly configuration with two semi-elliptical flappers actuated by the differential pressure produced by gas flow or by gravity for natural convection cooling using the CACS. Figure 3-4 shows a cutaway of the auxiliary circulator and isolation valve.

Fig. 3-3. CACS

Fig. 3-4. GCFR auxiliary circulator

3.3.2. Core Auxiliary Heat Exchanger (CAHE)

The CAHE is a helically coiled heat exchanger mounted vertically in the PCR. During residual heat removal (RHR) with the CACS, hot helium enters at the top of the CAHE bundle, flows down between the tubes, and exits at the bottom. Cold water enters through the lower tubesheet, flows upward inside the tubes, and exits through the top tubesheet. The water within the tubes is pressurized to 8.96 MPa (1300 psi) to ensure adequate subcooling for the spectrum of design basis events for the CACS. Each CAHE connects to a separate air-to-water heat exchanger located outside the containment confinement building for ultimate heat rejection.

The CAHE tube bundle is contained within two concentric shrouds and is provided with expansion loops at each end to accommodate differential expansion between the bundle and the fixed tubesheets. A flow distribution device is provided at the inlet of the bundle to maintain uniform helium flow. The CAHE is designed to perform under forced flow as well as natural convection helium flow conditions. Since the CAHE is a safety component, the design provides for in-service inspection and can be removed for detail inspection or repair if required.

Figure 3-5 shows a layout of the CAHE.

3.4. FUEL HANDLING

Refueling operations are predicated on a 3-yr core life, whereby one-third of the reactor core is replaced each year with new fuel. Replaceable shield assemblies that reside adjacent to the blanket regions may be replaced if required. All reactor core assemblies are transferred through refueling penetrations in the

Fig. 3-5. GCFR CAHE helical bundle

core cavity closure of the PCRV. These penetrations contain either control rod drives or instrument trees which must be removed from the appropriate penetration prior to installation of the fuel handling machine.

Each refueling region consists of seven fuel, blanket, or shield assemblies, any of which can be handled by a fuel handling machine in the penetration immediately above the region.

In this refueling system, the relative functions of the fuel handling machine and the fuel transfer cask are interrelated in that core assemblies are handled by one grapple only up to the point where they are transferred from the fuel transfer cask to the transfer lock.

The fuel handling machine performs the function of transferring assemblies between the reactor core and the plenum. This part of the sequence provides axial forces in both directions sufficient to overcome the maximum anticipated friction forces between core assemblies. From the plenum, assemblies are drawn upwards into the fuel transfer cask by the fuel transfer cask grapple.

The fuel transfer cask moves with the spent fuel assembly to the spent fuel chute. The fuel assembly is discharged from the transfer lock through the inclined chute into the fuel storage pool (see Fig. 3-6). New fuel is placed in the transfer cask within the containment building and inserted into the core using a reversed sequence.

3.5. CONTROL ROD DRIVE MECHANISMS (CRDMs)

The complete CRDM assemblies are sealed within the pressure boundary. Electrical connections penetrate the individual reactor nozzles utilizing hermetically sealed fittings.

Fig. 3-6. GCFR demonstration plant, general arrangement, section view

The CRDMs control the reactivity of the GCFR by positioning selected control rods in or out of the active core in response to signals from the plant control system.

When current is applied to the stator by the control system, a magnetic field is created around the segment arms at the upper end of the rotor assembly. The field draws the upper end of the segment arms outward, rotating them on the segment arm pivot pins; as the upper end is drawn outward, the lower end moves inward, causing the teeth on the four roller assemblies at the bottom end of the rotor assembly to engage with the threads on the leadscrew. When the current is applied to the motor phase in a programmed sequence, the rotor assembly rotates and the leadscrew is translated in either the IN or OUT direction, depending upon the sequence of the applied pulses. When stator power is interrupted, the segment arm springs force the lower end of the segment arms to pivot radially outward, releasing the leadscrew and allowing it to travel to the full IN position. Scram is initiated by opening the scram breaker in the controller which removes the electrical power from the stator.

3.6. PLANT CONTROL SYSTEM

The control system developed for the GCFR demonstration plant will provide stable manual or automatic control of the plant over the 25% to 100% load range for normal electrical power generation on base load, load following, and reactor following modes. The objectives of the plant control system are (1) to maintain constant main steam temperature and pressure, (2) to regulate reactor power to sustain plant output, and (3) to balance outlet steam temperatures from the three steam generators.

The control system is based on a reactor-follow-turbine load scheme. For a load change, a turbine load index is changed either

by the operator or by the area load dispatch signal. This causes a change in the turbine first-stage pressure. A signal derived from first-stage pressure is used as a feedforward signal to change feedwater flow and reactor power. The change in feedwater flow is used as a feedforward signal to change circulator speed. Load change control is achieved by using the reactor control rods to control reactor power and main steam temperature, by adjusting the feedwater pump drive turbine speed to control feedwater flow and main steam pressure and by varying the speed of each helium circulator motor to maintain helium flow proportional to feedwater flow. Conditions within the three steam generators are balanced by individual circulator speed and feedwater flow trim. The flux demand signal derived from the turbine load indicator and the measured steam temperatures is limited to prevent excessive changes.

With this method of control, the main steam temperature and pressure and the helium temperatures remain essentially constant over the entire load range. Also, the main steam flow rate, the helium flow rate, and the reactor power level change essentially in proportion to the load demand.

Figure 3-7 shows a schematic of the plant control system.

3.7. HELIUM SERVICE SYSTEM

The helium service system consists of three functional systems:

1. The helium purification system is a closed loop system which removes primary coolant helium from the PCRV, then returns it essentially free of activity and chemical impurities.
2. The PCRV seal system is a static pressurization system for sealing PCRV penetration closures. It also provides a means for leak testing the PCRV closures.

Fig. 3-7. Plant control system

3. The PES equalizes the static gas pressure across the cladding of the fuel and blanket rods within the fuel/blanket assemblies. It also collects the vented gaseous and volatile fission products in the core assemblies and directs them to the helium purification system for processing.

3.7.1. Helium Purification System

The helium purification system removes particulates and chemical impurities from the primary coolant loop and PES helium. The resulting gas can safely be used as a clean gas purge where needed throughout the plant. The system returns the purified helium to the primary coolant loop as a purge gas for selected PCRV penetrations, the main helium circulators, and the auxiliary helium circulators. It also compresses helium recycled from the main and auxiliary circulator service systems and purifies helium from the primary coolant loop before the gas is transferred to the helium storage system. The helium purification system serves as the primary control on the chemical impurity level in the primary coolant. The normal flow requirements for purified helium from the helium purification system are established by the various clean helium purge requirements throughout the reactor plant and the PES sweep gas requirements.

3.7.2. PCRV Seal System

The PCRV seal system comprises the seal services to selective PCRV penetration closures and the piping, valves, and instrumentation outside of these penetrations. The system provides a means to detect penetration closure seal leakage. It also includes optional purge services to selective penetrations.

PCRv penetration closures are secured with a pair of elastomer seals, an inner one preventing PCRv leakage and an outer one preventing leakage to the ambient containment building. The seal system provides a means to depressurize seal interspaces for maintenance and/or PCRv depressurized conditions. The seal system also continuously monitors the seal integrity and provides a means to leak test newly installed seals.

3.7.3. Pressure Equalization System (PES)

The PES consists of piping, valving, instrumentation, and controls necessary to balance pressure across the rod cladding of core assemblies, transfer discharges from the core assemblies to the helium purification system, detect activity leakage from core assemblies, locate such leaking assemblies, and monitor changes in such leakage.

The PES equalizes the gas pressure inside the cladding of the fuel and blanket rods of the core assemblies with the pressure of the reactor coolant external to the rods during all normal and abnormal operating conditions. Such pressure balancing necessitates periodic helium transfer to and from the core assemblies. Accordingly, helium ingress is drawn from the coolant helium, and helium discharge is routed directly to the helium purification system.

The PES is designed to balance pressure between the core assemblies and the coolant helium. This system provides for sweep-gas (coolant helium) ingress at the vents of the core assemblies to transfer vented gaseous fission products to the helium purification system.

4. SAFETY RELATED FEATURES

There are three separate circuits for residual heat removal (RHR) in the GCFR:

1. Steam bypass to the condenser using the normal power conversion heat transfer system components.
2. Operation of the three shutdown cooling system (SCS) loops which make use of the steam generators and main helium circulators (driven by pony motors) with safety-class feed and heat dump components.
3. Operation of the three CACS loops which employ separate safety-class circulators, heat exchangers, and heat dump circuits. The latter system is also designed to naturally convect residual heat from the core all the way to the ultimate heat sink, ambient air.

4.1. MAIN LOOP COOLING SYSTEM (MLCS)

The MLCS goes first to the RHR mode when the main loops continue operation after reactor shutdown. Following a reactor trip, the main circulator speed is reduced to a shutdown cooling level (typically 15%), and the main turbine-generator is tripped, diverting the steam to the desuperheater then to the condenser (see Fig. 4-1). Concurrently, the feedwater flow is ramped to the shutdown rate (typically 15%). Eventually, the steam generators are fully flooded and the feedwater flow, steam generator pressure, and helium flow are adjusted for long-term RHR.

Fig. 4-1. MLCS shutdown conditions

The transient response of the MLCS in the GCFR core following a reactor trip is shown in Fig. 4-2. The hot spot cladding temperature of the maximum-powered fuel rod and the core inlet and outlet temperatures decrease rapidly after reactor trip, indicating adequate core cooling by the MLCS. The steam generator floods fully at about 5 min. The MLCS uses some nonsafety-class equipment and therefore is not a safety system.

4.2. SHUTDOWN COOLING SYSTEM (SCS)

The SCS is a safety system consisting entirely of safety-class equipment (see Fig. 4-3). The SCS shares the main circulator, the circulator shaft, and the steam generator with the MLCS. The SCS uses a pony motor to drive the circulator with a safety-class (1E) power source.

The SCS will be used for RHR under a number of accident conditions, such as loss of offsite and onsite power, loss of feedwater, etc. The sequence of operation for transferring from the MLCS to the SCS is as follows:

1. When the main circulators coast down below the shutdown MLCS speed (15%), the pony motors are energized to maintain the circulator speed at 15% and the SCS heat rejection system is activated.
2. The feedwater flow from the boiler feed pump (BFP) is replaced by that from the SCS floodout pump.
3. Steam is vented until the steam generator is fully flooded.
4. Using the circulating water pump, the water is then circulated in a closed loop through the air-water cooler

Fig. 4-2. GCFR core and coolant thermal response to spurious reactor trip (3 MLCS, best estimate)

Fig. 4-3. GCFR SCS pressurized cooling operation

(AWC), where the core decay heat is removed by fans to the atmosphere.

Figure 4-4 shows the reactor core response to SCS cooling during a loss of offsite and onsite power for three assumed cases that use the following:

1. All three SCS loops with the best estimate model.
2. Two SCS loops allowing a single failure and with the conservative model.
3. Three SCS loops with main loop isolation valve failure allowing core bypass flow and with the conservative model.

The best estimate and the conservative models are defined by the use of uncertainty factors for the system parameters as given in Table 4-1. The conservative margins of major significance are a 20% increase in the core decay heat rate and a 20% increase in the estimated primary loop pressure drop. For both cases, adequate core cooling is indicated in Fig. 4-4.

4.3. CORE AUXILIARY COOLING SYSTEM (CACS)

As a backup, the safety RHR function can be performed by the CACS. Activation of the CACS, however, will follow only after the unavailability of two or more SCS loops. Each of three CACS loops is comprised of an electrically driven auxiliary circulator, a check valve, and a helium-to-water heat exchanger called the CAHE.

The CACS is designed to provide abundant core cooling following all design basis events, including the DBDA of the primary cooling

Fig. 4-4. Maximum powered fuel hot spot cladding thermal response during SCS cooling indicating influence of conservatism

TABLE 4-1
SYSTEM UNCERTAINTY FACTORS USED FOR BEST ESTIMATE AND CONSERVATIVE MODELS

Parameter	Best Estimate Model	Conservative Model
Decay heat	1.0	1.2
Local power uncertainty	1.0	1.05
Overall conductance of CACS heat exchangers (CAHE and auxiliary loop cooler)	1.0	0.85
Loop pressure drop	1.0	1.2
Coolant thermal conductivity	1.0	0.95
Coolant viscosity	1.0	1.03
Containment absolute backpressure (only for DBDA)	1.0	0.9
Core bypass flow fraction (nominally 1%)	1	2
Fuel cladding engineering factor		
Hot channel factor	1.111	1.131
Hot film factor	1.248	1.296
Hot cladding factor	1.153	1.195

system. Additionally, if power to all active components is unavailable, the heat transfer components within the CACS are located at sufficient elevation differences such that natural circulation will transfer heat from the core all the way to the ultimate heat sink, which is atmospheric air, providing the PCRV is pressurized.

4.4. ANALYSIS OF MAJOR FAULT TRANSIENTS

The GCFR is designed to respond to and accommodate a complete spectrum of accident initiators. Each of the events is selected for one or more of the following reasons:

1. It is a design basis event for a system or component, included for completeness to demonstrate that it is bounded by a design basis event.
2. It is required by the Nuclear Regulatory Commission (NRC) to be included in a Safety Analysis Report.
3. It is required by industry code or standard.

Accident initiators are grouped into the following categories:

1. Breaks in high-energy water system piping.
2. Increase/decrease in core heat removal.
3. Reactivity and power distribution anomalies.
4. Moisture ingress into primary coolant system.
5. Decrease in reactor coolant inventory.
6. Radioactivity release from a subsystem or component.
7. Failure of normally operating auxiliary systems.
8. Events external to the plant.

The complete list of accident sequences to be evaluated is extensive, since several combinations of accident initiators and event

sequences must be considered in each category. Previous risk assessments, such as the Reactor Safety Study (WASH-1400) (Ref. 4-1), have shown that the more frequent transients may represent the more dominant sequences that could ultimately lead to loss of core cooling. To combat this situation, reliability assessments of safety performance are included as an integral part of GCFR safety activities. At the conceptual design stage, this activity assists in the selection of basic system configurations. As the design advances to progressively more detail, control and instrumentation systems are included.

4.4.1. Methods

The transient response of the reactor system to accidents is analyzed with the FASTRN program. This program simulates dynamics of the GCFR primary and secondary coolants of the main cooling systems, as well as the auxiliary safety systems, under pressurized and depressurized conditions. The primary coolant system is modeled with gas volumes which respond according to the perfect gas law and the energy and mass conservation equations. These volumes are connected by paths which represent the components of the system (i.e., the reactor core, the steam generators, the helium circulator, and a leak path to the containment). Each of these components are, in turn, dynamically modeled by their governing equations.

The core is modeled with six rods representing three fuel and three blanket rods: an average powered rod, a maximum powered rod with nominal (or expected) heat transfer parameters, and a maximum powered rod with degraded heat transfer parameters to include the effects of statistically evaluated hot spot factors for both the fuel and blanket groups. Variations in the convective heat transfer coefficient and friction factor with local Reynolds number are included for both the smooth and the artificially roughened regions of the representative fuel rods. Reactivity feedback contributions associated with six distinct sources are evaluated:

1. Doppler.
2. Core axial expansion.
3. Helium density changes.
4. Core element thermal bowing.
5. Core splaying associated with grid plate pressure drop loads.
6. Grid plate thermal expansion.

The resulting reactor power is determined by a six-group point kinetics solution. Provision for automatic control rod and/or backup shutdown rod insertion upon generation of either the first or a subsequent reactor trip signal is included.

The steam generator is modeled from the outlet of the feedwater pump to the inlet of the main turbine. Provisions for modelling the feedwater and steam valves with their associated controllers are also included. The circulator dynamic response is evaluated using circulator performance functions. Incompressible flow is assumed for all flow paths, with the exception of the leak path, which is treated in a compressible manner for both choked or unchoked flow.

Simulation of the CACS components is included. The coolant dynamics are modeled for forced or natural circulation through the primary helium, secondary water, and tertiary air loops of the CACS.

4.4.2. Loss of Coolant

Although PCRV seal failures are extremely improbable, the GCFR design is based on providing adequate RHR for a postulated depressurization accident (Ref. 4-2). A depressurization accident, based on a non-mechanistic, assumed 4840-mm^2 (75-in.^2) free flow leak area open to the reactor containment building, has been analyzed. Under such conditions, the reactor coolant pressure would decrease exponentially with time and would become equal to the containment pressure of ~ 200 kPa (29,010 psi) in ~ 1.5 min.

Since such a depressurization is an extremely low probability event, one RHR system is considered to be adequate. The CACS is chosen specifically to perform the RHR function under DBDA conditions. Significant margin to a DBDA is displayed for the 4840-mm² (75-in.²) leak given here as an example.

A typical sequence of events during a DBDA is as follows:

1. Soon after initiation of a DBDA, a reactor trip is actuated.
2. The plant protection system identifies this condition, and the CACS is phased in to take over the RHR function.
3. Following the reactor trip, the power from emergency diesel generators is used to bring the CACS equipment into a standby condition.
4. A depressurization accident is detected by a signal resulting from the primary coolant pressure decrease, and auxiliary circulators are accelerated to their full design speed.
5. As the auxiliary circulators accelerate, the auxiliary circulator pressure rise forces the auxiliary loop isolation valves open and the main loop isolation valves closed, completing the loop transfer.

At the end of depressurization blowdown, air from the containment atmosphere is dispersed into the primary coolant loop through the leak passage by means of thermally induced inhalation, natural circulation, and eventually, molecular diffusion. The effect of this

air ingress is included in the analysis. Air ingress makes the coolant a better heat transfer and heat transport medium for a given volumetric flow. However, with the CACS motor having both maximum torque and maximum speed limitations, air ingress has a small effect on the overall core cooling capability with the CACS operating.

The flow is expected to be laminar through the core when the core becomes depressurized. The heat transfer and friction characteristics of laminar flow are accounted for in the transient analysis presented herein. The hot spot cladding temperature response in Fig. 4-5 was obtained assuming transfer to two CACS loops at 1 min after reactor trip and incorporating the conservative model for the system parameters (see Table 4-1). The maximum hot spot cladding temperature of a maximum-powered typical fuel rod is $\sim 110^{\circ}\text{C}$ (230°F) lower than the cladding faulted limit of 1260°C (2300°F) and 210°C (410°F) lower than the cladding melting point of 1370°C (2498°F). This allows an adequate margin to accommodate local anomalies, such as the fuel assembly edge channel effect, and a margin for potential fuel rod damage below the cladding melting temperature. Depending on the fuel assembly design, the edge fuel rod adjacent to the fuel assembly duct wall can be 50° to 100°C (122° to 212°F) higher than the typical interior rod of the bundle under accident conditions. Separate studies are under way to define the edge channel effect accurately and to mitigate the effect by improving the fuel assembly design, if necessary.

4.4.3. Loss of Power

The SCS will be used for RHR under a number of accident conditions, such as loss of offsite and onsite power, loss of feed-water, etc. The sequence of operation for transferring from the MLCS to the SCS is as previously described (see Fig. 4-4 and Table 4-1).

Fig. 4-5. Core inlet and maximum-powered fuel hot spot cladding temperature following a rapid depressurization through a 4840-mm² (75-in.²) leak

4.4.4. Loss of Flow

An inherent safety characteristic which significantly improves accident prevention capability of the GCFR was recently incorporated into the design. This design feature is an upflow core which provides natural circulation RHR capability. This system provides a diverse backup to three redundant forced circulation RHR systems. To improve system reliability and diversity and to eliminate potential failure mechanisms associated with mechanical systems and operator actions, the natural circulation system is designed to provide passive short- and long-term RHR to the ultimate atmospheric heat sink in a pressurized GCFR.

During normal operation, the pressure head of the main helium circulator keeps the main loop isolation valves open and the auxiliary loop isolation valves closed. Upon loss of forced circulation in the main loops, the main circulators coast down. When the circulator pressure head reaches a small value (typically at 9% circulator speed when all circulators are in operation), the main loop isolation valves are closed and the auxiliary loop isolation valves are opened by gravity. If the main loop valves fail to close during auxiliary loop natural circulation, flow bypassing the core is limited to a small amount by buoyancy forces in the main loop. This self-isolation feature of main loops is a function of relative CAHE, steam generator, and upper main loop cross-duct elevations. Criteria for these elevations have been established to assure that the self-isolation feature is maintained in the design.

The GCFR performance following a complete loss of forced circulation was analyzed first in a base case, and sensitivities to the assumed conditions were then determined by varying individual assumptions.

The initiating event of the base case is a total loss of drive power to the main circulators. The main circulators immediately begin coasting down followed by a reactor scram. Under such a scenario, the SCS pony motors would be expected to maintain forced circulation in the main loops. However, it is assumed that the pony motors are unavailable. In addition, an extremely conservative assumption is made that none of the three CACS circulators start. Ninety seconds after the loss of circulator drive power, core flow decays to less than 10%. The low flow then allows the main loop isolation valves to shut by gravity, while the auxiliary loop isolation valves fall open. The temperature profile existing in the auxiliary loop rapidly induces a natural circulation flow between the core and the CAHEs. This helium coolant flow is seen to increase from near 0% to 3.5% of its normal design flow in ~ 3 s. Simultaneously, the increased heat load to the CAHE increases the natural circulation in the CACS water loop and in the natural draft chimney. These natural circulation flows are found to maintain hot spot fuel cladding temperatures below 602°C (1148°F). [The faulted temperature limit for the cladding is 1300°C (2372°F)]. The maximum temperature reached in the pressurized water system is 185°C (365°F). [The saturation temperature of the water loop is 302°C (576°F)]. Clad and water temperature histories are illustrated in Figs. 4-6 and 4-7. Also displayed are sensitivities to transfer time and number of CACS loops available.

4.5. HYPOTHETICAL ACCIDENT BEYOND THE DESIGN BASIS

An accident sequence considered in the GCFR is initiated by a loss of all forced circulation and the postulated failure to establish natural convection in a shutdown reactor (Ref. 4-3). This hypothetical accident is analyzed to investigate the potential for consequence mitigation and containment margin, because its potential for core disruption may be greater than for unprotected accidents.

Fig. 4-6. Sensitivity of CAHE water outlet temperature to number of auxiliary loops in operation

Fig. 4-7. Sensitivity of hot spot fuel clad temperature to loop transfer time in primary coolant natural circulation

This type of postulated accident has several characteristics which are distinctly different from loss of flow accidents initiated at full power and without scram. The initial fuel heating rates are dominated by decay heat. Therefore, the time scale of melting is two to three orders of magnitude longer. At the time of initial fuel melting, the reactor is shut down, and delayed neutron precursor concentrations are orders of magnitude smaller. Therefore, larger amounts of reactivity may be inserted before fission power can become a significantly greater heat producer than decay heat and before Doppler feedback is important. Furthermore, as fission power increases, the neutron spectrum is harder because of the control rod insertion and nearly complete decladding of fuel rods. Hence, the Doppler coefficient is smaller than in the unprotected case. Work is in progress to determine realistic magnitudes of reactivity feedback.

The initiating event phase of the hypothetical loss of secondary cooling accident sequence requires a series of common mode failures to be postulated that leads to a total loss of decay heat removal. The reference scenario considered for analysis of the core melting progression postulates the following occurrences:

1. A simultaneous loss of all drive power to the electrically driven main circulators is assumed to occur as the initiating event.
2. A common mode failure of the main circulator pony motors is postulated to fail the SCS when the main circulators have coasted down to the pony motor design speed.
3. Failure of the CACS to energize by common mode is postulated, which disables the CACS forced convection cooling mode.

4. A common mode failure of the CACS isolation valves to open by gravity is postulated to prevent the removal of decay heat by natural circulation.

Cladding and assembly wall melting is expected within several minutes in the absence of loop coolant circulation under decay heat generation because of the limited core heat capacity. The accident sequence is summarized in Fig. 4-8, and the phenomenological event timing below is based on neglecting interassembly and intra-assembly natural convection in an upflow GCFR core. This postulated accident sequence is substantially the same whether the reactor is tripped by the engineered plant protection system signals or is already shut down when circulator coastdown begins. However, the length of time between each significant event is extended as the loss of flow is delayed after shutdown.

A new computer program called SCORIA (Slumped CORE Integrated Analysis) has aided in this event sequence quantification. SCORIA is essentially a lumped heat capacity, thermal network analysis tool, which includes conduction, forced convection, and radiation heat transfer from one node to another and accounts for the change of phase of steel and fuel.

A timewise series of core material distributions deduced from SCORIA was analyzed with an R-Z diffusion theory model using the 2DB computer code (Ref. 4-4). The purpose was to determine the approximate time and ramp rate of a postulated recriticality. Results are presented in Table 4-2.

The major phenomena which lead to concern over the potential for recriticality should this accident occur are the following:

1. Steel blockage formation in the lower axial blanket.

Fig. 4-8. LOSC event sequence diagram, upflow core pressurized case

TABLE 4-2
REACTIVITY VERSUS TIME FOR POSTULATED EVENT

Transient Time (s)	Configuration Detail	k_{eff}
0	Hot shutdown	0.89
300	44% of clad slumped Top 77 mm of lower axial blanket plugged	0.91
770	6.6% of fuel molten Nearly all cladding slumped Some duct wall melting	0.92
870	21% of fuel molten Most of assembly walls melted Molten fuel in every assembly	0.93
970	31% of fuel molten All but outermost assembly wall slumped	0.99
1110	45% of the fuel molten	1.09

2. Subsequent axial assembly wall melting and molten steel pool buildup leading to molten steel spillover into the interassembly spacing and solidification there.
3. Declad fuel column crumbling prior to melting.
4. Molten fuel slumping.

The accident sequence also suggests mechanisms for preventing recriticality. The most important are the following:

1. In-situ poisoning of a molten core with a dispersed and mixed poison material.
2. Molten fuel drainage.
3. A combination of poisoning and drainage.

The potential for fuel drainage may prove to be an attractive feature of gas-cooled systems. The feasibility of these mechanisms is currently under investigation at General Atomic.

4.6. STATISTICAL ANALYSIS OF SAFETY

Previous reliability analyses (Ref. 4-5) have identified that, with respect to loss of coolable core geometry, the RHR system is more critical than the reactor shutdown systems or structures. A detailed reliability assessment of the RHR systems was performed (Ref. 4-6) and was one of the considerations that led to a major design option assessment for the GCFR. The design option assessment was recently completed and led to the selection of a new reference configuration. The principal focus of that assessment was on the core flow direction, because an upflow core design offers the capability for pressurized decay heat removal by natural convection. The

reliability portion of the design option assessment concentrated on the quantitative evaluation of the RHR trains and electric power supplies. Quantitative reliability evaluations of other RHR subsystems, such as component cooling systems and the controls and instrumentation, have yet to be performed. The RHR system consists of three cooling systems, each having multiple redundant loops:

1. The MLCS utilizes the normal power conversion equipment with a main turbine bypass for RHR.
2. The SCS shares the steam generators and the main circulators with the MLCS, except that the circulator can be driven either by the main motor or by a pony motor with a safety-grade power supply. Heat rejection in the SCS is accomplished through three AWCs that reject heat to the atmosphere. Initiation of cooling by the SCS requires floodout of the steam generators, which is accomplished by the feedwater pumps in the MLCS. After floodout has been completed, the water is circulated through the steam generators, AWCs, and connecting piping by three separate SCS feedwater pumps. The entire SCS is safety class.
3. The CACS is a totally independent safety-class system that consists of three redundant helium-water-air loops. Auxiliary circulators provide heat transport from the core to the CAHEs. A water loop transports the heat from the CAHEs to the auxiliary loop coolers, where heat is rejected to the atmosphere. All CACS loops can operate on either forced or natural circulation in the helium loops and in the water and air loops. There are auxiliary loop isolation valves in the CACS that are closed during normal operation to essentially prevent bypass flow around the core. These valves open when core cooling is performed

by the CACS. The main loop isolation valves are open during CACS operation to prevent bypass flow around the core.

There are two basic sources of electric power: (1) the preferred power supplied by either the onsite main turbine generator or the offsite grid and (2) the emergency power provided by batteries (primarily for controls and instrumentation) and by auxiliary generator systems. Two forms of auxiliary generator systems were analyzed. One uses three separate standby electric power supplies (such as diesel generators), each of which supplies one SCS loop and one CACS loop. The second has six such supplies, three individually dedicated to the three SCS loops and three individually dedicated to the three CACS loops.

A large number of events can initiate a need for the RHR function. A list of more than 40 such events was used in this analysis, covering the full spectrum of conditions that have traditionally been classified in the licensing process as normal, upset, emergency, and faulted. For each event in the list, estimates were developed for its occurrence frequency and downtime. A sample of individual events and of collections of events from the complete list of initiating events is given in Table 4-3.

The detailed numerical evaluation yielded three different types of results:

1. The estimated annual occurrence frequency for each individual event.
2. The restoration time associated with each individual event.

TABLE 4-3
INITIATING EVENTS FOR RHR

Typical Event ^(a)	Estimated Frequency ^(b) occurrence/yr	Downtime (h)
Shutdown for refueling	1.0	700
Control rod malfunction (total)	0.05	6-46
Inadvertent valve operation (water/steam)	0.25	10-12
Inadvertent trip (reactor or MLCS)	2.0	6-10
Turbine trip	2.7	6
Heat exchanger leak (total)	0.56	396
Total loss of feedwater	0.06	40
Loss of offsite power with turbine trip	0.01	0.25
Accidental depressurization (total)	3×10^{-5}	312-1440
Feed/steam line rupture (total)	2.2×10^{-4}	75
Earthquake	2×10^{-6}	720

(a) Individual initiators and collections of initiators from list of 42 initiating events.

(b) Initiator frequency for mature plant.

3. The probability of failure (per demand) to provide RHR for each event (including the effects of unavailability of offsite electrical power).

The product of the occurrence frequency for an event and the corresponding RHR failure probability yields the estimated frequency of loss of RHR associated with that event. Summing all such products over all events yields the estimated total frequency of loss of RHR. Three such results were determined: one for forced convection RHR systems only and two for the combined forced and natural convection systems, one with and one without repressurization. Comparing these results provides a basis for quantifying the benefits to be gained from a pressurized natural convection cooling capability. These results also provide an indication of the absolute RHR reliability achievable with the reference concept and with revisions thereto. They also serve as a basis for deciding whether three standby electric power supplies (serving SCS and CACS in common) are adequate, what improvements are achievable if one set of three such supplies is used for the SCS and a separate set of three supplies for the CACS, whether the addition of three floodout pumps (and associated equipment) in the SCS significantly enhances reliability, and what maximum reliability improvement might be attainable by incorporating a repressurization capability in the design.

A summary of the numerical results obtained thus far is presented in Table 4-4, where a measure of analysis uncertainty was provided by basing the calculations on two different statistical assumptions: independence and dependence (common cause failures). The top portion of the table addresses the reference design concept and provides a breakout of pressurized and depressurized events. It can be seen that the addition of a natural convection capability (without repressurization) yields a substantial improvement for pressurized events but only a small improvement for depressurized events. It is clear that, because of this latter consideration, the overall gain is

TABLE 4-4
RHR FAILURE PROBABILITY SUMMARY
FOR HEAT REMOVAL TRAIN AND POWER SUPPLY SYSTEMS
COMPARISON OF FORCED CIRCULATION AND FORCED/NATURAL CIRCULATION

Configurations	RHR Failure Probability Per Year		
	CACS Forced Circulation Only	CACS Forced/Natural Circulation	
		Without Repressurization	With Repressurization
Reference design ^(a)			
Statistical independence estimate			
Pressurized events, only	1.0E-6	3.3E-8	3.3E-8
Depressurized events, only	<u>1.8E-6</u>	<u>1.7E-6</u>	<u>3.3E-9</u>
Total	2.8E-6	1.7E-6	3.6E-8
Common cause estimate			
Pressurized events, only	9.8E-5	2.2E-5	2.2E-5
Depressurized events, only	<u>8.5E-5</u>	<u>8.4E-5</u>	<u>1.0E-6</u>
Total	1.8E-4	1.1E-4	2.3E-5
Revised design ^(b)			
Statistical independence estimate			
Pressurized events, only	2.3E-10	4.9E-11	4.9E-11
Depressurized events, only	<u>5.2E-8</u>	<u>2.3E-8</u>	<u>3.4E-9</u>
Total	5.2E-8	2.3E-8	3.4E-9
Common cause estimate			
Pressurized events, only	9.0E-6	1.3E-6	1.3E-6
Depressurized events, only	<u>3.7E-6</u>	<u>1.6E-6</u>	<u>5.2E-7</u>
Total	1.2E-5	2.9E-6	1.8E-6

(a) No floodout pump in SCS; three electric power supplies, serving SCS and CACS.

(b) Three floodout pumps added in SCS; three power supplies for CACS and an additional three for SCS.

only moderate. However, with the addition of a repressurization capability, the depressurized events become less significant, resulting in a much better overall improvement from the addition of natural convection.

Displaying the results for the reference design in terms of the major unreliability contributors (as in Table 4-5) is helpful in identifying the most effective ways for improving reliability. Two areas of major unreliability contribution were found to be (1) flood-out of the steam generators (required for SCS startup) by the BFPs and (2) the use of one emergency electric generating system serving both the SCS and the CACS. As a result, a revised design was proposed and analyzed in which redundant floodout pumps were incorporated in the SCS and two emergency electric generating systems were provided (one for the CACS and a separate one for the SCS). The results for this revised design are displayed in the lower portion of Table 4-4. These two revisions yielded about an order of magnitude reduction in the frequency of RHR failure.

From the standpoint of absolute reliability, it is believed that the frequency values presented in Table 4-4 (for the heat removal trains and electric power supplies only) for the reference design are too high except for the concept with a repressurization capability. However, the frequencies for the revised design appear low enough to be acceptable, even without reliance on natural convection.

The study results indicate that the quantitative gains in RHR reliability achievable by natural convection are limited by (1) the demand and (2) the reliability of equipment required to change state to establish natural convection (i.e., valves, dampers, etc.). The major improvements in RHR reliability due to natural convection are from the following:

1. Increased reliability of pressurized RHR.

TABLE 4-5
MAJOR UNRELIABILITY CONTRIBUTORS IN REFERENCE DESIGN

	Statistical Independence Estimate			Common Cause Estimate		
	Occurrence/yr	(%)	Event	Occurrence/yr	(%)	Event
FC ^(a) (Only)	1.2E-6	43	Refueling, electrical	6.1E-5	34	Refueling, electrical
				5.7E-5	32	Feedwater loss, MCS and SCS
FC ^(a) + NC ^(b)	1.2E-6	71	Refueling, electrical	6.0E-5	55	Refueling, electrical
FC ^(a) + NC/R ^(c)	3.2E-8	88	Feedwater loss, MCS and SCS	1.8E-5	78	Feedwater loss, MCS and SCS

(a) FC: Forced convection.

(b) NC: Natural convection without repressurization.

(c) NC/R: Natural convection with repressurization.

2. Decreased dependence on offsite and onsite power supplies.
3. Decreased dependence on auxiliary and support systems, such as the plant cooling water system, circulator support systems, and the control and instrumentation systems.
4. Substantially increased resistance to common mode failures in the helium, water, and air circulation due to the inherent diversity between forced and natural circulation.

To optimally exploit natural circulation decay heat removal, a repressurization capability for normally depressurized conditions would be most beneficial. Additional substantial improvements can be made by minimizing the dependence on equipment for the natural circulation system. Studies to date have only included a forced convection CACS design which could operate in the natural circulation mode. A study of optimized natural circulation systems is currently in progress.

REFERENCES

- 4-1. "Accident Definition and Use of Event Trees," Appendix I to "Reactor Safety Study," U.S. Nuclear Regulatory Commission Report WASH-1400, Appendix I, October 1975 (NUREG-75/014).
- 4-2. Chung, H. S., et. al., "GCFR Residual Heat Removal Capability," General Atomic Report GA-A15545, August 1979.
- 4-3. Frank, M., and J. Reilly, "Event Sequence Quantification for a Loss of Shutdown Cooling Accident in the GCFR," General Atomic Report GA-A15556, October 1979.
- 4-4. Little, W. W., Jr., and R. W. Hardie, "2DB User's Manual - Revision 1," Battelle Pacific Northwest Laboratories Report BNWL-831, Rev. 1, 1969.

- 4-5. Kelley, A. P., Jr., "Gas Cooled Fast Breeder Reactor Accident Initiation and Progression Analysis Progress Report for the Period July 1, 1975 through June 30, 1976," General Atomic Report GA-A14079, March 1977.
- 4-6. Kelley, A. P., Jr., and T. Taniguchi, "A Reliability Analysis of the Residual Heat Removal Systems for a 300 MWe GCFR," General Atomic Report GA-A14653, January 1978.

5. LICENSING

In March 1971, an application was submitted to the Directorate of Licensing of the Atomic Energy Commission (DOL) requesting preapplication review of the GCFR as described in the Preliminary Safety Information Document (PSID) (Ref. 5-1). This document is a smaller version of a Preliminary Safety Analysis Report (PSAR). Both the DOL and the Advisory Committee on Reactor Safeguards (ACRS) were requested to judge whether the conceptual design could be developed into a plant without undue hazard to public health and safety. The application was accepted and docketed, and an active review of the GCFR was conducted by the DOL and ACRS during the next three years.

Formal review of the PSID was concluded in 1974 when the DOL published a Preapplication Safety Evaluation Report (SER) (Ref. 5-2), and the ACRS issued an interim letter (Ref. 5-3) to identify safety areas needing research and development prior to issuing a construction permit. The DOL assessment of the GCFR design was summarized as follows:

"We conclude that the GCFR Demonstration Plant proposed in concept by GA, and as conditioned by this safety evaluation, is potentially capable of being designed and constructed so that it can be operated without undue risk to the health and safety of the public."

REFERENCES

- 5-1. "Gas-Cooled Fast Breeder Reactor, Preliminary Safety Information Document," Gulf General Atomic Report GA-10298, February 15, 1971.

- 5-2. "Preapplication Safety Evaluation of the Gas-Cooled Fast Breeder Reactor, Project No. 456," USAEC Report, August 1, 1974.
- 5-3. Stratton, W. R., "Conceptual Design for Prototype Gas-Cooled Fast Breeder Reactor (GCFBR)," Advisory Committee on Reactor Safeguards letter to Dixie Lee Ray, November 8, 1974.

6. CONCLUSION

The long-term GCFR project goal is to ensure that GCFR power plants will be commercially available when they are needed, since General Atomic believes that breeder reactors are essential to nuclear production of long-term, reasonable cost, electrical energy. Furthermore, General Atomic believes that the GCFR promises to be an attractive breeder, and that two breeder concepts should be concurrently developed to assure commercial success and to provide the benefits of competition.

This report describes the design features of the first GCFR, a 350-MW(e) demonstration plant. The GCFR is an attractive breeder concept technically, because helium is used as the reactor coolant, rather than sodium, as in the LMFBR designs. Helium should improve breeding performance, reduce capital costs, and improve operability, maintenance, and safety.

Since helium is molecularly light and has an extremely small neutron absorption coefficient, neutron energies are higher in the GCFR than in other breeder reactor types. Thus, the GCFR has a fundamentally better breeding ratio and a decreased compound doubling time, making it a better fuel producer.

Since helium is a chemically and neutronically inert single-phase gas which does not become radioactive, the GCFR does not require an intermediate heat transport loop nor special cell liners, fire protection, and decontamination and cleanup systems. Simple-loop design arrangements can be obtained, because helium coolant allows the primary loops to be integrated within a PCR. This system has been

tried and proven in many years of operation in 23 European and one U.S. power plant. Since the system makes maximum use of carbon steels rather than the more expensive stainless steels, the GCFR should be a less expensive fuel and energy producer.

The single-phase helium coolant also eliminates the problem of coolant void coefficients and the attendant consequences, simplifying safety research and development. Potential chemical reactions between the primary coolant and other plant fluids are eliminated, because helium is chemically inert. Maintenance access problems tend to be minimized, because the helium is not radioactive. Additionally, helium is optically transparent, permitting remote visual inspection of primary coolant system components and facilitating fuel handling and other maintenance and surveillance operations.

In addition to the above advantages, the development costs required for the GCFR promise to be relatively small, because they are incremental in nature, relying on existing core technology developed in the LMFBR program and on existing NSSS component technology from the high-temperature gas-cooled reactor (HTGR) programs.

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