DOE/NE-0003/5
UC-2

# PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENT

## VOLUME V

# GAS-COOLED FAST REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY
WASHINGTON, D.C. 20545

8006160226

## NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, expressed or implied, nor assumes any legal liability nor responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, nor represents that its use would not infringe privately owned rights.

**Available from:**

National Technical Information Service (NTIS)
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

**Price:** Printed Copy: $ 6.50
Microfiche: $ 3.00

This report has been prepared jointly by Argonne National Laboratory, Oak Ridge National Laboratory, and NUS Corporation for the Office of Fuel Cycle Evaluation, U.S. Department of Energy.

## FOREWORD

The Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. The NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The NASAP objectives are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

The NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The DOE is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to ensure that their views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which the NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in U-235 to 20% or more, uranium containing U-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e., spiking), the use of radioactive sleeves in the fresh fuel shipping casks, and other measures. The basis for the safeguards review by the NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference LWR once-through case and may affect their licensing. When issues exist, this document briefly describes the research, development, and demonstration requirements that would help resolve them with the normal engineering development of a reactor/fuel cycle system.

The preparation of this document takes into consideration NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made in this document. Additional comments were incorporated as Appendix B. Comments that are beyond the scope and resources of the NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (continued)

## LIST OF FIGURES

# Chapter 1

# GENERAL DESCRIPTION

The gas-cooled fast-breeder reactor (GCFR) is a nuclear steam supply system (NSSS) in which fission heat generated by a fast-spectrum reactor is transported by pressurized helium coolant to a number of parallel heat exchangers to generate steam. The reactor core and the steam generators are contained in a prestressed-concrete reactor vessel (PCRV) for pressure containment. Figures 1-1 and 1-2 show the general layout of the PCRV for the 1,200-MWe NSSS, which have been derived based on the study reported in Reference 1.

Figure 1-3 illustrates normal, full-power plant-operating conditions. Helium from the electric motor-driven circulator flows upward through the reactor core, where it is heated to 1,030°F. It then flows downward over the tubes of the once-through steam generators, where it is cooled to 575°F, recompressed and recirculated to the core. Feedwater enters the steam generator at 351°F and is heated to produce superheated steam at 950°F and 1,800 psia. The steam is expanded through a conventional turbine-generator to produce power. Exhaust steam is condensed and pumped back to the steam generators via conventional feedwater heaters.

The reactor core is made up of hexagonal assemblies of three types. These assemblies contain either fuel rods (in a radially zoned enrichment pattern), control and shutdown rods for reactivity control, or blanket rods. The general fuel design uses the background and technology developed by the national liquid-metal fast-breeder reactor (LMFBR) fuels program.

Heat transfer in the GCFR is increased by surface roughening over much of the length of the fuel rods. Replaceable, fixed-diameter orifices are installed in the outlet ends of the assemblies. The fuel is continuously vented to an out-of-core fission-gas collection system.

Post-shutdown heat removal from the core is normally provided by the main cooling loops. In addition, two safety class systems are provided for long-term residual heat removal. These systems are the shutdown cooling system (SCS) and the core auxiliary cooling system (CACS). Each is a Seismic Category I system and is independent of the other.

Each loop of the SCS consists of a main-cooling-loop steam generator, a pony motor on the main helium circulator, water-air cooling system rejecting heat to the atmosphere, and a feedwater pump. The system operates as a closed loop. Figure 1-4 illustrates the SCS. Helium is circulated through the reactor core by the main helium circulators, which are driven by pony motors. The steam generators are used to cool the helium. Initially, water stored in an external tank is pumped into the steam generator (via the floodout pump) to ensure that the system is liquid-full. Subsequently, water is recirculated through the system via the circulating water pumps. Water leaving the steam generators at 510°F is cooled in a forced-draft cooling tower where atmospheric air is circulated across the heat-transfer surface. An accumulator, pressurized with $N_2$, is provided to maintain pressure on the circulating water at a level high enough to prevent boiling.

When the SCS is initially placed in service, steam from the steam generator in excess of the condensing capability of the cooler is vented to atmosphere via relief valves.

The standby pump circulates water during normal plant operation to maintain water purity in the external equipment.

Each loop of the CACS consists of a circulator driven by an electric motor, a shutoff valve, and a helium-to-water heat exchanger, all contained in a cavity inside the PCRV. A cooling-water supply system external to the reactor containment provides pressurized cooling water to the core auxiliary heat exchanger and rejects the heat to an external heat sink. Diesel generators ensure an electric supply to the auxiliary circulator motor. In addition, the CACS is so designed that it will cool the reactor by operation in a natural circulation mode independent of electrical supply.

Figure 1-5 illustrates the CACS in its forced circulation mode.

Operation is identical to the SCS, except for the following:

1. Helium is circulated through the core by an auxiliary circulator (motor-driven) which is completely independent of the main helium circulator.
2. Circulating helium is cooled in a core auxiliary heat exchanger (CAHE) which is completely independent of the steam generator.
3. The circulating water system is a true closed loop and does not require any venting of steam when the system is placed in operation.
4. The system is always water-filled and does not require additional water to fill it when placed in operation.
5. Heat is rejected to the atmosphere by means of air/water coolers which are not part of the shutdown-cooling system.

Figure 1-6 illustrates operation of the CACS under natural convection.

The CACS can provide adequate core cooling, assuming complete loss of all forced circulation capability, as long as the reactor is pressurized ($> \sim 150$ psia). Helium circulation through the core and the CAHE is maintained by the 31-foot elevation difference between the thermal centerlines of the core and the CAHE. Similarly, water circulation through the CAHE and the air/water cooler is maintained by the 86-foot elevation difference between these components. A bypass around the circulating pump (not shown) is provided, even though the shutdown pump would not introduce significant resistance to the flow of water under natural circulation conditions. During natural circulation cooling, the following components are <u>not</u> in operation:

| | | |
|---|---|---|
| 1. | Primary coolant loop | - auxiliary circulator |
| 2. | Secondary coolant loop | - circulating pump |
| | | - standby pump |
| 3. | Tertiary loop | - air fan |

The general arrangement of the air/water cooler relative to the CAHE and the reactor core is shown in Figure 1-7. The cooler is at the top of the confinement in order to assure natural convection heat removal from the CAHE located in the PCRV.

On the steam side external to the PCRV, condensate is fed to the once-through steam generators through feedwater heaters. Feedwater and superheated-steam connections are made through tubesheets at the bottom and top of the PCRV, respectively.

Flow of feedwater and steam is upward through the single-coil tube bundle, with the superheated steam being brought up to the top of the PCRV.

The PCRV is a thick-walled multicavity cylindrical pressure vessel constructed from high-strength reinforced concrete. The concrete is prestressed in the vertical direction by unbonded longitudinal tendons and in the circumferential direction by wire winding. The core cavity, steam-generator cavities, and CACS cavities are closed by concrete plugs secured by the tendons. A continuous steel liner attached to the PCRV inner surface provides containment for the helium coolant. This steel liner is insulated from the hot helium by thermal barriers, and heat passing through the barriers is removed by cooling-water tubes attached to the outer surface of the liner and embedded in the concrete. Thermal and radiation shields surround the core to protect the PCRV concrete.

The fuel cycle that has been the reference for most design efforts is based on the plutonium-uranium mixed-oxide fuel technology developed by the LMFBR program. The GCFR can use either uranium oxide or thorium oxide in the blankets with little change in thermal performance parameters.

The system operates on an approximately annual refueling cycle in which one-third of the fuel and control assemblies are replaced at each refueling. In the equilibrium cycle, effectively one-quarter of the radial blanket assemblies are replaced annually.

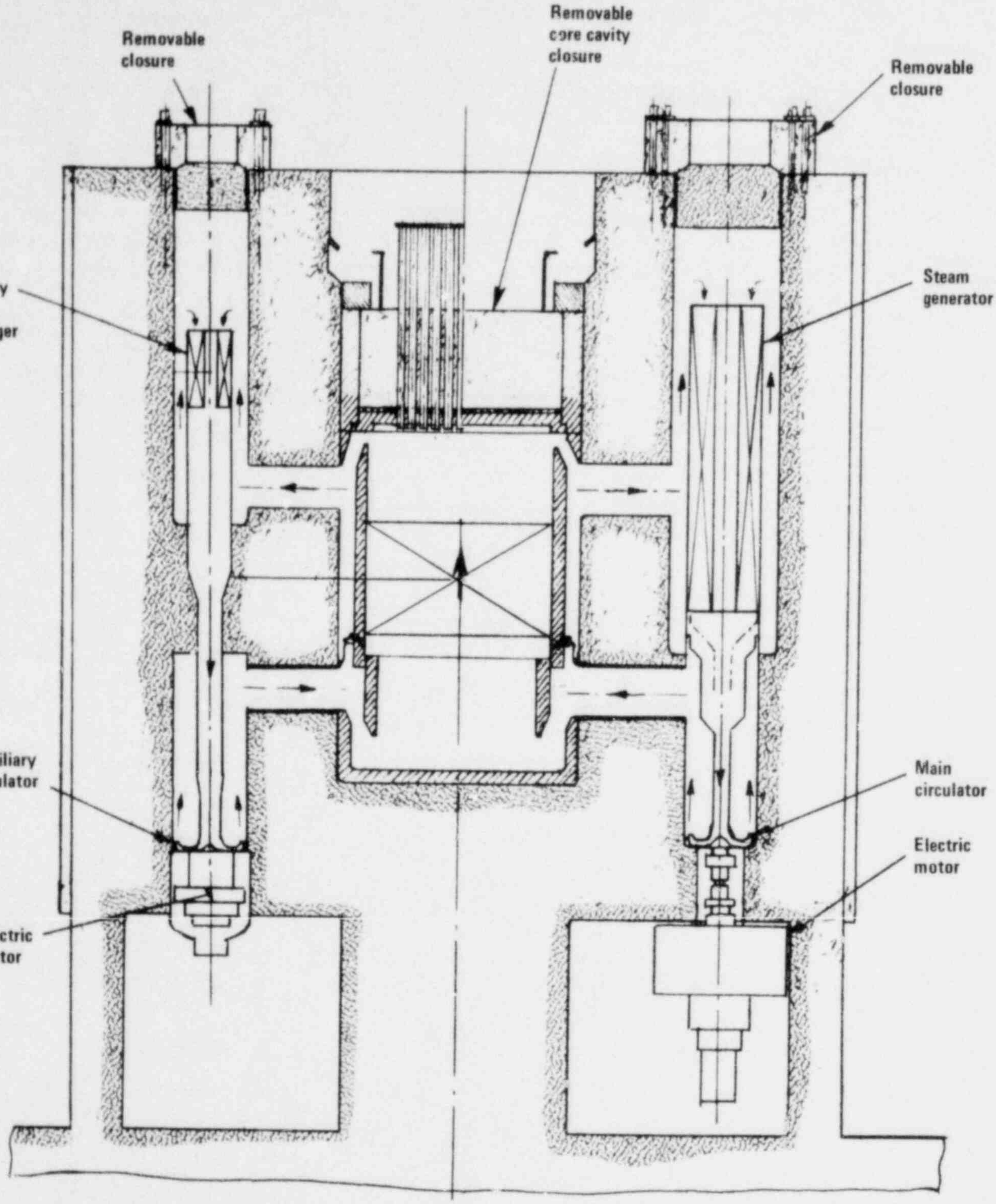



Figure 1-1. Vertical cross section of a 1,200-MWe GCFR.

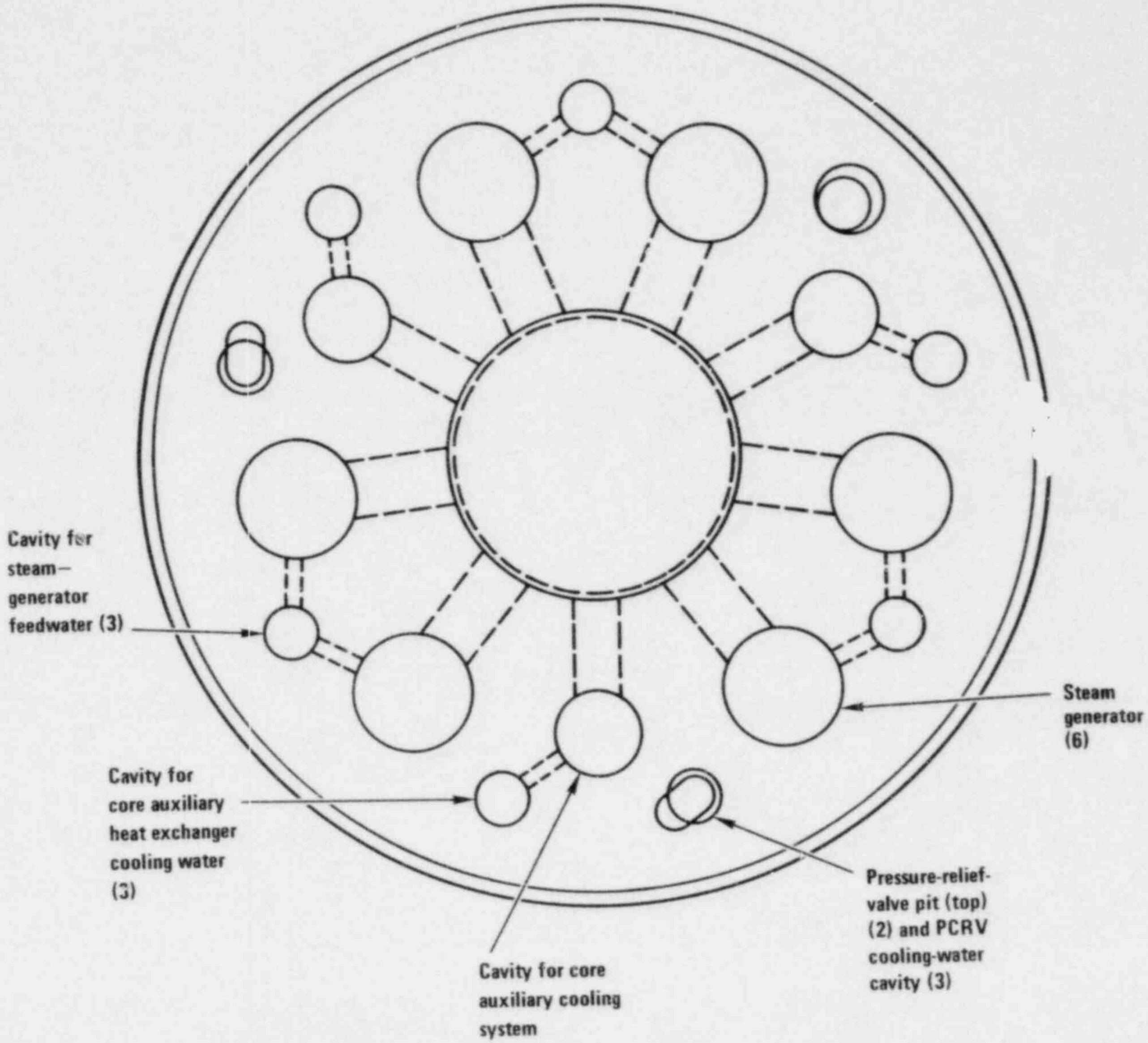



Figure 1-2. Horizontal cross section of a 1,200-MWe GCFR.

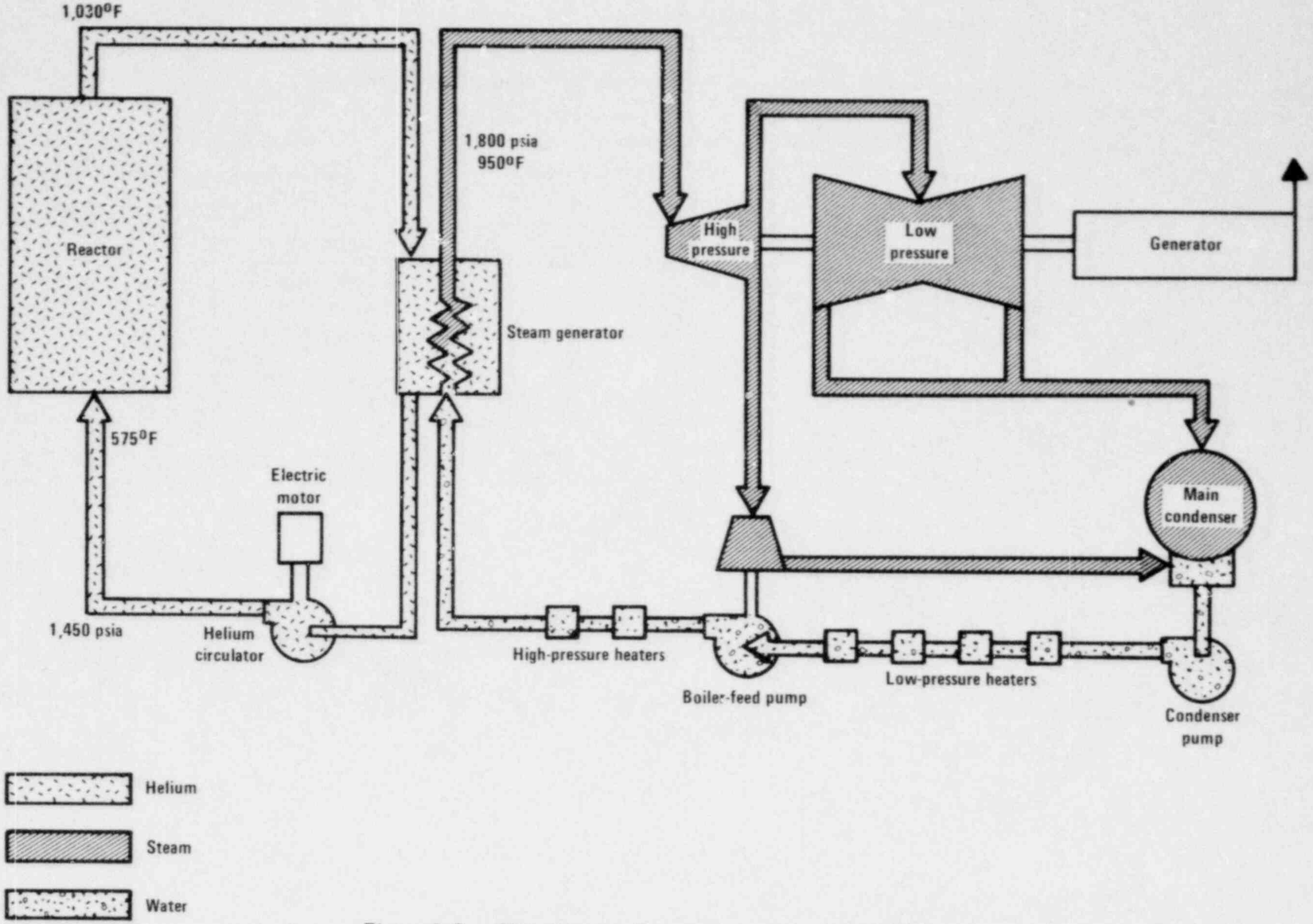



Figure 1-3. Main loop cooling system—operating conditions.

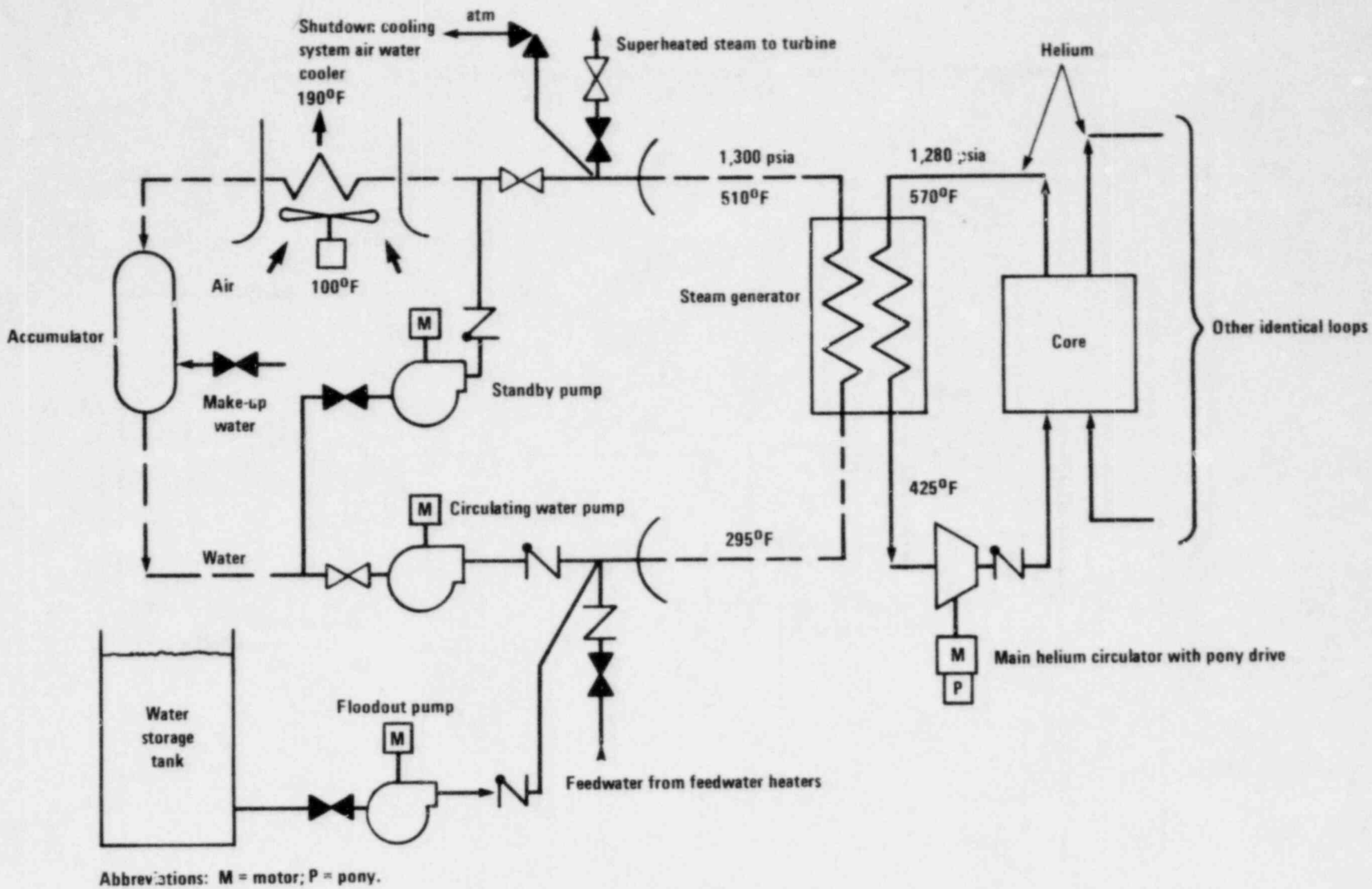
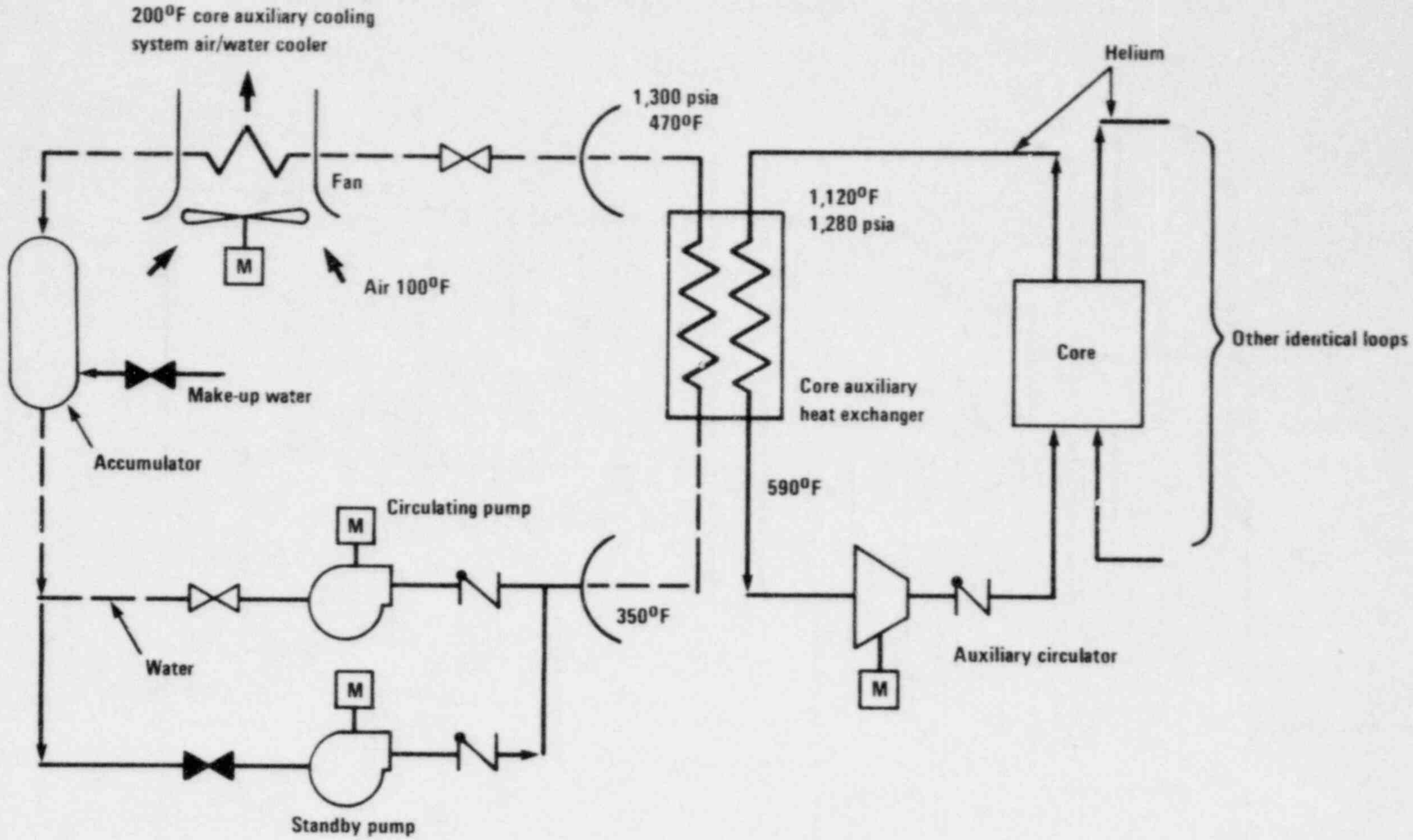



Figure 1-4. GCFR shutdown cooling system—pressurized cooling operation.




Abbreviation: M = motor.

Figure 1-5. GCFR core auxiliary cooling system—pressurized cooling operation.

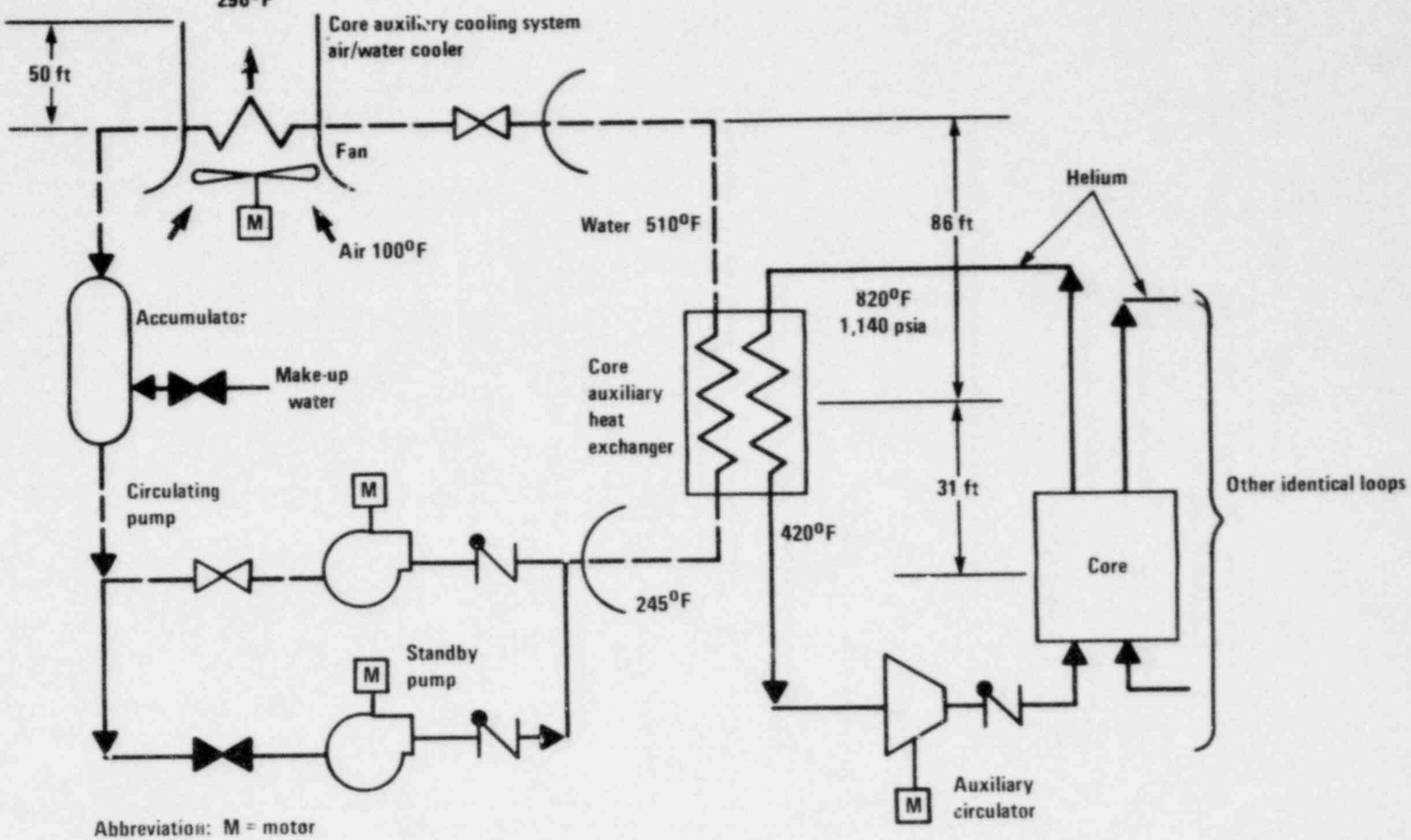



Figure 1-6. GCFR core auxiliary cooling system—pressurized natural convection cooling.

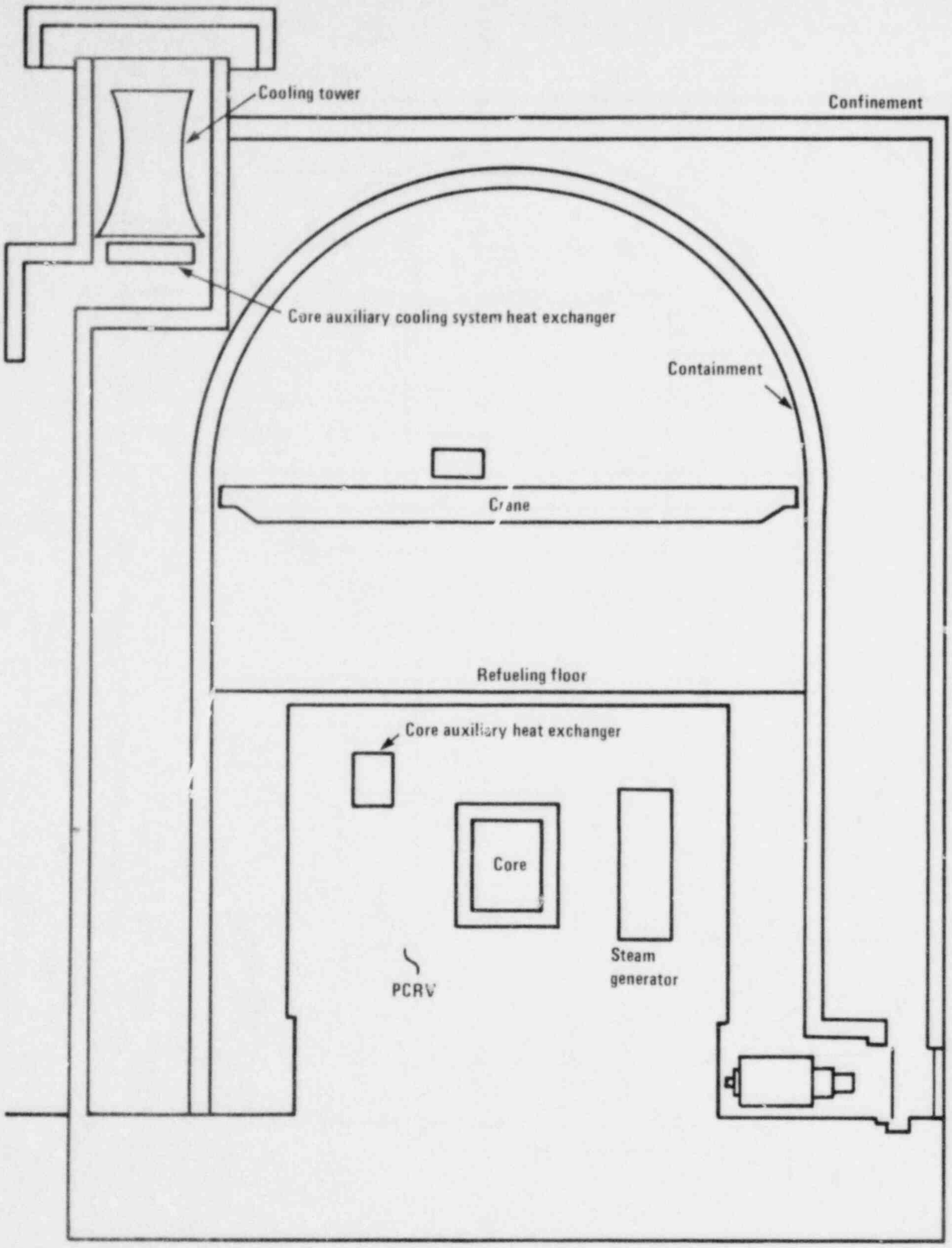



Figure 1-7. General arrangement of the air/water cooler relative to the core auxiliary heat exchanger and reactor core.

## REFERENCE FOR CHAPTER 1


1. GCFR Upflow/Downflow Study Summary Report, GA-A15455, General Atomic Company, in publication.

Chapter 2

# FUEL CYCLE: HOMOGENEOUS PLUTONIUM/URANIUM OXIDE CORE, THORIUM OXIDE AXIAL AND RADIAL BLANKETS, SPIKED RECYCLE

## 2.1 DESCRIPTION

This reactor/fuel-cycle combination is a gas-cooled fast reactor using a uranium/plutonium mixed-oxide homogeneous core and thorium oxide blankets. The core is coprocessed to recover plutonium mixed with uranium which is blended with makeup plutonium/uranium, 20% fissile, from a safe secure storage facility and with depleted uranium to attain the desired 14% fissile assay and quantity for feed to the fuel-fabrication operations. The core assemblies are preirradiated for spiking before shipment to the reactor. Depleted uranium is mixed with the recovered uranium-233 from blanket reprocessing to produce 12% fissile assay denatured uranium-233 for storage or sale. The thorium recovered from blanket reprocessing is sent to storage for a decay period of at least 10 years. New or decayed thorium is fabricated into blanket elements. Wastes from core fabrication and reprocessing are sent to a geologic waste repository. Wastes from blanket fabrication are sent to a shallow land disposal site.

The radial blanket consists of three rows. Row 1 contains 60 assemblies, row 2, 66 assemblies, and row 3, 72 assemblies. Every two cycles, 60 assemblies (30 each cycle) are discharged from the inner row (row 1), 60 assemblies are shuffled from row 2 to row 1, and 60 assemblies are shuffled from row 3 to row 2. Sixty fresh assemblies are then reloaded into row 3. In addition, once every six cycles, six assemblies are discharged from row 2; once every nine cycles, 12 assemblies are discharged from row 3. The axial blanket is an integral part of the fuel element and is replaced with the element.

Table 2-1 lists the general performance specifications for this reactor; Table 2-2 presents the reactor-design data specifications. The fuel-loading data are given in Table 2-3. Tables 2-4 and 2-5 give the fuel assembly volume fractions and the core-region volume fractions, respectively. A schematic diagram showing the dimensions and zones is presented in Figure 2-1, and a cross section of part of the core is shown in Figure 2-2. The entire core and blanket are assumed to be discharged at the end of the thirtieth year. The general performance specifications, reactor-design data, fuel-loading data, and core characteristics appearing in the above tables and figures are representative of the 1,200-MWe design for which data are available. However, for the purpose of comparing the environmental impacts of the GCFR reactor/fuel-cycle combination with those of the other concepts, pertinent data have been normalized to a 1,000-MWe design. This has been indicated in the figures and tables.

### 2.1.1 FUEL MANAGEMENT

The fuel-management information is summarized in Table 2-6. The isotopic distributions of the fuel inventory at the beginning and at the end of the equilibrium cycle are listed in Table 2-7. The total reactor charge and discharge data for a 30-year lifetime are given in Tables 2-8 and 2-9, respectively. The corresponding core charge and discharge data, axial-blanket charge and discharge data, and radial-blanket charge and discharge data are given in Tables 2-10 through 2-15, respectively.

The charge and discharge data are normalized to 1,000 MWe at a capacity factor of 75% and are shown in Table 2-16. The material-flow diagram for 0.75 GWe-yr is shown in Figure 2-3.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination, except for blanket fabrication, are discussed in the following sections of Volume VII:

| | |
|---|---|
| Blanket fabrication 1 | Chapter 4 |
| Core fabrication 2 | Chapter 4 |
| Reprocessing (Purex 2) | Section 5.2 |
| Reprocessing (Thorex 1) | Section 5.4 |
| Thorium storage | Section 6.1 |
| Plutonium storage | Section 6.2 |
| Depleted uranium storage | Section 6.4 |
| Uranium-233 storage | Section 6.5 |
| Waste disposal 2 | Section 7.2 |
| Waste disposal 3 | Section 7.3 |

Table 2-1. Generalized reactor-performance specifications: 1,200-MWe GCFR (plutonium/uranium core, all-thorium blankets)

| Power-plant performance parameters | |
|---|---|
| Reactor thermal power output, MW | 3,290 |
| Net electrical power output, MW | 1,200 |
| Plant heat rate, Btu/kW-hr | 9,360 |
| **Core design and performance parameters** | |
| Core heat output, MW | 3,165 + 125 blankets |
| Core volume, liters | 14,605 |
| Core loading, kg | |
| Heavy metal | 33,560 |
| Fissile fuel | 4,439 |
| Conversion ratio | 1.51 |
| Average discharge burnup, MWd/MTHM[a] | 81,000 |
| Peak discharge burnup, MWd/MTHM[a] | 92,000 |
| Fuel type | Oxide |
| Reactor inlet temperature, °F | 600 |
| Reactor outlet temperature, °F | 1,067 |
| End-of-cycle excess reactivity | 0 |

[a]Heavy-metal charged.

Table 2-2. Reactor design-data specifications: 1,200-MWe GCFR (plutonium/uranium core, all-thorium blankets)

| | | |
|---|---|---|
| Geometric information | | |
| Core height, cm | 127 | |
| Number of core enrichment zones | 4 | |
| Number of assemblies | 253 | |
| Equivalent diameters, cm | Figure 2-1 | |
| Number of pins per assembly | 324 | |
| Pin pitch-to-diameter ratio | 1.43 | |
| Overall assembly length, cm | 447 | |
| Lattice pitch, cm | 1.14 | |
| Assembly material | Type 316 stainless steel, 20% cold worked | |
| Cladding parameters | | |
| Cladding outside diameter, mils | 315 | |
| Cladding wall thickness, mils | 17 | |
| Cladding material | Type 316 stainless steel, 20% cold worked | |
| | Pu | U-233 |
| Fissile inventory at beginning of equilibrium cycle, kg | 4,450 | 647 |
| External fissile inventory, kg | 1,485 | 216 |
| Fissile gain or loss, kg/cycle | 175 (loss) | 525 (gain) |
| Specific power, kWt/kg fissile in core | 740 | |
| Power density, kWt/kg HM in core | 98 | |

Table 2-3. Fuel loading: 1,200-MWe GCFR, 253 core elements (plutonium/uranium oxide core, all-thorium blanket)

| Parameter | Zone 1 | Zone 2 | Zone 3 | Zone 4 |
|---|---|---|---|---|
| Fuel standard assemblies | 73 | 78 | 48 | 54 |
| Fuel standard assemblies loaded per cycle[a] | 27 | 21 | 21 | 15 |
| Fissile-material loading per cycle, kg[a] | 408 | 342 | 398 | 355 |
| Heavy-metal loading per cycle, kg[a] | 3,590 | 2,767 | 2,649 | 2,067 |
| Fuel residence time, effective full-power days | 825 | 825 | 825 | 825 |

[a]At equilibrium cycle (segment C).

Table 2-4. Fuel-assembly volume fractions:
1,200-MWe GCFR (plutonium/uranium core, all-thorium blankets)

| Component | Assembly type 1 | Blanket assembly type 2 | Control assembly (control in) |
|---|---|---|---|
| Fuel | 0.285 | 0.521 | 0.201 |
| Coolant | 0.577 | 0.377 | 0.456 |
| Structure | 0.138 | 0.102 | 0.134 |
| Control | -- | -- | 0.209 |
| Total | 1.000 | 1.000 | 1.000 |

Table 2-5. Core-region volume fractions:
1,200-MWe GCFR (plutonium/uranium core, all-thorium blankets)

| Component | Zone 1 | Zone 2 | Zone 3 | Zone 4 | Blanketed zone 5 | Reflector, zone 6 |
|---|---|---|---|---|---|---|
| Fuel | 0.277 | 0.272 | 0.264 | 0.285 | 0.521 | -- |
| Coolant | 0.565 | 0.558 | 0.558 | 0.577 | 0.377 | 0.100 |
| Structure | 0.138 | 0.137 | 0.137 | 0.138 | 0.102 | 0.900 |
| Control | 0.020 | 0.033 | 0.052 | 0 | 0 | |
| Total | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 |

Table 2-6. Fuel-management information
(GCFR uranium-plutonium/thorium GCFR spiked recycle)

| | | |
|---|---|---|
| Average capacity factor, % | 75 | |
| Approximate fraction of core replaced annually | 1/3 | |
| Lag time assumed between fuel discharge and recycle reload, years | 2 | |
| Fissile material reprocessing loss fraction, % | 1 | |
| Fissile material fabrication loss fraction, % | 1 | |
| Thorium dioxide requirements, MT/GWe | | |
| Initial core | 116 | |
| Annual equilibrium reload | 23 | |
| 30-year cumulative | 783 (gross)<br>16 (net) | |
| Separative-work requirements, $10^3$ SWU/GWe | Not applicable | |
| Requirements for special fuel materials, kg HM/GWe | **Fissile Pu** | **Fissile U-233** |
| Initial load | 3,700 | 0 |
| Annual equilibrium charge, discharge | 1,288/1,143 | 0/421 |
| 30-year cumulative | 4,350 (net) | 13,200 (produced) |
| Other data for proliferation-resistance assessment | | |
| Fresh- and discharge-fuel radiation level, R/hr at 1 meter | Approximately same as LMFBR | |
| Discharge-fuel energy-generation rate after 90-day cooling (W/hr-element) | Approximately same as LMFBR | |

Table 2-7. Fuel inventory at beginning and end of equilibrium cycle (1,200-MWe GCFR, 253 core elements, 31 control rods)

| Zone | Th-232 | Pa-233 | U-233 | U-232 | U-234 | U-235 | U-236 | U-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 | Fission products |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Fuel inventory (grams) | | | | | | | | | | | | |
| | Beginning-of-equilibrium cycle | | | | | | | | | | | | |
| 1 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.643+04 | 8.454-02 | 8.009+06 | 9.810+05 | 3.118+05 | 1.220+05 | 3.822+04 | 3.279+05 |
| 2 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.648+04 | 9.705+02 | 8.248+06 | 1.097+06 | 3.565+05 | 1.349+05 | 4.402+04 | 4.124+05 |
| 3 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.083+04 | 2.740+02 | 4.829+06 | 7.834+05 | 2.490+05 | 1.090+05 | 3.045+04 | 1.355+05 |
| 4 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.242+04 | 3.820+02 | 5.644+06 | 1.073+06 | 3.489+05 | 1.471+05 | 4.288+04 | 2.158+05 |
| 5 | 8.618+06 | 7.463+03 | 9.307+04 | 8.436+00 | 1.158+03 | 1.472+01 | 1.597-01 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 6.442+03 |
| 6 | 9.036+06 | 7.344+03 | 1.072+05 | 8.992+00 | 1.156+03 | 1.301+01 | 1.204-01 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 6.942+03 |
| 7 | 5.436+06 | 2.579+03 | 2.870+04 | 1.471+00 | 1.976+02 | 1.380+00 | 8.368-03 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.228+03 |
| 8 | 6.598+06 | 2.775+03 | 4.127+04 | 1.722+00 | 2.285+02 | 1.347+00 | 6.463-03 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.593+03 |
| 9 | 1.407+07 | 9.098+03 | 1.836+05 | 9.351+00 | 1.349+03 | 1.127+01 | 7.898-02 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.033+04 |
| 10 | 1.563+07 | 3.878+03 | 7.059+04 | 7.524-01 | 2.083+02 | 7.739-01 | 2.638-03 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.497+03 |
| 11 | 1.712+07 | 8.758+02 | 7.964+03 | 9.139-03 | 6.722+00 | 4.818-03 | 2.802-06 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 4.643+01 |
| 12 | 1.343+07 | 2.593+03 | 5.415+04 | 3.342-01 | 1.162+02 | 2.916-01 | 6.102-04 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 9.248+02 |
| 13 | 1.481+07 | 1.088+03 | 2.029+04 | 3.023-02 | 1.748+01 | 1.920-02 | 1.914-05 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.465+02 |
| 14 | 1.618+07 | 2.531+02 | 2.306+03 | 4.199-04 | 5.960-01 | 1.309-04 | 2.318-08 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 5.985+00 |
| | End-of-equilibrium cycle | | | | | | | | | | | | |
| 1 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.251+04 | 1.624+03 | 7.711+06 | 9.795+05 | 3.304+05 | 9.903+04 | 4.078+04 | 6.830+05 |
| 2 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.316+04 | 1.621+03 | 7.994+06 | 1.077+06 | 3.715+05 | 1.128+05 | 4.625+04 | 7.418+05 |
| 3 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 9.109+03 | 6.242+02 | 4.713+06 | 7.506+05 | 2.561+05 | 9.264+04 | 3.200+04 | 3.216+05 |
| 4 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.099+04 | 6.730+02 | 5.548+06 | 1.029+06 | 3.556+05 | 1.300+05 | 4.437+04 | 3.919+05 |
| 5 | 8.496+06 | 1.720+04 | 1.886+05 | 2.449+01 | 3.034+03 | 5.310+01 | 8.039-01 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.841+04 |
| 6 | 8.933+06 | 1.459+04 | 1.890+05 | 2.282+01 | 2.662+03 | 4.172+01 | 5.472-01 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 1.721+04 |
| 7 | 5.389+06 | 6.610+03 | 6.758+04 | 5.065+00 | 6.037+02 | 5.902+00 | 4.951-02 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 3.984+03 |
| 8 | 6.559+06 | 5.599+03 | 7.476+04 | 4.467+00 | 5.384+02 | 4.437+00 | 3.009-02 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 3.881+03 |
| 9 | 1.393+07 | 1.920+04 | 2.889+05 | 3.159+01 | 3.318+03 | 4.132+01 | 4.237-01 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 2.612+04 |
| 10 | 1.557+07 | 8.023+03 | 1.196+05 | 2.081+00 | 5.091+02 | 2.532+00 | 1.092-02 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 3.723+03 |
| 11 | 1.710+07 | 3.064+03 | 2.686+04 | 4.698-02 | 3.018+01 | 3.502-02 | 3.395-05 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 2.064+02 |
| 12 | 1.339+07 | 5.538+03 | 8.810+04 | 1.091+00 | 2.926+02 | 1.102+00 | 3.372-03 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 2.303+03 |
| 13 | 1.480+07 | 2.234+03 | 3.442+04 | 8.313-02 | 4.249+01 | 6.275-02 | 7.939-05 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 3.491+02 |
| 14 | 1.618+07 | 8.861+02 | 7.797+03 | 2.163-03 | 2.668+00 | 9.493-04 | 2.803-07 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 | 2.417+01 |

Note: $1.362+04 = 1.362 \times 10^4$.

Table 2-8. Total reactor-charge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Reactor charge (grams) | | | | | | | Total |
|---|---|---|---|---|---|---|---|---|
| | Th-232 | U-235 | U-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 | |
| 1 | 1.216+08 | 6.866+04 | 2.775+07 | 3.855+06 | 1.162+06 | 5.835+05 | 1.392+05 | 1.552+08 |
| 2 | 2.408+07 | 2.294+04 | 9.273+06 | 1.428+06 | 4.303+05 | 2.161+05 | 5.155+04 | 3.551+07 |
| 3 | 2.378+07 | 2.245+04 | 9.073+06 | 1.331+06 | 4.011+05 | 2.014+05 | 4.805+04 | 3.486+07 |
| 4 | 2.378+07 | 2.259+04 | 9.129+06 | 1.291+06 | 3.891+05 | 1.954+05 | 4.662+04 | 3.485+07 |
| 5 | 2.408+07 | 2.307+04 | 9.307+06 | 1.404+06 | 4.230+05 | 2.124+05 | 5.068+04 | 3.550+07 |
| 6 | 2.378+07 | 2.247+04 | 9.082+06 | 1.324+06 | 3.991+05 | 2.004+05 | 4.781+04 | 3.486+07 |
| 7 | 2.656+07 | 2.253+04 | 9.108+06 | 1.306+06 | 3.936+05 | 1.976+05 | 4.715+04 | 3.763+07 |
| 8 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 9 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 10 | 2.933+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 4.041+07 |
| 11 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 12 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 13 | 2.656+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 3.763+07 |
| 14 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 15 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 16 | 2.378+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 3.485+07 |
| 17 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.003+04 | 3.550+07 |
| 18 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 19 | 3.211+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 4.318+07 |
| 20 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 21 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 22 | 2.378+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 3.485+07 |
| 23 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 24 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 25 | 2.656+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 3.763+07 |
| 26 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 27 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |
| 28 | 2.933+07 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 4.041+07 |
| 29 | 2.408+07 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 3.550+07 |
| 30 | 2.378+07 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 3.486+07 |

Note: $1.144+08 = 1.144 \times 10^8$.

Table 2-9. Total reactor-discharge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Reactor discharge (grams) | | | | | | | | | | | | Total | Fission products |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Th-232 | Pa-233 | U-233 | U-232 | U-234 | U-235 | U-236 | U-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 | | |
| 1 | 2.389+07 | 2.835+04 | 1.631+05 | 8.355+00 | 1.171+03 | 1.887+04 | 9.468+02 | 9.149+06 | 1.300+06 | 4.168+05 | 1.655+05 | 5.148+04 | 3.518+07 | 3.778+05 |
| 2 | 2.339+07 | 2.731+04 | 3.351+05 | 3.285+01 | 3.566+03 | 1.482+04 | 1.622+03 | 8.627+06 | 1.226+06 | 4.164+05 | 1.335+05 | 5.188+04 | 3.422+07 | 7.357+05 |
| 3 | 2.326+07 | 2.681+04 | 4.424+05 | 5.557+01 | 5.996+03 | 1.216+04 | 2.110+03 | 8.378+06 | 1.201+06 | 4.302+05 | 1.158+05 | 5.336+04 | 3.392+07 | 1.076+06 |
| 4 | 2.354+07 | 2.681+04 | 4.647+05 | 5.677+01 | 6.170+03 | 1.266+04 | 2.070+03 | 8.512+06 | 1.307+06 | 4.788+05 | 1.326+05 | 5.979+04 | 3.454+07 | 1.123+06 |
| 5 | 2.322+07 | 2.684+04 | 4.727+05 | 5.908+01 | 6.395+03 | 1.219+04 | 2.061+03 | 8.310+06 | 1.234+06 | 4.485+05 | 1.222+05 | 5.589+04 | 3.391+07 | 1.096+06 |
| 6 | 2.595+07 | 2.776+04 | 5.168+05 | 6.130+01 | 6.803+03 | 1.228+04 | 2.073+03 | 8.363+06 | 1.214+06 | 4.362+05 | 1.186+05 | 5.423+04 | 3.670+07 | 1.078+06 |
| 7 | 2.351+07 | 2.680+04 | 4.829+05 | 5.956+01 | 6.459+03 | 1.276+04 | 2.069+03 | 8.548+06 | 1.295+06 | 4.709+05 | 1.306+05 | 5.875+04 | 3.455+07 | 1.108+06 |
| 8 | 2.321+07 | 2.683+04 | 4.835+05 | 6.084+01 | 6.582+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 9 | 2.872+07 | 2.749+04 | 5.221+05 | 6.117+01 | 6.719+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.947+07 | 1.084+06 |
| 10 | 2.351+07 | 2.679+04 | 4.840+05 | 5.981+01 | 6.480+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 11 | 2.321+07 | 2.682+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.126+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 12 | 2.595+07 | 2.775+04 | 5.189+05 | 6.200+01 | 6.842+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.669+07 | 1.085+06 |
| 13 | 2.351+07 | 2.680+04 | 4.839+05 | 5.992+01 | 6.479+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 14 | 2.321+07 | 2.682+04 | 4.840+05 | 6.083+01 | 6.590+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 15 | 2.321+07 | 2.683+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.392+07 | 1.083+06 |
| 16 | 2.351+07 | 2.679+04 | 4.840+05 | 5.981+01 | 6.480+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 17 | 2.321+07 | 2.682+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 18 | 3.146+07 | 2.841+04 | 5.571+05 | 6.223+01 | 6.972+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 4.225+07 | 1.086+06 |
| 19 | 2.351+07 | 2.680+04 | 4.839+05 | 5.992+01 | 6.479+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 20 | 2.321+07 | 2.682+04 | 4.840+05 | 6.083+01 | 6.590+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 21 | 2.321+07 | 2.683+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.392+07 | 1.083+06 |
| 22 | 2.351+07 | 2.679+04 | 4.840+05 | 5.981+01 | 6.480+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 23 | 2.321+07 | 2.682+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 24 | 2.595+07 | 2.775+04 | 5.189+05 | 6.200+01 | 6.842+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.669+07 | 1.085+06 |
| 25 | 2.351+07 | 2.680+04 | 4.839+05 | 5.992+01 | 6.479+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 26 | 2.321+07 | 2.682+04 | 4.840+05 | 6.083+01 | 6.590+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 27 | 2.872+07 | 2.749+04 | 5.221+05 | 6.117+01 | 6.719+03 | 1.225+04 | 2.068+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 3.947+07 | 1.084+06 |
| 28 | 2.351+07 | 2.679+04 | 4.840+05 | 5.981+01 | 6.480+03 | 1.278+04 | 2.072+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 3.455+07 | 1.103+06 |
| 29 | 2.321+07 | 2.682+04 | 4.839+05 | 6.094+01 | 6.589+03 | 1.222+04 | 2.062+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 3.391+07 | 1.092+06 |
| 30 | 1.203+08 | 8.295+04 | 1.114+06 | 9.229+01 | 1.116+04 | 4.595+04 | 4.540+03 | 2.597+07 | 3.838+06 | 1.311+06 | 4.349+05 | 1.630+05 | 1.533+08 | 2.205+06 |

Note: $2.250 + 07 = 2.250 \times 10^7$.

Table 2-10. Core-charge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Core charge (grams) | | | | | | |
|---|---|---|---|---|---|---|---|
| | U-235 | U-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 | Total |
| 1 | 6.866+04 | 2.775+07 | 3.855+06 | 1.162+06 | 5.835+05 | 1.392+05 | 3.356+07 |
| 2 | 2.294+04 | 9.273+06 | 1.428+06 | 4.303+05 | 2.161+05 | 5.155+04 | 1.142+07 |
| 3 | 2.245+04 | 9.073+06 | 1.331+06 | 4.011+05 | 2.014+05 | 4.805+04 | 1.108+07 |
| 4 | 2.259+04 | 9.129+06 | 1.291+06 | 3.891+05 | 1.954+05 | 4.662+04 | 1.107+07 |
| 5 | 2.303+04 | 9.307+06 | 1.404+06 | 4.230+05 | 2.124+05 | 5.068+04 | 1.142+07 |
| 6 | 2.247+04 | 9.082+06 | 1.324+06 | 3.991+05 | 2.004+05 | 4.781+04 | 1.108+07 |
| 7 | 2.253+04 | 9.108+06 | 1.306+06 | 3.936+05 | 1.976+05 | 4.715+04 | 1.107+07 |
| 8 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 9 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 10 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 11 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 12 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 13 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 14 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 15 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 16 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 17 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 18 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 19 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 20 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 21 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 22 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 23 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 24 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 25 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 26 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 27 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |
| 28 | 2.251+04 | 9.098+06 | 1.313+06 | 3.957+05 | 1.987+05 | 4.740+04 | 1.108+07 |
| 29 | 2.306+04 | 9.321+06 | 1.394+06 | 4.201+05 | 2.110+05 | 5.033+04 | 1.142+07 |
| 30 | 2.248+04 | 9.085+06 | 1.322+06 | 3.984+05 | 2.001+05 | 4.773+04 | 1.108+07 |

Note: 5.833+04 = 5.833 x $10^4$.

Table 2-11. Core-discharge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Core discharge (grams) | | | | | | | | Fission products |
|---|---|---|---|---|---|---|---|---|---|
| | U-235 | U-236 | U-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 | Total | |
| 1 | 1.886+04 | 9.468+02 | 9.149+06 | 1.300+06 | 4.168+05 | 1.655+05 | 5.148+04 | 1.110+07 | 3.712+05 |
| 2 | 1.478+04 | 1.622+03 | 8.627+06 | 1.226+06 | 4.164+05 | 1.335+05 | 5.188+04 | 1.047+07 | 7.124+05 |
| 3 | 1.206+04 | 2.108+03 | 8.378+06 | 1.201+06 | 4.302+05 | 1.158+05 | 5.336+04 | 1.019+07 | 1.035+06 |
| 4 | 1.256+04 | 2.069+03 | 8.512+06 | 1.307+06 | 4.788+05 | 1.326+05 | 5.979+04 | 1.050+07 | 1.080+06 |
| 5 | 1.208+04 | 2.060+03 | 8.310+06 | 1.234+06 | 4.485+05 | 1.222+05 | 5.589+04 | 1.018+07 | 1.050+06 |
| 6 | 1.217+04 | 2.071+03 | 8.363+06 | 1.214+06 | 4.362+05 | 1.186+05 | 5.423+04 | 1.020+07 | 1.029+06 |
| 7 | 1.265+04 | 2.067+03 | 8.548+06 | 1.295+06 | 4.709+05 | 1.306+05 | 5.875+04 | 1.052+07 | 1.062+06 |
| 8 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 9 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+66 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 10 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 11 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 12 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 13 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 14 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 15 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 16 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 17 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 18 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 19 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 20 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 21 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 22 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 23 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 24 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 25 | 1.267+04 | 2.070+03 | 8.561+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 26 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 27 | 1.214+04 | 2.066+03 | 8.344+06 | 1.222+06 | 4.408+05 | 1.200+05 | 5.485+04 | 1.019+07 | 1.036+06 |
| 28 | 1.267+04 | 2.070+03 | 8.[illegible]+06 | 1.290+06 | 4.679+05 | 1.297+05 | 5.834+04 | 1.052+07 | 1.057+06 |
| 29 | 1.211+04 | 2.060+03 | 8.320+06 | 1.231+06 | 4.464+05 | 1.216+05 | 5.561+04 | 1.019+07 | 1.045+06 |
| 30 | 4.580+04 | 4.538+03 | 2.597+07 | 3.838+06 | 1.311+06 | 4.349+05 | 1.630+05 | 3.177+07 | 2.128+06 |

Note: 1.575+04 = 1.575 x $10^4$.

Table 2-12. Axial-blanket thorium-232 charge (1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Thorium-232 charge (grams) |
|---|---|
| 1 | 3.000+07[a] |
| 2 | 1.020+07 |
| 3 | 9.898+06 |
| 4 | 9.898+06 |
| 5 | 1.020+07 |
| 6 | 9.898+06 |
| 7 | 9.898+06 |
| 8 | 1.020+07 |
| 9 | 9.898+06 |
| 10 | 9.898+06 |
| 11 | 1.020+07 |
| 12 | 9.898+06 |
| 13 | 9.898+06 |
| 14 | 1.020+07 |
| 15 | 9.898+06 |
| 16 | 9.898+06 |
| 17 | 1.020+07 |
| 18 | 9.898+06 |
| 19 | 9.898+06 |
| 20 | 1.020+07 |
| 21 | 9.898+06 |
| 22 | 9.898+06 |
| 23 | 1.020+07 |
| 24 | 9.898+06 |
| 25 | 9.898+06 |
| 26 | 1.020+07 |
| 27 | 9.898+06 |
| 28 | 9.898+06 |
| 29 | 1.020+07 |
| 30 | 9.898+06 |

Note: 3.000+07 = 3.000 x $10^7$.

Table 2-13. Axial-blanket discharge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Axial-blanket discharge (grams) | | | | | | | | Fission products |
|---|---|---|---|---|---|---|---|---|---|
| | Th-232 | Pa-233 | U-233 | U-232 | U-234 | U-235 | U-236 | Total | |
| 1 | 1.009+07[a] | 1.556+04 | 8.919+04 | 4.512+00 | 7.434+02 | 5.238+00 | 3.137-02 | 1.020+07 | 3.774+03 |
| 2 | 9.683+06 | 1.493+04 | 1.820+05 | 1.768+01 | 2.262+03 | 3.024+01 | 3.354-01 | 9.882+06 | 1.372+04 |
| 3 | 9.580+06 | 1.445+04 | 2.646+05 | 3.800+01 | 4.354+03 | 8.327+01 | 1.337+00 | 9.864+06 | 2.885+04 |
| 4 | 9.892+06 | 1.448+04 | 2.605+05 | 3.622+01 | 4.102+03 | 7.528+01 | 1.158+00 | 1.017+07 | 2.736+04 |
| 5 | 9.588+06 | 1.450+04 | 2.591+05 | 3.691+01 | 4.161+03 | 7.754+01 | 1.212+00 | 9.865+06 | 2.771+04 |
| 6 | 9.588+06 | 1.451+04 | 2.588+05 | 3.677+01 | 4.148+03 | 7.708+01 | 1.201+00 | 9.865+06 | 2.763+04 |
| 7 | 9.894+06 | 1.447+04 | 2.587+05 | 3.573+01 | 4.036+03 | 7.327+01 | 1.113+00 | 1.017+07 | 2.696+04 |
| 8 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 9 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 10 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 11 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 12 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 13 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 14 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 15 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 16 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 17 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 18 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 19 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 20 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | [illegible].761+04 |
| 21 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 22 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 23 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 24 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 25 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 26 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 27 | 9.588+06 | 1.451+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.200+00 | 9.865+06 | 2.761+04 |
| 28 | 9.894+06 | 1.447+04 | 2.587+05 | 3.571+01 | 4.035+03 | 7.324+01 | 1.113+00 | 1.017+07 | 2.695+04 |
| 29 | 9.588+06 | 1.450+04 | 2.587+05 | 3.673+01 | 4.145+03 | 7.699+01 | 1.199+00 | 9.865+06 | 2.761+04 |
| 30 | 2.938+07 | 4.397+04 | 5.196+05 | 5.736+01 | 6.885+03 | 1.074+02 | 1.494+00 | 2.995+07 | 4.381+04 |

Note: 1.009+07 = $1.009 \times 10^7$.

Table 2-14. Radial-blanket thorium-232 charge (1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Thorium-232 charge (grams) |
|---|---|
| 1 | 9.162+07 |
| 2 | 1.388+07 |
| 3 | 1.388+07 |
| 4 | 1.388+07 |
| 5 | 1.388+07 |
| 6 | 1.388+07 |
| 7 | 1.666+07 |
| 8 | 1.388+07 |
| 9 | 1.388+07 |
| 10 | 1.943+07 |
| 11 | 1.388+07 |
| 12 | 1.388+07 |
| 13 | 1.666+07 |
| 14 | 1.388+07 |
| 15 | 1.388+07 |
| 16 | 1.388+07 |
| 17 | 1.388+07 |
| 18 | 1.388+07 |
| 19 | 2.221+07 |
| 20 | 1.388+07 |
| 21 | 1.388+07 |
| 22 | 1.388+07 |
| 23 | 1.388+07 |
| 24 | 1.388+07 |
| 25 | 1.666+07 |
| 26 | 1.388+07 |
| 27 | 1.388+07 |
| 28 | 1.943+07 |
| 29 | 1.388+07 |
| 30 | 1.388+07 |

Note: 1.379+07 = $1.379 \times 10^7$.

Table 2-15. Radial-blanket discharge data
(1,200-MWe GCFR, 253 core elements, 31 control rods)

| Year | Radial-blanket discharge (grams) | | | | | | | | Fission products |
|---|---|---|---|---|---|---|---|---|---|
| | Th-232 | Pa-233 | U-233 | U-232 | U-234 | U-235 | U-236 | Total | |
| 1 | 1.379+07[a] | 1.279+04 | 7.391+04 | 3.843+00 | 4.272+02 | 2.224+00 | 9.907-03 | 1.388+07 | 2.846+03 |
| 2 | 1.370+07 | 1.238+04 | 1.531+05 | 1.517+01 | 1.304+03 | 1.267+01 | 1.020-07 | 1.387+07 | 9.516+03 |
| 3 | 1.368+07 | 1.226+04 | 1.778+05 | 1.756+01 | 1.642+03 | 1.775+01 | 1.584-01 | 1.387+07 | 1.245+04 |
| 4 | 1.364+07 | 1.233+04 | 2.042+05 | 2.055+01 | 2.069+03 | 2.511+01 | 2.527-01 | 1.386+07 | 1.605+04 |
| 5 | 1.363+07 | 1.235+04 | 2.137+05 | 2.217+01 | 2.234+03 | 2.814+01 | 2.938-01 | 1.386+07 | 1.752+04 |
| 6 | 1.636+07 | 1.325+04 | 2.580+05 | 2.453+01 | 2.655+03 | 3.389+01 | 3.610-01 | 1.663+07 | 2.090+04 |
| 7 | 1.362+07 | 1.232+04 | 2.241+05 | 2.384+01 | 2.423+03 | 3.178+01 | 3.460-01 | 1.386+07 | 1.918+04 |
| 8 | 1.362+07 | 1.234+04 | 2.248+05 | 2.412+01 | 2.437+03 | 3.206+01 | 3.502-01 | 1.386+07 | 1.932+04 |
| 9 | 1.913+07 | 1.298+04 | 2.634+05 | 2.444+01 | 2.573+03 | 3.287+01 | 3.550-01 | 1.941+07 | 2.034+04 |
| 10 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 11 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 12 | 1.636+07 | 1.324+04 | 2.602+05 | 2.527+01 | 2.697+03 | 3.474+01 | 3.737-01 | 1.663+07 | 2.128+04 |
| 13 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 14 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 15 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 16 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 17 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 18 | 2.187+07 | 1.390+04 | 2.984+05 | 2.550+01 | 2.827+03 | 3.541+01 | 3.765-01 | 2.218+07 | 2.225+04 |
| 19 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 20 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 21 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 22 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 23 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 24 | 1.636+07 | 1.324+04 | 2.602+05 | 2.527+01 | 2.697+03 | 3.474+01 | 3.737-01 | 1.663+07 | 2.128+04 |
| 25 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 26 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 27 | 1.913+07 | 1.298+04 | 2.634+05 | 2.444+01 | 2.573+03 | 3.287+01 | 3.550-01 | 1.941+07 | 2.034+04 |
| 28 | 1.362+07 | 1.232+04 | 2.253+05 | 2.410+01 | 2.445+03 | 3.223+01 | 3.526-01 | 1.386+07 | 1.938+04 |
| 29 | 1.362+07 | 1.233+04 | 2.252+05 | 2.421+01 | 2.444+03 | 3.220+01 | 3.522-01 | 1.386+07 | 1.938+04 |
| 30 | 9.094+07 | 3.897+04 | 5.941+05 | 3.492+01 | 4.272+03 | 4.523+01 | 4.373-01 | 9.158+07 | 3.330+04 |

Note: $3.000+07 = 3.000 \times 10^7$.

Table 2-16. Charge and discharge data normalized[a] to 1,000 MWe at a 75% capacity factor

(CCFR uranium-plutonium/thorium spiked recycle)

| Isotope | Charge (kg/0.75 GWe) Core | Charge Axial blanket | Charge Radial blanket | Discharge (kg/0.75 GWe-yr) Core | Discharge Axial blanket | Discharge Radial blanket |
|---|---|---|---|---|---|---|
| Thorium-232 | | 8,332 | 11,567 | | 8,075 | 11,350 |
| Protactinium-233 | | | | | 12.08 | 10.27 |
| Uranium-232 | | | | | 0.030 | 0.02 |
| Uranium-233 | | | | | 215.6 | 187.7 |
| Uranium-234 | | | | | 3.4 | 2.04 |
| Uranium-235 | 18.9 | | | 10.26 | 0.06 | 0.027 |
| Uranium-236 | | | | 1.72 | 0.001 | 0.0003 |
| Uranium-238 | 6,807 | | | 7,007 | | |
| Plutonium-239 | 1,119 | | | 1,040 | | |
| Plutonium-240 | 337.3 | | | 376.4 | | |
| Plutonium-241 | 169.4 | | | 103.1 | | |
| Plutonium-242 | 40.4 | | | 46.9 | | |
| Total heavy metal | 9,328 | 8,332 | 11,567 | 8,583 | 8,306 | 11,550 |
| Fission products | | | | 871.6 | 22.8 | 16.2 |

[a]Average of charge/discharge data for years 20, 21, 22 (Tables 2-10 through 2-15) normalized from a 1,200-MWe reactor.

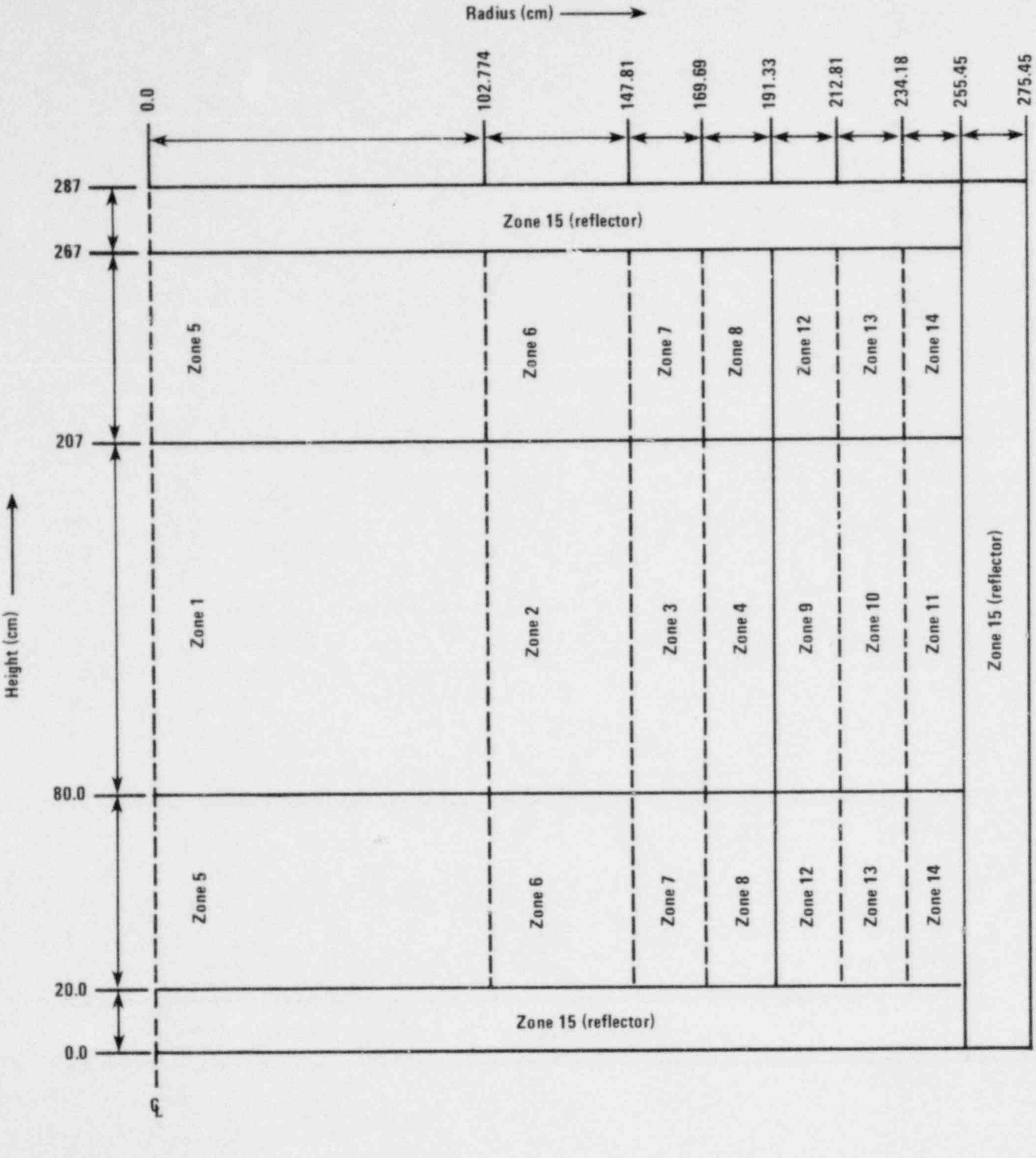



Figure 2-1. Schematic diagram of advanced 1,200-MWe GCFR (253 core elements, plutonium/uranium oxide core, all-thorium blankets).



1072 v

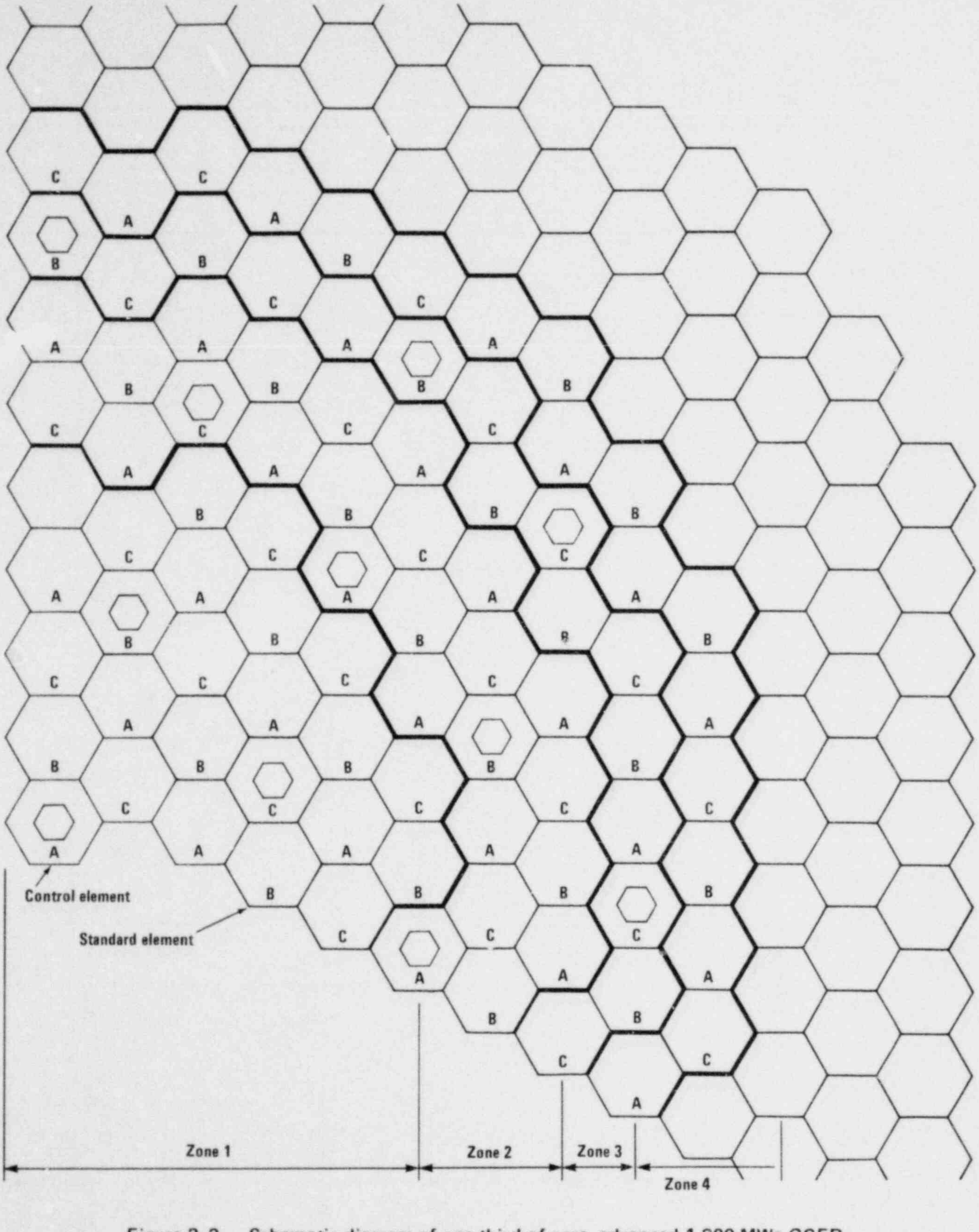



Figure 2-2. Schematic diagram of one-third of core, advanced 1,200-MWe GCFR (253 core elements, plutonium/uranium oxide core, thorium oxide axial and radial blankets).



1072 v

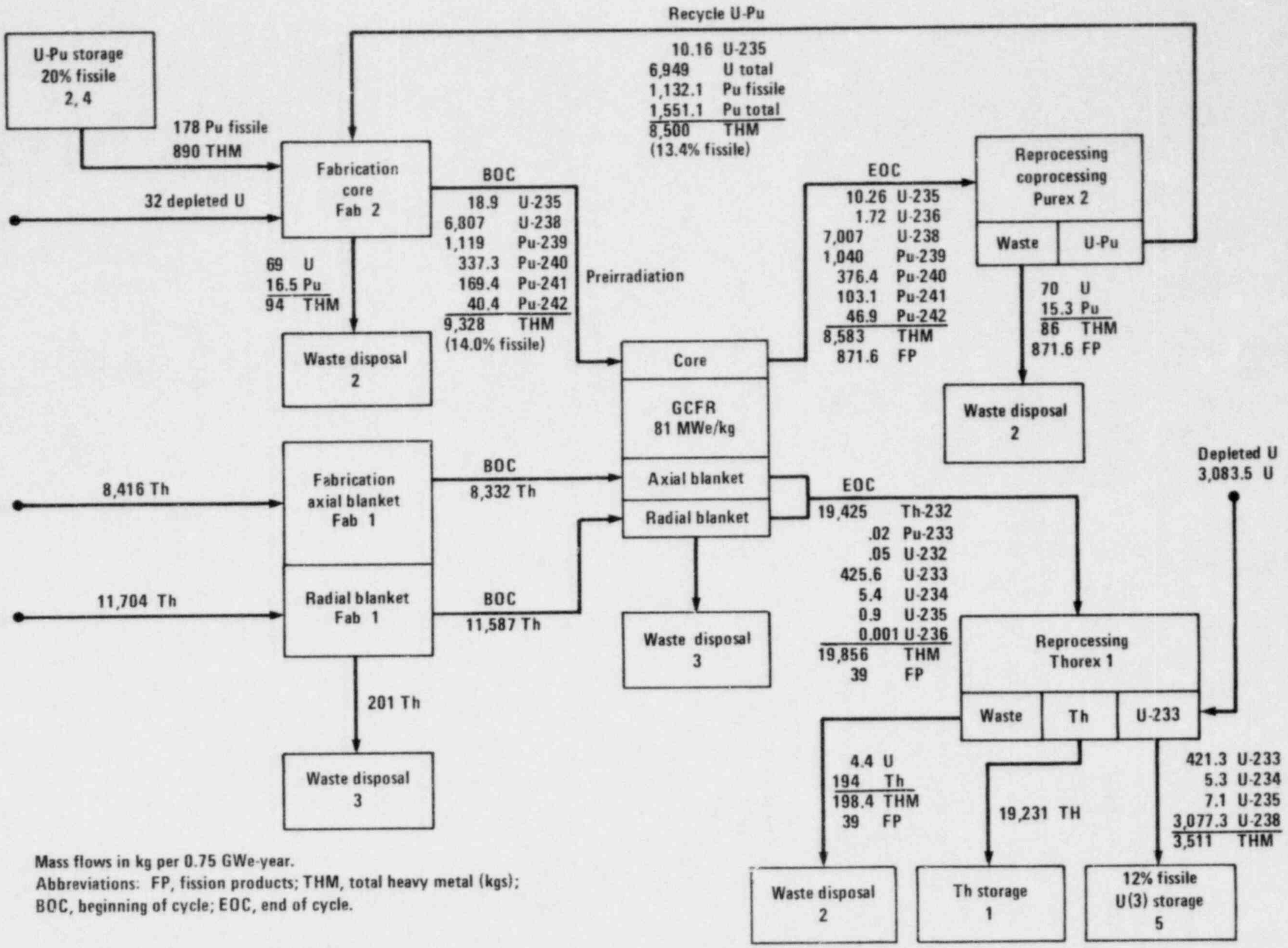



Figure 2-3. Material flow diagram for GCFR uranium/plutonium core, thorium blankets, spiked recycle.

## 2.2 SAFETY CONSIDERATIONS

In August 1974, the U.S. Atomic Energy Commission (AEC) published a Preapplication Safety Evaluation Report (Ref. 1) for the GCFR concept, based on the GCFR Preliminary Safety Information Document (PSID) (Ref. 2). It concluded conditionally that the plant proposed in concept by the General Atomic Company can be designed, constructed, and operated without undue risk to the health and safety of the public. This general conclusion is similar to that reached for the LMFBR demonstration plant.

The major safety considerations identified in the AEC safety evaluation of the GCFR and considered to be unresolved are as follows:

1. Acceptable power-density levels and thermal margins
2. Definition of depressurization accidents
3. Definition of core-disruptive accidents
4. Diversity in reactor-shutdown system
5. Adequacy of core cooling
6. Containment-system design
7. Fuel design
8. Nuclear design
9. Prestressed-concrete reactor vessel
10. Generic scale-up of nuclear design areas and analyses of core-disruptive accidents
11. Primary-system components
12. Accident-analysis studies
13. In-service inspection

The GCFR design studies and research programs are progressing, and many of the conditions in the preapplication safety evaluation report are being satisfied. In particular, progress is being made in the key areas of core-cooling reliability, thermal margins, and core-disruptive accidents. Pertinent information is summarized below. Appendix B provides further discussion of U.S. Nuclear Regulatory Commission (NRC) safety considerations.

### 2.2.1 THERMAL MARGINS

The GCFR program is designed to use the core and fuel technology of the LMFBR and the primary-system-component technology of the high-temperature gas-cooled reactor (HTGR). Thus the GCFR cladding thermal limit (700°C) for normal operation has been established to be in the range encountered in LMFBR technology, rather than that of the HTGR. A program to develop the GCFR fuel design, including establishing thermal limits and emphasizing areas of difference between LMFBR and GCFR design, was formulated in 1975 and is continuing. The first major irradiation test to demonstrate the GCFR normal-operation thermal limit was recently completed successfully in the EBR II reactor, where a burnup of 14 atom% was achieved without any indications of failure. As part of the GCFR fuel-design program, a core flow test loop (CFTL) has been scheduled for construction at the Oak Ridge National Laboratory (ORNL). The CFTL is currently in the Title II design and component-development phase. When completed, this out-of-reactor facility will perform steady-state and transient tests on rod bundles for normal and abnormal plant conditions.

The most severe design-basis event, the design-basis depressurization accident (DBDA), was the topic of an amendment to the GCFR PSID in the first part of 1976. In this amendment, the General Atomic Company examined a DBDA flow area of 75 square inches, which is a threefold increase in the DBDA flow area over the design value (25 square inches) used at the time of the 1974 Safety Evaluation Report. To match this change in flow area, the CACS has been redesigned, and a comprehensive set of analyses of the new reference DBDA as well as a wide range of conservative sensitivity analyses has been completed. These analyses showed that, even under the most conservative assumptions, the cladding temperature stayed below 2,300°F, providing a margin of several hundred degrees below cladding melting. This is comparable with accepted light-water reactor (LWR) cladding-temperature limits for LOCA events.

### 2.2.2 CORE-DISRUPTIVE ACCIDENTS

Since the completion of the NRC preapplication review in 1974, a comprehensive GCFR safety program has been initiated to investigate, through mechanistic analyses and supporting experimental programs, all classes of core-disruptive accidents (CDA) applicable to the GCFR, including the following:

1. Complete flow blockage in a single subassembly
2. Total loss of forced circulation with reactor shutdown
3. Loss of flow without shutdown
4. Continued reactivity insertion without shutdown

Participants in this safety program include the Argonne National Laboratory, the Los Alamos Scientific Laboratory, the Idaho National Engineering Laboratory, as well as the General Atomic Company.

Analytical results to date for these accident classes are most favorable to the GCFR design, indicating only moderate fuel vapor fractions and a potential for mechanical work well within the structural integrity limits of the PCRV. The latter safety margin was experimentally confirmed by tests-to-failure run by the Naval Ordnance Laboratory on one-twentieth scale models of the PCRV in 1976 and was reported in Reference 3. The analytical results for the fuel vapor fractions generated during various core-disruptive accidents will be confirmed by a series of out-of-reactor tests followed by integrated in-reactor tests.

Supporting out-of-reactor experiment programs for fuel behavior under CDA conditions include direct electric heating tests and thermal fuel motion tests at the Argonne National Laboratory and the program on electrically heated rod bundles at the Los Alamos Scientific Laboratory. These programs are all in the test initiation phase. The full-size-bundle experiments are scheduled to begin at Los Alamos in late 1979. The in-reactor test program includes the helium-circulating GRIST-2 facility to be installed in the TREAT upgrade reactor. This program is currently in the Title I design phase. The capability of the GCFR containment system designs--both the PCRV and the secondary containment--to mitigate the consequences of core-melt and core-disruptive accidents is also under detailed study as an integral part of the GCFR safety program.

Studies by the General Atomic Company and the Argonne National Laboratory, and laboratory-scale experiments by the Argonne National Laboratory, are continuing to assess the GCFR design for its ability to contain the products of a core-meltdown accident. General Atomic analyses of various in-vessel containment designs include the cooled-crucible, the solution-bath (BORAX), and the heavy-metal-bath concepts. The detailed design work on the selected concept will continue through fiscal year 1984. The Argonne National Laboratory is continuing with analytical and experimental work to define the time-dependent thermal behavior of molten-fuel pools and their chemical reactions with both the molten-core-container materials and with the materials proposed for the GCFR lower shield and the PCRV. This work is also expected to continue through fiscal year 1984.

### 2.2.3 ADEQUACY OF CORE COOLING

Core-cooling reliability for decay-heat removal is recognized as a critical GCFR safety concern. Since the NRC preapplication review, detailed reliability analyses of the GCFR core-cooling system designs have been conducted at the General Atomic Company and at the Massachusetts Institute of Technology. These studies have shown that the designs for the GCFR core-cooling system can have a very high reliability, but they have also revealed a number of areas where shutdown cooling reliability can be improved. These improvements are being made. Methods have been established to make reliability considerations an integral part of the design process, rather than merely a check function, in order to insure highly reliable GCFR cooling systems.

In particular, the following GCFR core-cooling system design changes have been made since the preapplication review:

1. A new or second safety class cooling system has been incorporated into the GCFR design. The SCS is capable of responding to all frequent events and providing core residual heat removal for an indefinite period. This system is independent of the second safety class core-cooling system, the CACS.
2. The diversity and reliability of the core cooling have also been enhanced by requiring the system design to incorporate design features that will enable the CACS to operate in a natural circulation mode. For residual heat removal, it is a design objective that the CACS be operational in a natural convection mode with either little or preferably no power activation of equipment.

The design features described above have significantly enhanced the adequacy of GCFR core cooling.

## 2.3 ENVIRONMENTAL CONSIDERATIONS

### 2.3.1 SUMMARY ASSESSMENT

The thermal impacts of the GCFR are less than those of the reference LWR because the GCFR has a higher thermal efficiency and rejects less heat for a given amount of electricity generated. Releases of chemicals and biocides are similar in kind and quantity to those from the reference LWR; hence the impacts are similar. The estimated releases of radioactivity (and corresponding impacts) are much smaller than those for the reference LWR. The GCFR estimates, however, are based on design values only, but actual experience is available for the reference LWR. In summary, the estimated impacts from normal operation of this reactor are less than those from the reference LWR under the conditions of the comparison.

### 2.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM

In the GCFR system, superheated steam is produced in the steam generators. Thus, the plant has a higher efficiency than an LWR, which produces saturated steam (37% vs. 33%).

A design study of a 1,000-MWe GCFR conducted in the Federal Republic of Germany, a similar design study by the European Association for the Gas-Cooled Breeder Reactor, and a preliminary environmental report prepared by the General Atomic Company for a 300-MWe demonstration-plant gas-cooled fast-breeder reactor (Ref. 4) were selected as models to provide data on the GCFR system. Basic parameters describing the plant are given in Table 2-17.

### 2.3.3 STATION LAND USE

There are no outstanding features of the GCFR concept that would indicate differences in land use from that of LWR plants.

Comparison of various sites for LWRs shows that there is a wide variation in land requirements. This variation results from differences in specific site characteristics and specific plant-design features. Similar differences would be expected for various GCFR designs and sites.

### 2.3.4 STATION WATER USE

The principal single use of water, as in the reference LWR, is for makeup to the heat-dissipation system. Much smaller amounts are required for the plant (after demineralization) as well as for such uses as laundry, showers, and sanitary facilities (Table 2-18). Compared to the reference LWR, the annual average quantity of makeup water required is approximately 70%.

### 2.3.5 HEAT-DISSIPATION SYSTEM

Any of various heat-dissipation systems may be used for the GCFR, depending on site conditions and other factors. One of the more commonly used is a wet natural-draft cooling tower. This system, with freshwater makeup, was assumed for this study, as it was for the reference LWR.

A typical natural-draft cooling tower for a 1,000-MWe GCFR unit has a single shell with a height of about 500 feet and a maximum shell diameter of about 380 feet (slightly smaller than that of the reference LWR). Heat is dissipated to the atmosphere by evaporation and by sensible heat transfer. The balance depends on air temperature and humidity, although evaporation is much more significant. The average rate of water use therefore varies from month to month. Compared to the reference LWR the heat-dissipation rate for the GCFR is approximately 85%. Blowdown is required to limit the concentration of solids in the circulating water. For the reference plant discussed here, a maximum concentration of 5 is used, although other values are frequently found. The same value has been used for the reference LWR. Design data for the heat-dissipation system are presented in Table 2-19 for a site in the north-central United States.

Circulating water is periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine is added as required to achieve a free residual chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual chlorine content during periods of chlorination.

## 2.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

Sources of radioactivity, release paths, and processing systems are described briefly in the sections that follow. Quantities of radioactivity released, taken from Reference 4 and normalized to 1,000 MWe, are also tabulated below.

### 2.3.6.1 Source Terms

In the GCFR system, radioactive materials are produced by fission and by neutron activation of impurities in the helium primary coolant.

The fuel-pin venting system (Figure 2-4) maintains the gas pressure inside the pin at a point just below the pressure of the coolant. As shown in Figure 2-5, the interior of each fuel pin is connected to a venting bypass between the fuel-element outlet and the element head through a suction hole. By this means, the pressure inside the fuel pins is held between the coolant pressures at the fuel-element outlet and at the circulator inlet. The system also continuously removes fission products released from the fuel pellets and sweeps them off to a helium-purification system, where fission products are removed by the fission-gas separator. Fission products and other impurities in the reactor coolant stream also are removed by the helium-purification system. The calculated release and venting fractions in GCFR fuel elements are given in Table 2-20.

Every 6 months, the helium-purification system must be regenerated. It is estimated that 20,000 scf of gas consisting of purified helium and a small volume fraction of radioactive impurities will be transferred to the gaseous-radwaste system in each regeneration.

During refueling operations, several components must be evacuated and purged. This results in approximately 48,600 scf of gas being transferred to the waste-gas-holding system and then to the gas-recovery system.

Continuous sampling of the coolant is required at the rate of 0.08 scfm. Samples are analyzed and transferred first to the holding system and then to the gas-recovery system.

Radioactive sources include the fuel-pin pressure-equalization system, the helium-purification system, fuel-handling-system purges, coolant sampling effluent, liquid-waste-tank vents, low-temperature-absorber vents, and displaced air at the drumming station. Unprocessed sources include tritiated water vapor from the secondary coolant, coolant leakage from the PCRV to the containment building, and fuel-element leakage to the spent-fuel pool. Other sources release negligible amounts of radioactivity.

The pathways of radioactive release to the environment during normal plant operation are shown in Figure 2-6. The pathways of radioactive releases to the environment through the gaseous-radwaste system are illustrated in Figure 2-7. (The factors involved in the pathways are defined and quantified in Tables 2-21 and 2-22.) Source activities calculated with a modified RAD2 computer code (RAD2C) and normalized to 1,000 MWe are listed in Table 2-23. Calculated activities are based on 24 effective full-power years of operation assuming a 30-year plant life and thus indicate the total buildup of long-lived isotopes that would be accumulated over the life of the plant. The activity levels of shorter lived isotopes are independent of times longer than a few half-lives. Plateout activities of iodine and other volatiles over the plant lifetime are also given. (Note that releases to the environment are given in curies per year.)

#### 2.3.6.2 Gaseous-Radwaste System

The gaseous-radwaste system is designed to collect potentially radioactive gases generated from the various sources enumerated above during plant operation. These gases will either be vented to the environment, retained within the helium-purification system, held for radioactive decay before disposal, or bottled for onsite storage. The choice depends on the level of activity present. Subsystems of the gaseous-radwaste system are conceptualized in Figure 2-8.

Gaseous effluents released from the fuel include helium, hydrogen, tritium, oxygen, carbon dioxide, carbon monoxide, krypton, xenon, and iodine. Only helium is returned to the reactor coolant. Tritium is removed from the coolant by oxidation into the chemical form of water, followed by suitable disposal as tritiated concrete castings.

#### 2.3.6.3 Liquid-Radwaste System

The liquid-radwaste system is designed to collect potentially radioactive liquids generated during plant operation and then either store, process, or dispose of them. Most of the liquid wastes generated in a plant result from decontamination operations, showers, laundry, process sampling, or maintenance liquid releases. Only limited quantities of other waste liquids are produced.

Liquid wastes with low levels of total dissolved solids are normally filtered, demineralized, and collected in waste-monitoring tanks. Those with high levels of total dissolved solids are normally filtered, treated by reverse osmosis and ion exchange, and collected in waste-monitoring tanks, where they are sampled before being recycled or discharged. Waste concentrates are ultimately processed in the solid-waste system.

The liquid-radwaste system is similar to that of the HTGR. As shown in Figure 2-9, liquid wastes from the containment and service buildings are drained to liquid-waste-storage tanks. Low-activity waste is then routed to the cooling-tower blowdown, and high-activity waste is transferred to the liquid-waste-processing system. Other liquid wastes from the radiochemistry laboratory, the helium-purification system, the gas-recovery system, and the decontamination system are collected in holding tanks for solidification and storage.

2.3.6.4 Solid-Radwaste System

The solid-radwaste system is designed to process, package, and store for ultimate disposal the solid radioactive waste generated during plant operation and maintenance. The wastes include tritiated concrete, reverse-osmosis concentrates, used filters, demineralizer resins, and other contaminated solid refuse. The solid-radwaste system is similar to that shown in Figure 2-10. The total volume of low-level solid waste is expected to be approximately 1,400 $ft^3$ annually with an activity content of 280 curies.

2.3.6.5 Comparison with Predicted Releases from Other Studies

Other studies have been made of potential radioactive-release rates from the normal operation of nuclear power plants. These studies have covered a variety of reactor and plant designs, assumptions, and calculation techniques. The results, in terms of liquid and gaseous releases, are shown in Tables 2-24 and 2-25, respectively. It should be noted that in the comparison of gaseous releases in Table 2-25, the reference GCFR plant employs a gaseous-waste-treatment system with onsite storage of noble gases, whereas the reference HTGR and LWR rely on temporary storage and release to the environment. A slight leakage of krypton-85 from the storage area may be postulated, but even this release would be small in comparison with that for the HTGR and LWR reference plants. It should be noted that there is considerable variation among the predicted releases of some isotopes as shown in the tables. However, there is a reasonable overall agreement, considering the differences in assumptions and methods of calculation.

## 2.3.7 CHEMICAL AND BIOCIDAL WASTES

The primary sources of chemical and biocidal wastes are the cooling-tower blowdown and the chemical effluents from the regeneration of the demineralizers used to treat makeup water. The cooling-tower blowdown contains dissolved solids that enter the makeup stream and are concentrated by evaporation during cooling-tower operation. This stream also intermittently contains a small amount of residual chlorine from chlorination of the condenser cooling water (see Section 2.3.5).

## 2.3.8 EFFECTS OF OPERATION OF HEAT-DISSIPATION SYSTEM

The impacts of cooling-tower operation are less for the GCFR than for the reference LWR because less heat is dissipated. At 1,000-MWe operation, the GCFR is predicted to release $5.8 \times 10^9$ Btu/hr, whereas the reference LWR releases $6.7 \times 10^9$ Btu/hr.

## 2.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS

The dose percentages from liquid pathways are presented, by isotope, in Table 2-26. It should be noted that both the adult whole-body and critical-organ doses are almost three orders of magnitude less than those for the reference LWR.

The dose fractions from airborne pathways are given in Tables 2-27 and 2-28: the dose contributions from radioiodines and particulates (to the critical infant and child thyroid doses) are given in Table 2-27; the dose contributions from noble-gas releases are given in Table 2-28. The doses from radioiodines and particulates are mainly from iodine-131, but the total doses are small compared with those for the reference LWR. Similarly, the doses from noble gases are smaller than those for the reference LWR.

### 2.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The quantity of cooling-tower blowdown, and hence the quantity of chemical and biocidal wastes, is somewhat less than for the reference LWR.

### 2.3.11 OCCUPATIONAL EXPOSURE

Occupational exposure for the GCFR concept cannot be readily quantified without actual experience from GCFR operation. There seems to be no reason to suppose that, if the individual occupational doses can be maintained within the limits of 10 CFR 20, and if NRC Regulatory Guide 8.8 is implemented, the total occupational dose for the plant cannot be equal to or less than the estimated 450 man-rem/yr-unit, which is based on the operating experience with LWRs.

In Volume IV of this study (High-Temperature Gas-Cooled Reactors), it has been assessed that the occupational exposure for the HTGR can be less than the 450-man-rem/yr value. The GCFR compares favorably with the HTGR in that the plateout activities in the primary circuit are substantially lower, resulting in a decrease of exposure from maintenance operations.

Occupational exposures would be increased when recycled or spiked fuel is used. Doses from operation and from radioactive-waste handling would not be affected. The contribution of the spiking material to primary-system activity should be negligible compared with that from activation and fission products, and hence there should be no significant effect on exposure incurred during maintenance. Exposures chargeable to refueling would be increased because fresh fuel would arrive in a shielded shipping cask. Additional man-hours and exposures would be incurred in handling the cask, removing the fuel, decontaminating the cask, and so on. The increase in occupational exposure would be a small percentage of the total annual occupational exposure.

Table 2-17. Principal design characteristics of the GCFR

| | |
|---|---|
| Fuel cycle | U-Pu/Th spiked recycle |
| Reactor power level, MWt | 2,700 |
| Electric power output, MWe | 1,000 |
| Heat rate, Btu/kW-hr | 9,217 |
| Heat-dissipation rate, Btu/hr | $5.81 \times 10^9$ |

Table 2-18. Water consumption for the GCFR

| Use | Quantity (gpm) |
|---|---|
| Makeup to cooling-tower system (maximum) | 10,000 |
| Makeup to cooling-tower system (average) | 6,000 |
| Input to laundry, showers, sanitary, and potable water | 3 |
| Input to demineralized-water system | 140 |
| Demineralized-water-system waste | 10 |

Table 2-19. Heat-dissipation-system design data for a natural-draft cooling tower

| | |
|---|---|
| Heat-dissipation rate (maximum--full power), Btu/hr x $10^9$ | 5.81 |
| Evaporation and drift (maximum--full power), gpm | 10,000 |
| Evaporation and drift (annual average), gpm | 6,000 |
| Blowdown (maximum), gpm | 2,600 |
| Blowdown (annual average), gpm | 1,500 |

Table 2-20. Calculated release and venting fractions in GCFR fuel elements[a]

| Isotope | Half-life | Solid-state release fraction (R/B)[c] (%) | Venting Fractions[b] (%) Fuel | Upper blanket | Charcoal rod trap | Upper element |
|---|---|---|---|---|---|---|
| Kr-83m | 1.87 h | 5.53 | 3.75 | 5.44-6[d] | 6.57 | 3.30-6 |
| Kr-85m | 4.40 h | 8.26 | 4.98 | 3.45-3 | 18.7 | 1.77-2 |
| Kr-85 | 10.3 y | 75.7 | 96.8 | 99.6 | 100 | 99.0 |
| Kr-87 | 1.30 h | 4.65 | 3.36 | 1.20-7 | 3.61 | 1.56-8 |
| Kr-88 | 2.80 h | 6.69 | 4.28 | 1.77-4 | 11.5 | 4.08-4 |
| Kr-89 | 3.20 m | 0.97 | 1.87 | 1.85-33 | 0 | 0 |
| Kr-90 | 33.0 s | 0.41 | 1.49 | 0 | 0 | 0 |
| Kr-91 | 10.0 s | 0.22 | 1.16 | 0 | 0 | 0 |
| Xe-131m | 12.0 d | 32.5 | 21.5 | 37.8 | 86.6 | 67.5 |
| Xe-133m | 2.30 d | 19.4 | 11.6 | 6.15 | 64.0 | 21.9 |
| Xe-133 | 5.27 d | 25.7 | 15.6 | 18.4 | 76.7 | 45.1 |
| Xe-135m | 15.3 m | 1.69 | 2.24 | 1.66-20 | 1.41-2 | 0 |
| Xe-135 | 9.10 h | 9.29 | 6.08 | 5.26-2 | 29.1 | 0.37 |
| Xe-137 | 3.90 m | 0.86 | 1.87 | 3.44-33 | 0 | 0 |
| Xe-138 | 17.0 m | 1.78 | 2.28 | 1.36-19 | 2.25-2 | 0 |
| Xe-139 | 41.0 s | 0.36 | 1.51 | 0 | 0 | 0 |
| Xe-140 | 16.0 s | 0.23 | 1.26 | 0 | 0 | 0 |

[a]Calculated for a maximum linear heating rate of 15 kW/ft, a maximum cladding surface temperature of 700°C with hot-spot allowance, and 85 atm of helium pressure, which represents the highest power and temperature fuel rod in the GCFR demonstration plant.

[b]Fraction of gaseous activity entering the region of interest that is vented from the region of interest.

[c]Release-to-birth-rate ratio.

[d]5.44-6 = 5.44 x 10-6.

Table 2-21. Definitions and values of factors for pathways leading to radioactive effluents[a]

| Factor | Definition | Value |
|---|---|---|
| A | Release-to-birth-rate ratio (R/B) | See Table 2-20 |
| B | Venting-to-release-rate ratio (venting fraction) | See Table 2-20 |
| C,D | Defects, suction hole, and vent connection escape rate | $3 \times 10^{-6}$ |
| E | Collection and storage fraction | 1.0 |
| F | HPS[b] 1 process rate fraction, lb/hr | 5,330 (15.6% per hr) |
| H | HPS 1 removal fraction | 1.0 (0.9[illegible] for tritium) |
| J | Plateout removal fraction per cycle, % | 20 for iodine, 0.2 for volatiles |
| K | Design and operating margin | To be established |
| L | PCRV leak rate, $10^{-2}$ lb/hr | 3.9 (1% per year) |
| M | Tritium permeation fraction | $9 \times 10^{-6}$ |
| N | Secondary containment volume, $10^6$ $ft^3$ | 4.03 |
| O | Mixing factor | 1.0 |
| P | Filter decontamination factor and venting rate | 1.0 (28,330 cfm) |
| Q | Stack dilution factor | 6.73 |
| R | Ejector steam fraction of secondary coolant, % per hour | 0.0258 |
| S | Ejector condenser return fraction, % | 84 |
| T | Ejector condenser vented fraction, % | 16 |
| U | Secondary-coolant leakage fraction | $4.54 \times 10^{-3}$ |
| V | Secondary-coolant leakage evaporated fraction, % | 30 |
| W | Secondary-coolant leakage drainage fraction, % | 70 |
| X | Cooling-tower-blowdown dilution factor | 20 |
| Y | Turbine-building ventilation rate, cfm | 94,660 |

[a]See Figure 2-6.
[b]Helium-purification system.

Table 2-22. Definitions and values of factors for pathways from the helium-purification system to radioactive effluents

| Factor | Definition | Value |
|---|---|---|
| A | Fraction of condensables trapped on charcoal bed | 1.00 |
| C | Fractions of krypton, xenon, and decay products on liquid-nitrogen charcoal bed | 1.00 |
| D | Fraction of tritium trapped as HTO on molecular sieve | 0.22 |
| E | Bed-replacement frequency per year | 0.1 |
| G | Desorption fraction | 1.00 |
| H | Regeneration fraction | 1.00 |
| I | Condensed fraction ($H_2O$ + HTO) | 0.99 |
| J | Purge fraction | 1.00 |
| K | Regeneration HPS leakage fraction per year | $5.68 \times 10^{-8}$ |
| L | Drainage fraction ($H_2O$ + HTO) per year | $1.0 \times 10^{-6}$ |
| M | Removal frequency during plant life | 1 |
| N | Tritiated-concrete-casting removal frequency per year | 2 |
| O | Tritiated-vapor leakage fraction | $1 \times 10^{-6}$ |
| P | Gaseous-radwaste drainage fraction | $1 \times 10^{-6}$ |
| Q | Liquid-radwaste venting fraction | 0.01 |
| R | Gas-waste-system flushing return fraction | 0.01 |
| S | Gas-waste-system processing fraction | 1.00 |
| T | Holding-system leakage fraction per year | $6.25 \times 10^{-7}$ |
| U | Process-vent-system leakage fraction | 0.00 |
| V | Krypton and xenon bottling fraction | 0.80 |
| W | Hydrogen and tritium coolant return fraction | 0.02 |
| X | Krypton and xenon fraction returned to reactor coolant | 0.00 |
| Y | Reactor-service-building vent dilution factor, cfm | 190,660 |
| Z | Filter decontamination factor | 1.00 |
| | Gas-recovery-system leakage fraction per year | $1.87 \times 10^{-7}$ |
| | Filter-replacement frequency per year | 1 |

Table 2-23. Distribution of fission-product activity in the GCFR plant

| Isotope | Half-life | Inventory (Ci) Reactor coolant ($10^{-6}$ release) | Plateout (24-year) | Helium-purification system[a] (24-year) | Secondary containment (1%/yr primary coolant leak) | Secondary vent to environment (Ci/yr) |
|---|---|---|---|---|---|---|
| H-3[b] | 12.26 y | 15.47 | -- | 347,330 | 4.07-5[c] | 0.1500 |
| Br-83 | 2.40 h | [illegible] | 1.55 | 0.8647 | 6.00-7 | 2.07-3 |
| Kr-83m | 1.86 h | 1.50 | -- | 1.00 | 2.15-6 | 7.93-3 |
| Br-84 | 31.80 m | 0.7500 | 0.7240 | 0.2063 | 4.93-7 | 1.83-3 |
| Br-85 | 3.00 m | 0.6163 | 0.0560 | 0.0009 | 3.90-7 | 1.44-3 |
| Kr-85m | 4.40 h | 0.8453 | -- | 0.6853 | 1.66-6 | 6.17-3 |
| Kr-85 | 10.70 y | 1.33-4 | -- | 2.55+6 | 3.60-10 | 1.33-6 |
| Br-87 | 55.00 s | 0.4523 | 0.0127 | 2.00-3 | 1.10-8 | 4.06-5 |
| Kr-87 | 1.27 h | 1.25 | -- | 0.0017 | 1.50-6 | 5.53-3 |
| Br-88 | 15.50 s | 0.2697 | 0.0020 | 3.30-4 | 1.67-9 | 6.17-6 |
| Kr-88 | 2.79 h | 1.30 | -- | 0.9077 | 2.21-6 | 8.17-3 |
| Rb-88 | 17.70 m | 0.7999 | 2.14 | 0.9077 | 3.30-7 | 1.22-3 |
| Br-89 | 4.50 s | 0.1327 | 0.0003 | 4.7-5 | 2.70-11 | 1.00-7 |
| Br-89 | 3.18 m | 1.55 | -- | 0.0217 | 1.32-7 | 4.87-4 |
| Br-89 | 15.20 m | 1.01 | 0.6017 | 0.0217 | 3.70-7 | 1.37-3 |
| Sr-89 | 50.80 d | 0.0006 | 1.70 | 0.5540 | 1.62-9 | 6.00-6 |
| Y-89m | 16.00 s | 1.2-7 | 0.0003 | 1.10-4 | 3.33-13 | 1.20-9 |
| Kr-90 | 32.30 s | 0.7313 | -- | 1.87-3 | 1.07-8 | 3.93-5 |
| Rb-90 | 2.70 m | 0.6687 | 0.0663 | 1.87-3 | 4.80-8 | 1.77-4 |
| Sr-90 | 28.90 y | 1.63-6 | 0.0727 | 0.0907 | 4.40-12 | 1.63-8 |
| Y-90 | 2.67 d | 1.20-8 | 0.0727 | 0.0907 | 4.40-12 | 1.63-8 |
| Kr-91 | 8.60 s | 0.3327 | -- | 2.23-4 | 1.33-9 | 4.93-6 |
| Rb-91 | 57.90 s | 0.3220 | 0.0110 | 2.23-4 | 9.67-9 | 3.57-5 |
| Sr-91 | 9.67 h | 0.0150 | 0.2767 | 2.23-4 | 3.43-8 | 1.27-4 |
| Y-91m | 50.50 m | 3.27-3 | 0.1710 | 1.33-4 | 2.13-8 | 7.83-5 |

Note: See footnotes at end of table.

Table 2-23. Distribution of fission-product activity in the GCFR plant (continued)

| | | Inventory (Ci) | | | | |
|---|---|---|---|---|---|---|
| Isotope | Half-life | Reactor coolant ($10^{-6}$ release) | Plateout (24-year) | Helium-purification system[a] (24-year) | Secondary containment (1%/yr primary coolant leak) | Secondary vent to environment (Ci/yr) |
| Y-91 | 58.80 d | 3.13-6 | 0.2877 | 2.23-4 | 8.33-12 | 3.13-8 |
| Kr-92 | 3.00 s | 0.1553 | -- | 3.67-5 | 2.13-10 | 8.00-7 |
| Rb-92 | 4.48 s | 0.1550 | 0.0007 | 3.67-5 | 5.33-9 | 1.97-5 |
| Sr-92 | 2.69 h | 0.0233 | 0.1143 | 3.67-5 | 3.87-8 | 1.43-4 |
| Y-92 | 3.53 h | 0.0027 | 0.1323 | 3.67-5 | 5.00-9 | 1.90-4 |
| Kr-93 | 2.00 s | 0.0787 | -- | 1.23-5 | 7.33-11 | 2.67-7 |
| Rb-93 | 1.00 s | 0.0787 | 1.20-4 | 1.23-5 | 3.67-11 | 1.33-7 |
| Sr-93 | 7.50 m | 0.0623 | 0.0143 | 1.23-5 | 1.27-8 | 4.67-5 |
| Y-93 | 10.20 h | 2.77-3 | 0.0657 | 1.23-5 | 6.33-9 | 2.38-5 |
| Kr-94 | 1.00 s | 0.0290 | -- | 2.27-6 | 1.33-11 | 4.90-8 |
| Rb-94 | 1.00 s | 0.0290 | 2.90-5 | 2.27-6 | 1.33-11 | 4.90-8 |
| Sr-94 | 1.29 m | 0.0277 | 0.0010 | 2.27-6 | 9.67-10 | 3.60-6 |
| Y-94 | 20.30 m | 0.0163 | 0.010 | 2.27-6 | 7.43-9 | 2.76-5 |
| Te-127m | 109.00 d | 2.97-4 | 1.417 | 1.63 | 8.00-10 | 2.97-6 |
| Te-127 | 9.30 h | 0.1197 | 3.43 | 3.39 | 2.73-7 | 1.016-3 |
| Te-129m | 34.10 d | 0.0120 | 17.9 | 20.5 | 3.23-8 | 1.20-4 |
| Te-129 | 1.15 h | 2.66 | 23.4 | 22.8 | 3.10-6 | 1.14-2 |
| I-129 | 1.60+7 y | 1.49-12 | 6.53+2 | 21.2 | 9.00-29 | 1.54-19 |
| Te-131m | 1.25 d | 0.1867 | 10.8 | 10.0 | 4.77-7 | 1.77-3 |
| Te-131 | 25.00 m | 7.27 | 11.26 | 3.70 | 3.97-6 | 1.46-2 |
| I-131 | 8.06 d | 8.33-4 | 49.1 | 33.1 | 2.23-9 | 8.27-6 |
| Xe-131m | 12.00 d | 0.0087 | -- | 14.6 | 1.80-8 | 6.63-5 |
| Te-132 | 3.25 d | 0.4150 | 58.8 | 65.4 | 1.10-6 | 4.07-3 |
| I-132 | 2.28 h | 0.0070 | 61.8 | 66.3 | 1.10-8 | 4.10-5 |
| Te-133m | 53.00 m | 4.03 | 6.48 | 2.30 | 3.80-6 | 1.41-2 |

Note: See footnotes at end of table.

Table 2-23. Distribution of fission-product activity in the GCFR plant (continued)

| Isotope | Half-life | Inventory (Ci): Reactor coolant ($10^{-6}$ release) | Plateout (24-year) | Helium-purification system[a] (24-year) | Secondary containment (1%/yr primary coolant leak) | Secondary vent to environment (Ci/yr) |
|---|---|---|---|---|---|---|
| Te-133 | 12.50 m | 1.66 | 1.47 | 0.4017 | 8.90-8 | 3.29-4 |
| I-133 | 20.80 h | 0.0053 | 27.8 | 15.7 | 1.33-8 | 4.93-5 |
| Xe-133m | 2.26 d | 0.1230 | -- | 151 | 3.23-7 | 1.19-3 |
| Xe-133 | 5.27 d | 2.37 | -- | 6.66+5 | 6.33-6 | 2.34-2 |
| Te-134 | 43.00 m | 5.33 | 6.94 | 2.24 | 4.43-6 | 1.64-2 |
| I-134 | 52.30 m | 0.0563 | 15.9 | 2.96 | 5.33-8 | 1.97-4 |
| Cs-134 | 2.06 y | 6.67-4 | 21.4 | 24.7 | 1.80-9 | 6.67-6 |
| I-135 | 6.70 h | 0.0080 | 9.66 | 6.33 | 1.73-8 | 6.43-5 |
| Xe-135m | 15.70 m | 3.59 | -- | 1.97 | 1.30-6 | 4.83-3 |
| Xe-135 | 9.16 h | 6.39 | -- | 15.9 | 1.40-5 | 5.13-2 |
| I-136 | 1.42 m | 0.4867 | 2.09 | 0.0170 | 1.87-8 | 6.90-5 |
| Cs-136 | 13.00 d | 0.0143 | 8.90 | 9.25 | 3.83-8 | 1.43-4 |
| I-137 | 24.00 s | 0.4106 | 0.4980 | 0.0017 | 4.33-9 | 1.60-5 |
| Xe-137 | 3.90 m | 6.94 | -- | 0.1133 | 7.00-7 | 2.65-3 |
| Cs-137 | 30.20 y | 1.90-4 | 39.7 | 41.7 | 5.00-10 | 1.90-6 |
| Ba-137m | 2.55 m | 1.60-4 | 36.6 | 38.3 | 4.67-10 | 1.73-6 |
| I-138 | 6.00 s | 0.1590 | 0.0483 | 9.67-5 | 4.37-10 | 1.62-6 |
| Xe-138 | 14.20 m | 8.40 | -- | 5490 | 3.33-6 | 1.24-2 |
| Cs-138 | 32.20 m | 11.6 | 14.0 | 2.69 | 7.73-6 | 2.86-2 |
| I-139 | 2.00 s | 0.0487 | 0.0050 | 8.33-6 | 4.43-10 | 1.64-7 |
| Xe-139 | 40.00 s | 1.33 | -- | 0.0040 | 2.47-8 | 9.10-5 |
| Cs-139 | 9.30 m | 7.44 | 2.13 | 0.3620 | 1.77-6 | 6.53-3 |
| Ba-139 | 1.39 h | 1.90 | 6.93 | 0.3620 | 2.37-6 | 8.70-3 |
| Xe-140 | 13.60 s | 0.4373 | -- | 4.67-4 | 3.00-9 | 1.11-5 |
| Cs-140 | 1.06 m | 2.67 | 0.0087 | 0.0120 | 8.00-8 | 2.94-4 |

Note: See footnotes at end of table.

Table 2-23. Distribution of fission-product activity in the GCFR plant (continued)

| | | Inventory (Ci) | | | | |
|---|---|---|---|---|---|---|
| Isotope | Half-life | Reactor coolant ($10^{-6}$ release) | Plateout (24-year) | Helium-purification system[a] (24-year) | Secondary containment (1%/yr primary coolant leak) | Secondary vent to environment (Ci/yr) |
| Ba-140 | 12.80 d | 0.0040 | 2.40 | 0.0120 | 1.00-8 | 3.70-5 |
| La-140 | 1.68 d | 4.67-5 | 2.40 | 0.0120 | 1.00-8 | 3.70-5 |
| Xe-141 | 2.00 s | 0.0647 | -- | 1.00-5 | 6.00-11 | 2.19-7 |
| Cs-141 | 1.00 s | 0.2637 | 2.00-4 | 2.57-5 | 1.20-10 | 4.47-7 |
| Ba-141 | 18.30 m | 0.1606 | 0.0893 | 2.57-5 | 6.67-8 | 2.48-4 |
| La-141 | 3.87 h | 0.0177 | 0.2133 | 2.57-5 | 3.33-8 | 1.23-4 |
| Cs-141 | 32.50 d | 1.067-5 | 0.2287 | 2.57-5 | 2.87-11 | 1.06-7 |
| Cs-142 | 1.00 m | 0.7847 | 0.0167 | 0.0027 | 1.47-8 | 5.43-5 |
| Ba-142 | 10.70 m | 0.5710 | 0.2020 | 0.0027 | 1.57-7 | 5.73-4 |
| La-142 | 1.54 h | 0.1350 | 0.5800 | 0.0027 | 1.63-7 | 6.07-4 |
| Cs-143 | 1.00 s | 0.0600 | 3.03-5 | 4.67-6 | 2.73-11 | 1.016-7 |
| Ba-143 | 30.00 s | 0.0590 | 0.0010 | 4.67-6 | 8.00-10 | 2.98-6 |
| La-143 | 14.00 m | 0.0400 | 0.0177 | 4.67-6 | 2.13-9 | 7.83-6 |
| Ce-143 | 1.37 d | 5.67-4 | 0.0517 | 4.67-6 | 1.47-9 | 5.40-6 |
| Pr-143 | 13.60 d | 8.33-7 | 0.0520 | 4.67-6 | 2.23-12 | 8.16-19 |

[a]The helium-purification system will be cleaned out periodically. Thus, the indicated inventories of long-lived gases after 24 full-power years will be accordingly diminished.

[b]Tritium release to the environment only as gas leakage is quoted here. Leakage to the secondary-coolant system is considered separately.

[c]$4.07\text{-}5 = 4.07 \times 10^{-5}$.

Table 2-24. Comparison of gaseous-effluent releases to the environment from the GCFR, HTGR, and LWR

| Isotope | Release (Ci/yr) | | |
|---|---|---|---|
| | GCFR | HTGR | LWR |
| Krypton-83m | 0.15 | 3.5 | 1.0 |
| Krypton-85m | 0.0062 | 6.0 | 11.0 |
| Krypton-85 | 1.33-6[a] | 3,607.0 | 380.0 |
| Krypton-87 | 0.0055 | 8.0 | 2.0 |
| Krypton-90 | 3.93-5 | 1.5 | -- |
| Xenon-131m | 6.63-5 | -- | 44.0 |
| Xenon-133m | 0.0012 | -- | 80.0 |
| Xenon-133 | 0.0234 | 8.0 | 7,200.0 |
| Xenon-135m | 0.0049 | 3.5 | 1.0 |
| Xenon-135 | 0.0513 | 6.0 | 50.0 |
| Iodine-131 | 8.27-6 | -- | 0.05 |
| Iodine-132 | 4.10-5 | -- | 0.06 |
| Iodine-134 | 0.0002 | 0.0001 | -- |
| Iodine-135 | 6.43-5 | 0.0002 | -- |
| Tritium | 0.15 | 78.0 | 580.0 |

[a]1.33-6 = $1.33 \times 10^{-6}$.

Table 2-25. Comparison of liquid releases to the environment from the GCFR, HTGR, and LWR

| Isotope | Release (Ci/yr) | | |
|---|---|---|---|
| | GCFR | HTGR | LWR |
| Bromine-83 | 0.0021 | -- | 0.0001 |
| Bromine-84 | 0.0018 | 4.00-5[a] | -- |
| Bromine-85 | 0.0014 | 0.0002 | -- |
| Rubidium-88 | 0.0012 | 0.0003 | -- |
| Strontium-89 | 6.00-6 | 0.0001 | 0.0002 |
| Strontium-91 | 0.0001 | -- | 6.00-5 |
| Strontium-90 | 1.63-8 | 0.0008 | -- |
| Strontium-94 | 3.60-6 | 2.00-5 | -- |
| Yttrium-90 | 1.63-8 | 0.0064 | -- |
| Yttrium-91 | 3.13-8 | -- | 0.0001 |
| Yttrium-91m | 7.83-5 | -- | 2.00-5 |
| Tellurium-127m | 2.97-6 | 1.40-4 | 0.0001 |
| Tellurium-127 | 0.0010 | 1.40-4 | 0.0002 |
| Tellurium-129m | 0.0001 | 1.70-4 | 0.0003 |
| Tellurium-129 | 0.0114 | 1.70-4 | 0.0005 |
| Tellurium-131 | 8.27-6 | 2.50-5 | 0.0001 |
| Tellurium-132 | 0.0041 | -- | 0.01 |
| Tellurium-133m | 0.0141 | 3.50-5 | -- |
| Tellurium-133 | 0.0003 | 2.60-5 | -- |
| Tellurium-134 | 0.0164 | 4.30-5 | -- |
| Iodine-131 | 8.27-6 | -- | 0.14 |
| Iodine-132 | 4.10-5 | 1.70-5 | 0.01 |
| Iodine-133 | 4.93-5 | -- | 0.1 |
| Iodine-134 | 0.0002 | 4.30-5 | 7.00-5 |
| Iodine-135 | 6.43-5 | -- | 0.02 |
| Iodine-136 | 6.90-5 | 1.30-4 | -- |
| Cesium-134 | 6.67-6 | 0.015 | 0.01 |
| Cesium-136 | 1.43-4 | -- | 0.005 |
| Cesium-138 | 0.0286 | 3.50-5 | 2.00-5 |
| Cesium-139 | 0.0065 | 2.60-5 | -- |
| Cesium-140 | 0.0003 | 8.60-5 | -- |
| Barium-137m | 1.73-6 | 0.029 | 0.01 |
| Barium-139 | 0.0087 | -- | 4.00-5 |
| Barium-140 | 3.70-5 | -- | 0.0002 |
| Lanthanum-140 | 3.70-5 | -- | 0.0001 |
| Cerium-143 | 5.40-6 | -- | 1.00-5 |
| Praseodymium-143 | 8.16-19 | -- | 2.00-5 |

[a]Note: 4.00-5 = 4.00 x $10^{-5}$.

Table 2-26. Dose contributions from liquid effluents

| Isotope | Contribution to organ dose (%) | |
|---|---|---|
| | Adult whole body | Critical organ |
| Tellurium-129m | <1 | 2 |
| Tellurium-132 | 24 | 96 |
| Cesium-134 | 18 | <1 |
| Cesium-136 | 56 | <1 |
| Others | 2 | 2 |
| Total | 100 | 100 |
| Ratio of GCFR dose to LWR reference dose | 0.002 | 0.003 |

Table 2-27. Contributions to the critical-organ doses of an infant and a child from the airborne releases of radioiodines and particulates

| Isotope | Contribution to organ dose (%) | |
|---|---|---|
| | Infant | Child |
| Iodine-131 | 99 | 95 |
| Iodine-132 | (a) | (a) |
| Iodine-134 | (a) | (a) |
| Iodine-135 | (a) | (a) |
| Tritium | (a) | 5 |
| Ratio of GCFR dose to LWR reference dose | 0.001 | 0.0001 |

[a]Less than 1%.

Table 2-28. Contributions to whole-body and skin doses from airborne releases of noble gases

| Isotope | Contribution to organ dose (%) Whole Body | Skin |
|---|---|---|
| Krypton-83m | (a) | 1 |
| Krypton-85m | 5 | 5 |
| Krypton-85 | 0 | (a) |
| Krypton-87 | 21 | 27 |
| Krypton-88 | (a) | (a) |
| Krypton-89 | (a) | (a) |
| Krypton-90 | (a) | (a) |
| Xenon-131m | 0 | (a) |
| Xenon-133m | (a) | (a) |
| Xenon-133 | 4 | 4 |
| Xenon-135m | 10 | 5 |
| Xenon-135 | 60 | 57 |
| Xenon-137 | (a) | (a) |
| Xenon-138 | (a) | (a) |
| Ratio of GCFR dose to LWR reference dose | 0.00006 | 0.00005 |

[a]Less than 1%.

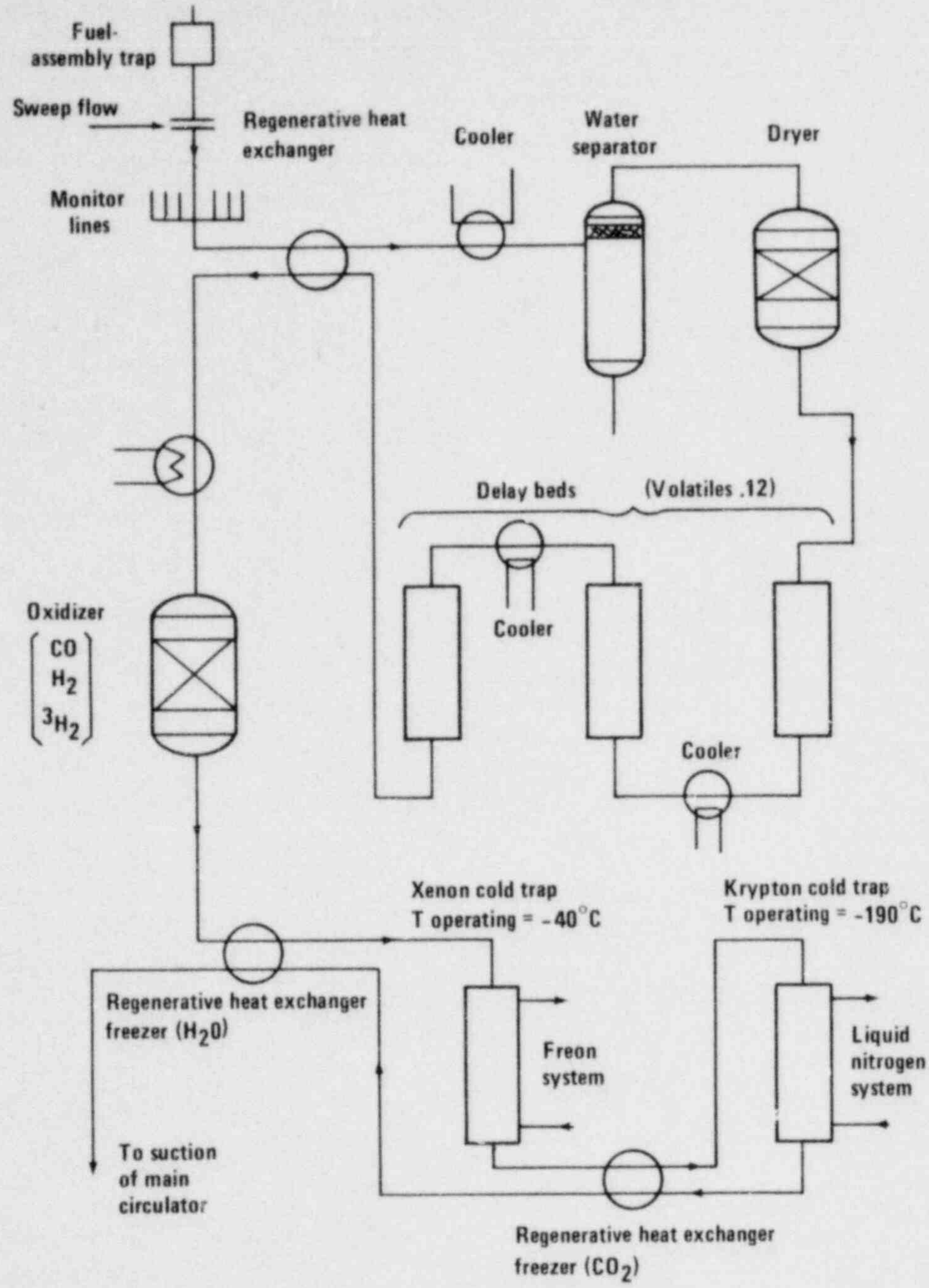



Figure 2-  Fuel-pin venting system.



1072 v

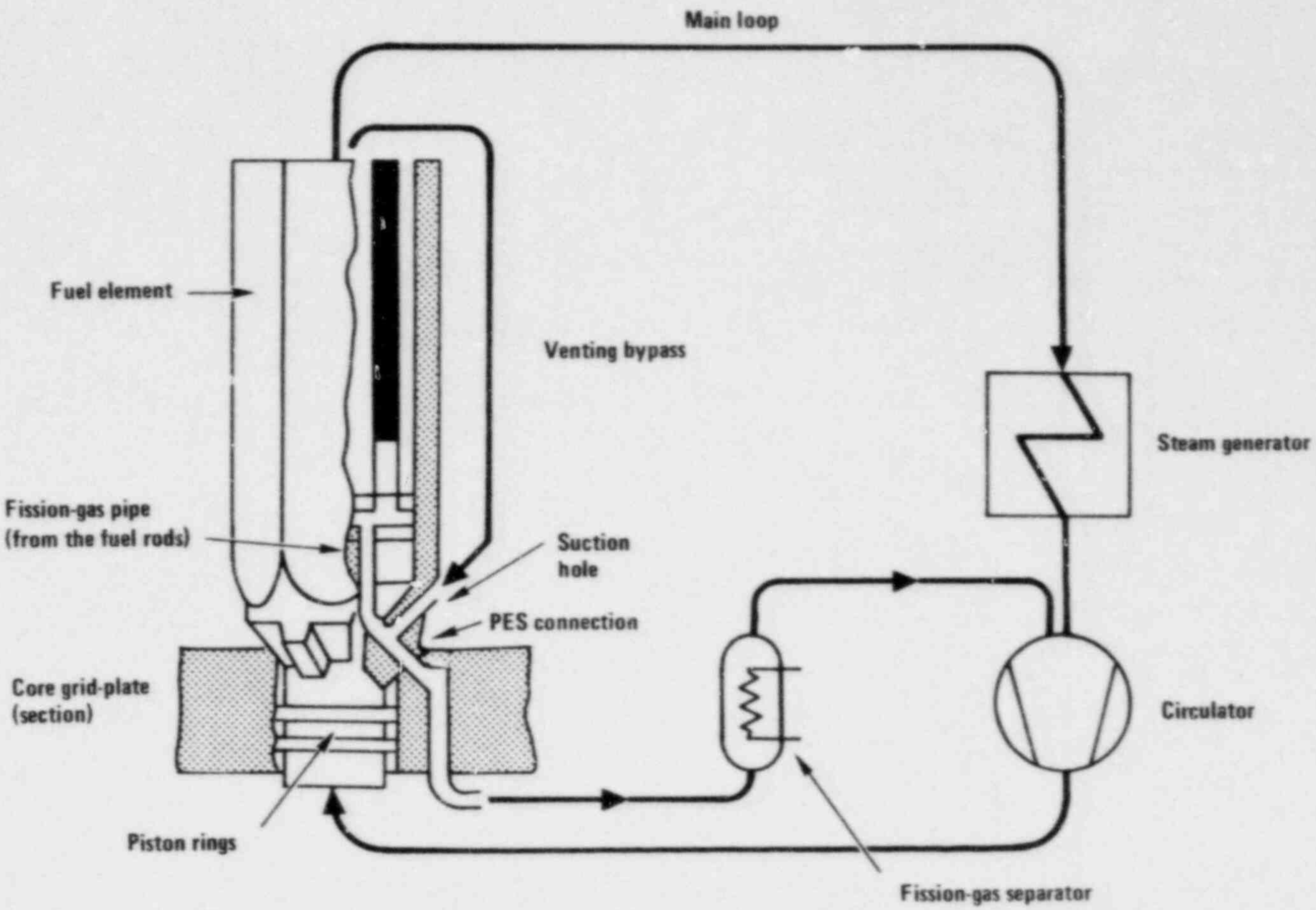
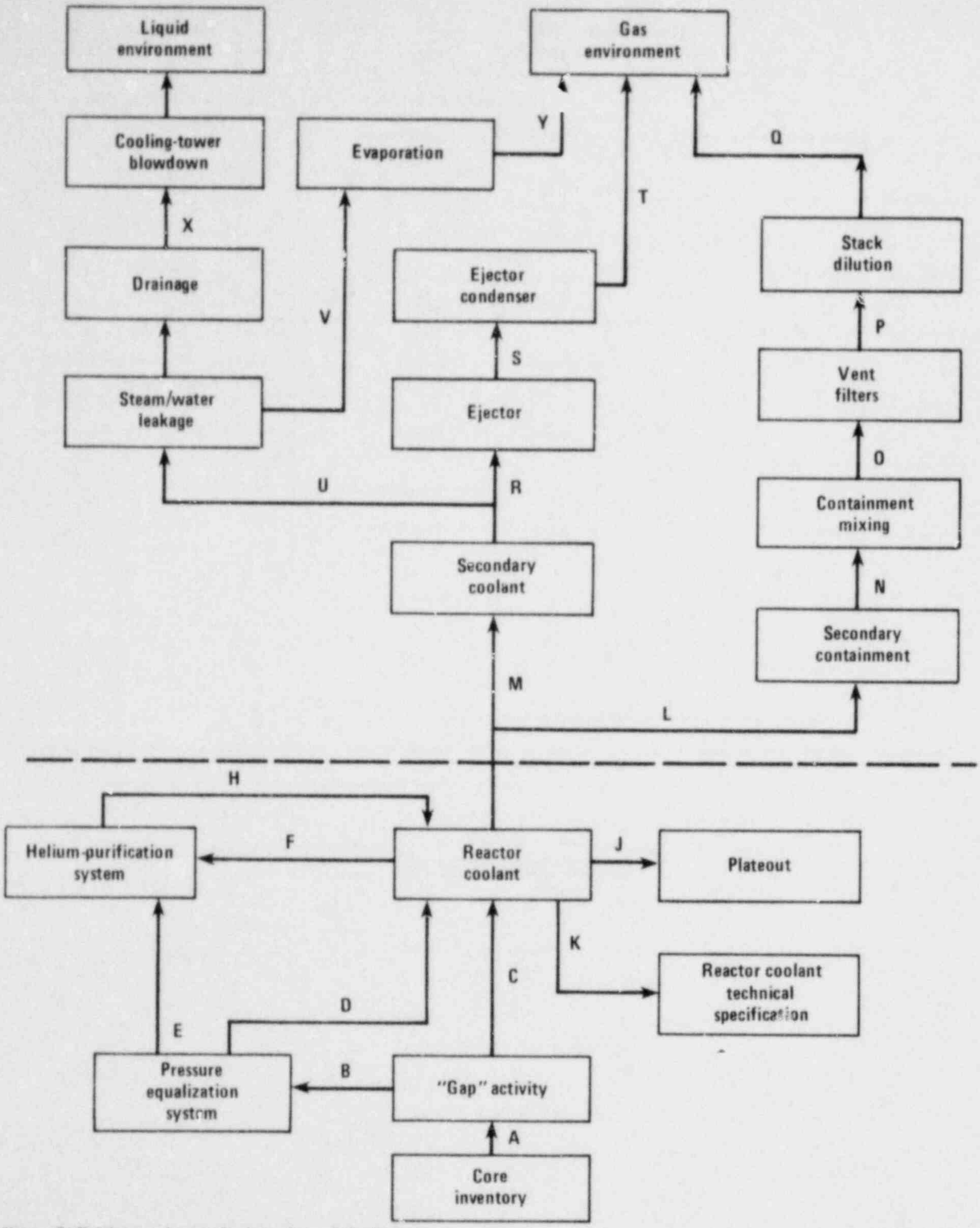



Figure 2-5. Schematic diagram of the GCFR pressure-equalization system (PES).




Note: Definitions and quantitative values of the factors shown along pathways are given in Table 2-21.

Figure 2-6. Pathways during normal plant operation leading to radioactive effluents.

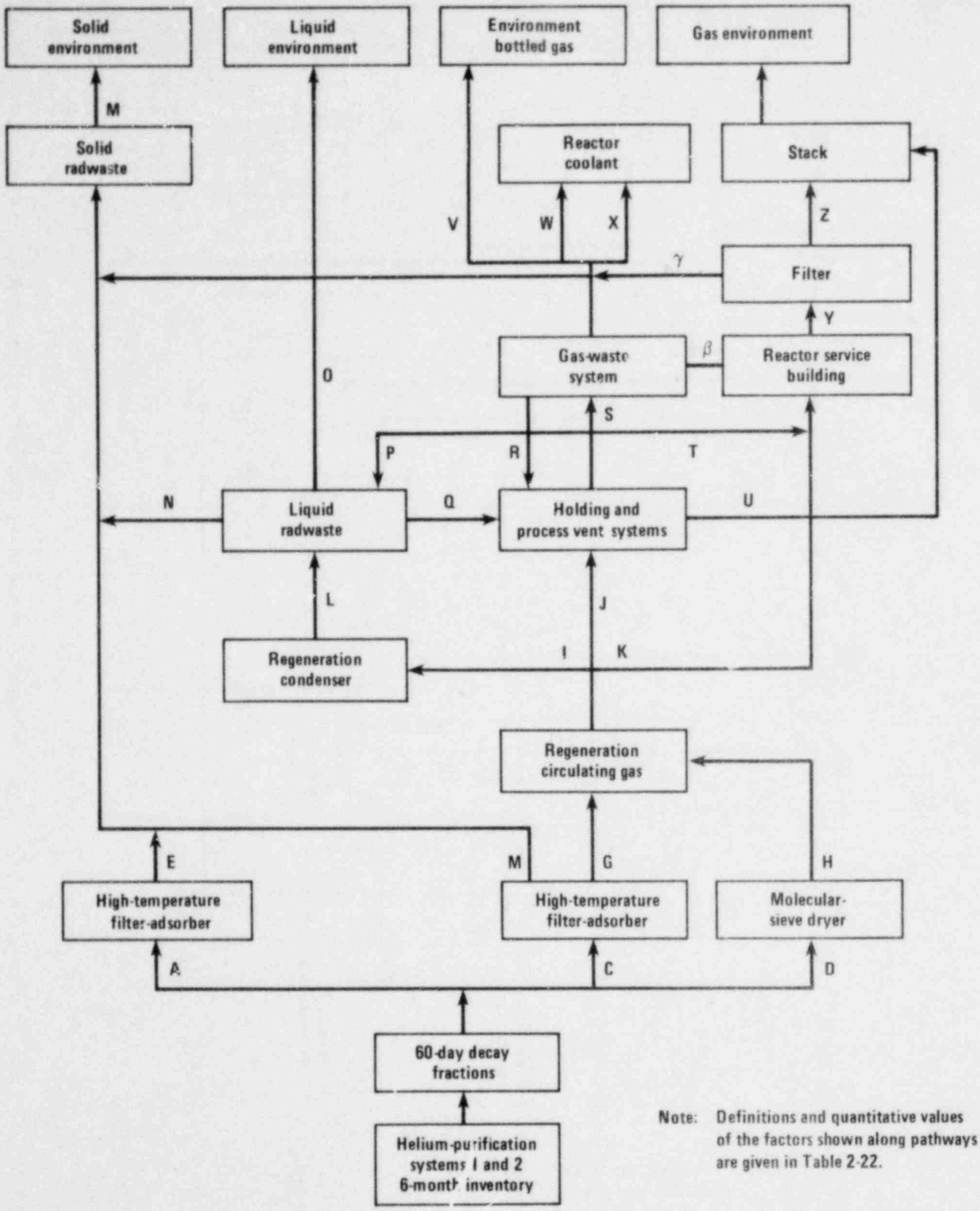



Figure 2-7. Pathways during helium-purification system regeneration and radioactive gas waste system operation leading to radioactive effluents.



1072 v

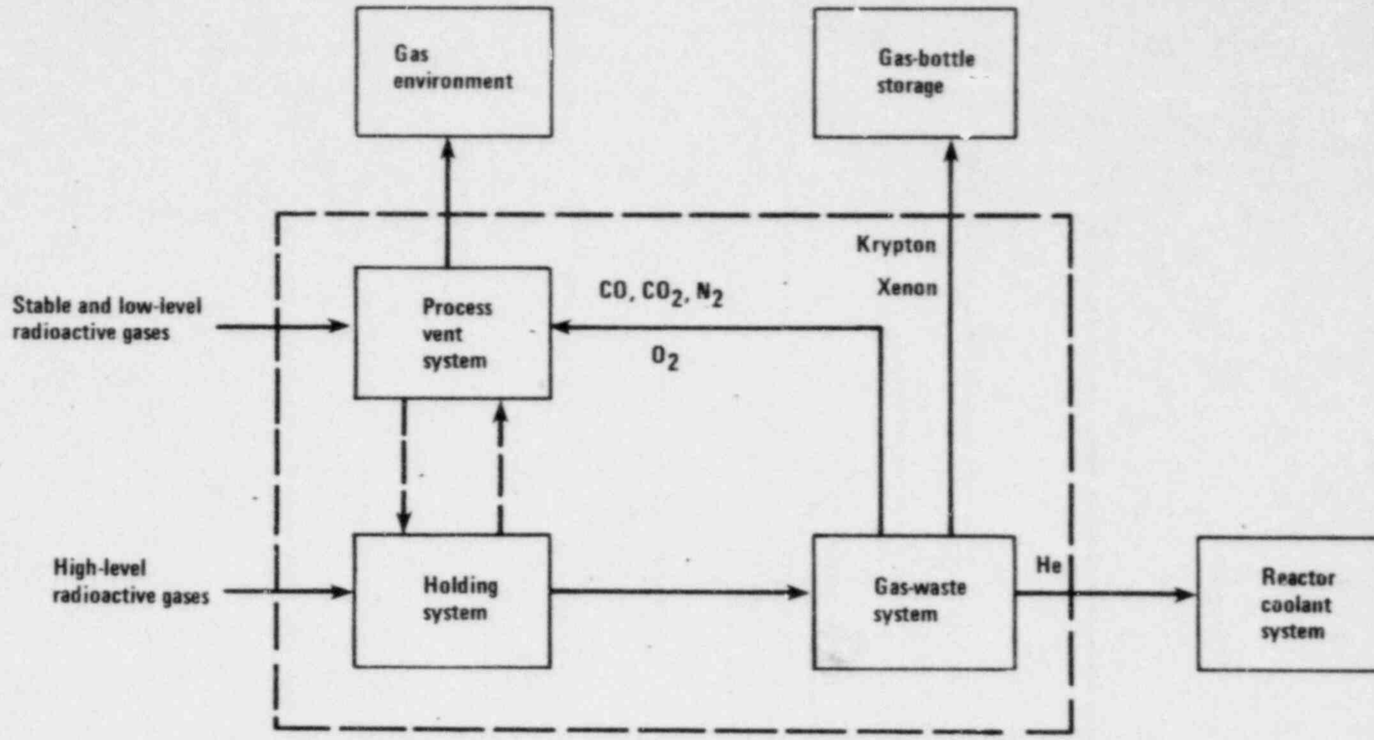



Figure 2-8. Conceptual relationship of the subsystems of the gaseous-radwaste system.

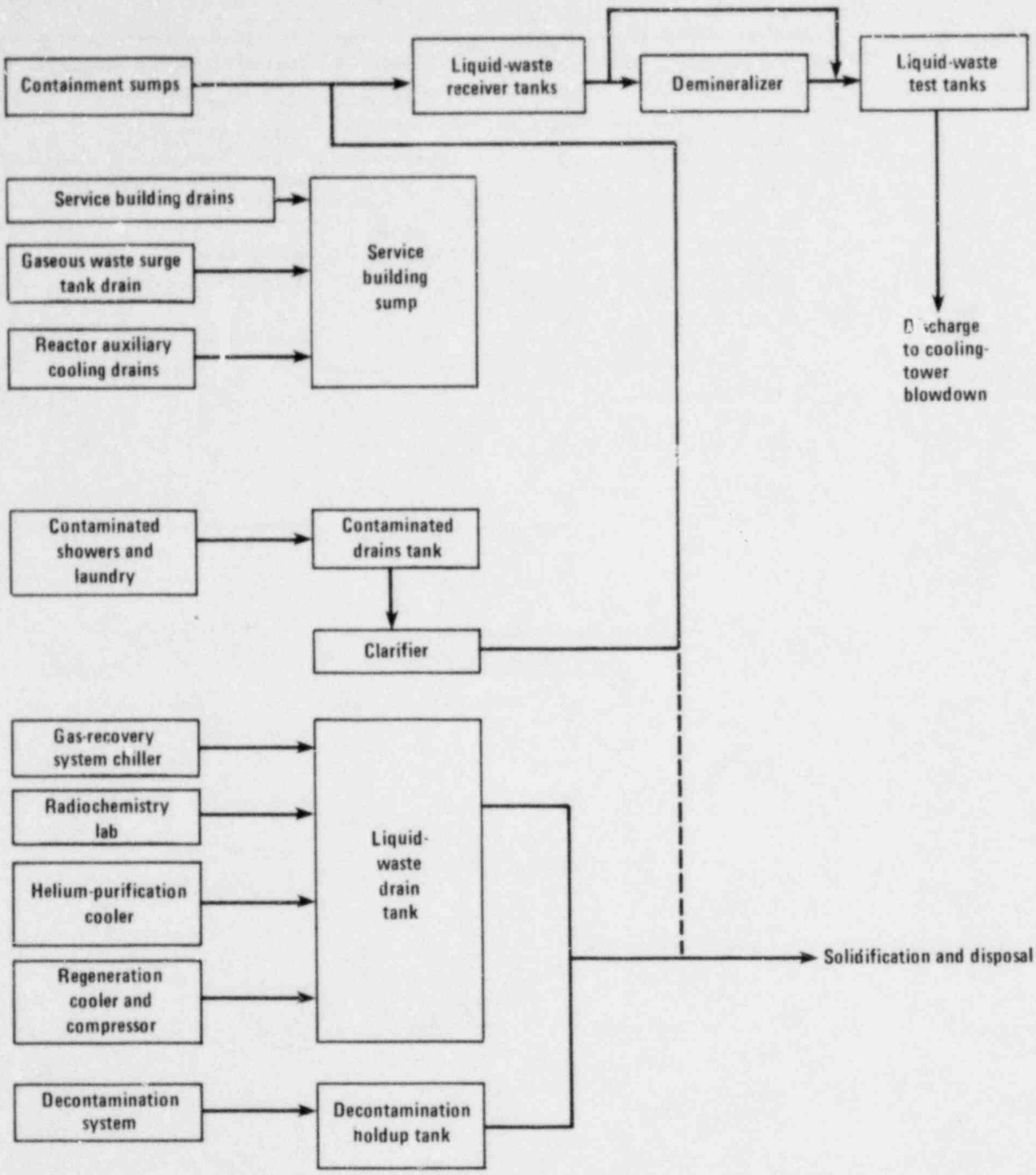



Figure 2-9. Flow chart of the liquid-radwaste system proposed for the Fulton generating station.

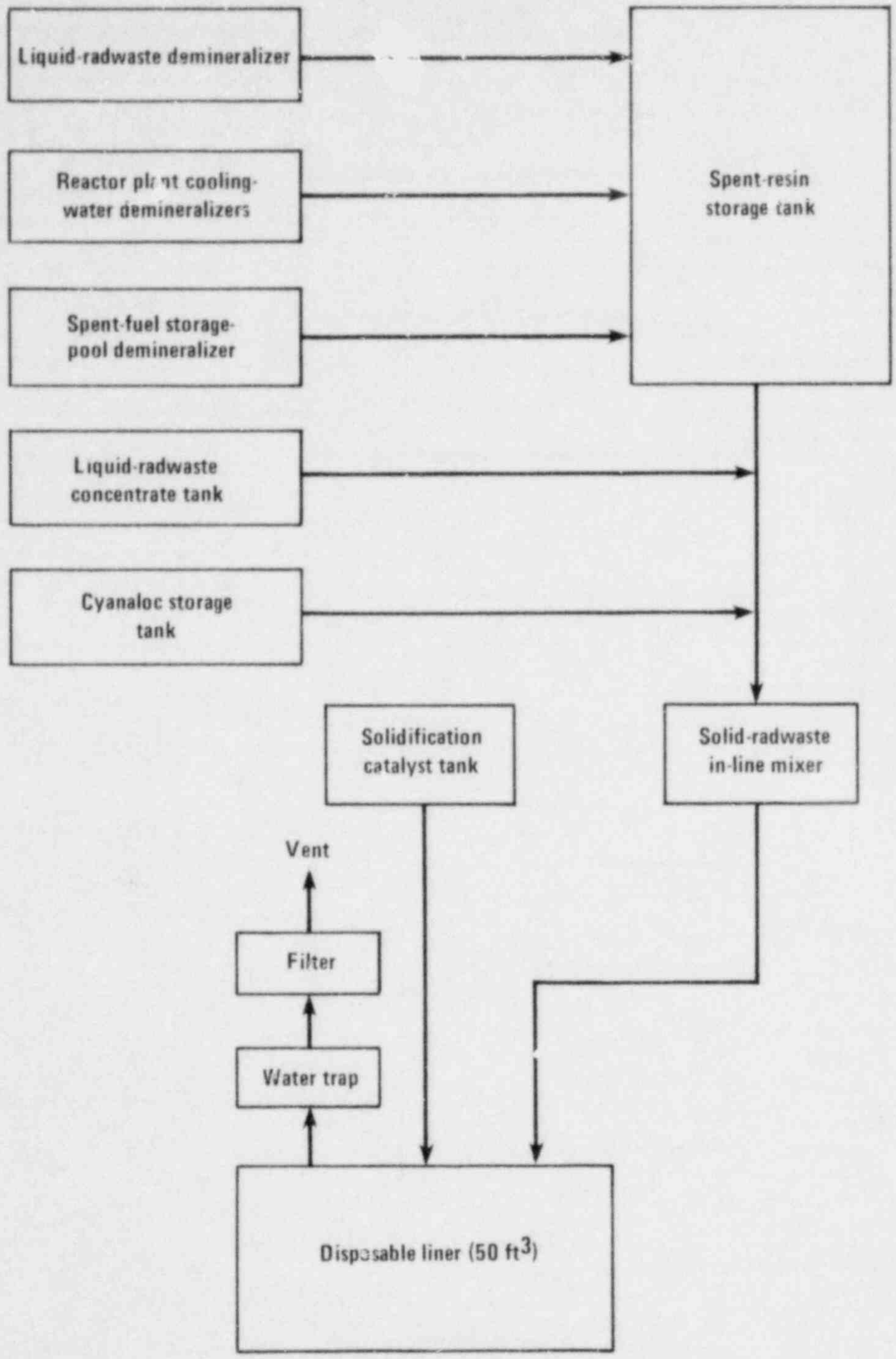



Figure 2-10. Flow chart for the solid-radwaste system.

## 2.4 LICENSING STATUS AND CONSIDERATIONS

The U.S. Atomic Energy Commission (AEC) and the Advisory Committee on Reactor Safeguards (ACRS) have performed a licensability review of a 300-MWe demonstration-size GCFR conceptual design. In August 1974, the AEC published a Preapplication Safety Evaluation Report (Ref. 1) identifying safety issues requiring resolution. These issues are listed in Section 2.2 of this volume. Subsequent to the AEC review, the NRC has provided informal reviews of the design. The conceptual design is currently undergoing some extensive changes in order to promote resolution of the problems that have been identified.

The ACRS issued an interim letter report on the GCFR in November 1974 that left open the decision of "licensing feasibility." The ACRS report indicated the following concerns: core-cooling reliability, common-mode failure potentials for circulators, primary-circuit valve reliability, effects of fuel damage on core-cooling reliability, containment design bases, core-disruptive accidents, reactor-vessel and containment response to core-disruptive accidents, reliability of shutdown systems, reactor-vessel failure mechanisms, in-service inspection of the reactor-vessel liner and the impact of liner loss, engineering data on components, and the maintenance of design flexibility in view of the many outstanding issues. The ACRS also recognizes that the GCFR has certain advantageous safety characteristics. These include the PCRV for the containment of accidents, a small reactivity effect associated with the helium coolant, and enhanced access and maintenance of the system due to the limited radioactivation of the helium coolant.

Several of the issues identified by the ACRS are generic to gas-cooled reactors, and these were successfully resolved in the licensing proceedings for the Fulton and Summit HTGR power stations. With respect to GCFR specific safety issues, recent design changes have enhanced the safety of the GCFR. The reliability and diversity of the core auxiliary cooling system have been upgraded by designing the system so that it can provide core cooling in both forced-circulation and natural circulation operating modes. A second independent closed-loop safety class cooling system, the shutdown cooling system, has been added to further enhance the safety of the design; it is independent from the core auxiliary cooling system.

Since 1974, the General Atomic Company has been developing information in response to the licensing concerns expressed in the "Preapplication" review. Progress on this topic was briefly presented to the ACRS and the NRC staff in July 1977.

Studies of the commercial-size GCFRs show that the overall safety characteristics of the GCFR are expected to improve as plant size increases. Reactivity and kinetics parameters of GCFR cores ranging from 300 to 1,500 MWe have been evaluated, and the results indicate that, as plant size increases, the Doppler coefficient improves by becoming more strongly negative and fuel, cladding, and coolant specific worths decrease significantly, so that whole-core cladding and coolant worths for the 1,500-MWe core are only slightly higher than those for the 300-MWe core.

Larger plants require a lower core enrichment, which reduces the burnup reactivity swing. Larger plants have a larger reactor-coolant inventory, which will result in slower depressurization rates for a given flow-restrictor area. As plant size increases, the number of main cooling loops generally increases, which should improve the reliability of the reactor residual heat-removal system.

The commercial introduction of the GCFR, like that of the LMFBR, requires assurance that the potential consequences of a core-disruptive accident are within specified guidelines. Programs are under way to provide such assurance before the GCFR is introduced commercially.

The GCFR Safety Test Program includes a series of out-of-reactor tests to resolve phenomenological uncertainties followed by in-reactor tests to verify analytical techniques for assessing the consequences of core-disruptive accidents.

In order to ensure compliance of the GCFR safety program with the AEC concerns in the 1974 Safety Evaluation Report, Preapplication Safety Information Document amendments on key issues will be submitted for NRC review in 1979. Pertinent information on these issues is available in References 6 to 21.

## 2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Since the GCFR uses components and systems that are generically similar to those of the steam-cycle HTGR and fuel, fuel-cycle facilities, and physics that are generally similar to those of the LMFBR, the GCFR commercialization study (Ref. 5) concluded that a large-scale proof-of-principle experiment, such as an experiment with a reactor of 100-MWe capacity or less to demonstrate the function of the core, core-cooling methods, and physics under power-reactor conditions, is unnecessary in the GCFR development program. The first plant recommended in the commercialization strategy would be a demonstration plant in the 300- to 800-MWe power range.

Details that are unique to the GCFR are addressed in individual experimental tasks or test loops. Examples of those that appear in the GCFR program are discussed below.

### 2.5.1 PROOF-OF-PRINCIPLE EXPERIMENTS

#### 2.5.1.1 Fuel Irradiation Testing

The extensive irradiation program that is being conducted for the LMFBR is generating much directly useful information in the areas of fuel characterization, fuel cladding, and failure statistics. Additional information is needed in the areas of cladding roughening, fuel-rod venting and pressure equalization, helium-coolant effects, and in-rod fission-product holdup. These areas are currently being addressed in a series of experiments at the Oak Ridge Reactor (ORR), the BR-2 reactor at Mol, Belgium, and the EBR-II reactor.

#### 2.5.1.2 Nuclear Design Data

A major source of nuclear data for core-physics design has been developed by the LMFBR program through basic cross-section research and evaluation programs and through integral experiments at the Argonne National Laboratory critical facilities. Almost all of these data are directly applicable to GCFR development; incremental data needed for confident predictions of GCFR inventories and performance are being addressed through additional critical experiments at the Argonne National Laboratory. Specifically, data that are characteristic of hard-spectrum effects on reactivity and the effects of directional neutron diffusion (streaming) have been derived from an initial set of "benchmark" physics experiments. The performance and economic parameters characteristic of the GCFR are also being evaluated by these and planned future experiments.

#### 2.5.1.3 Heat-Transfer Data

Basic data on heat transfer from the roughened or ribbed fuel cladding have been gathered from experiments at the Agathe test loop at the Swiss Federal Institute for Reactor Research. Heat-transfer and fluid-flow tests have been performed on a 37-rod bundle, and heat-transfer correlations have been established. Additional experiments on blanket-element heat transfer have been performed by the University of California in a simulated blanket array.

#### 2.5.1.4 Natural Convection Verification

Given the timing of the recent selection of an upflow core configuration, the development of a natural convection verification program is still in progress.

However, a natural convection verification plan encompassing both analytical and experimental elements will be prepared and will contain a program leading to the validation of the analysis computer programs used to predict CACS performance in the natural circulation mode.

#### 2.5.1.5 Shielding Design Data

Radiation transport and shielding data applicable to the GCFR are being obtained at the Oak Ridge National Laboratory. The Tower Shield Facility (TSF) reactor is being used as a radiation source for investigating streaming effects and grid-plate shielding adequacy. Additional experiments on radial shielding are planned for the future.

### 2.5.2 EXPERIMENTAL FACILITIES

The dedicated test facilities and test-loop operations that are considered necessary in the GCFR development program are described below.

#### 2.5.2.1 Core Flow Test Loop (CFTL)

A series of out-of-reactor tests will be performed at the Oak Ridge National Laboratory for the following purposes:

1. To demonstrate the ability of the GCFR fuel, blanket, and control assemblies to meet design goals
2. To verify predictions of analytical models that describe design operation and accident behavior.

Electrically heated rod bundles in a dynamic helium loop will be used to obtain thermal and structural data for steady-state, transient, and marginal conditions. Program planning is under way for flow tests to start in 1981.

#### 2.5.2.2 Gas Reactor In-Reactor Safety Test Loop (GRIST-2)

This in-reactor test loop is to be used to aid understanding of the safety characteristics of GCFR fuel and to explore experimentally the consequences of loss-of-flow and reactivity-insertion transients with failure to scram. This loop will be located at the Idaho National Engineering Laboratory at the TREAT-Upgrade reactor. This loop is scheduled to be operational in 1985.

#### 2.5.2.3 Low-Power Safety Experimental Program (LPSE)

The objective of this experimental program is to determine the limiting characteristics of the GCFR core assemblies by investigating the physical phenomena of low-power/low-flow accident conditions up to and including conditions resulting in fuel-cladding melting. Development of electrical resistance heating elements as fuel-rod simulators is under way at the Los Alamos Scientific Laboratory, and initial exploratory tests were conducted in 1978.

#### 2.5.2.4 Circulator Test Facility (CTF)

The purpose of this facility is to perform qualification testing of the GCFR main helium circulator. An electric-motor drive will be used with a full-scale helium circulator in a closed helium-flow loop. Qualification testing is scheduled to begin in 1984.

### 2.5.3 TECHNOLOGY STATUS OF PLANT COMPONENTS

A substantial amount of work has already been completed on materials and methods development that is applicable to both the HTGR and GCFR programs. Additional development work is required for the GCFR nuclear steam supply system because of increased primary-coolant (helium) operating pressures (1,305 versus 740 psia) and different reactor internals and core arrangements. Operating helium temperatures for the GCFR are lower than those for the HTGR (1,000 versus 1,370$^{o}$F). The lower operating temperatures permit a wider latitude in the selection of materials.

The GCFR was originally developed because it offers the potential for superior breeding performance resulting from the hard-neutron spectrum and lower capital costs because there is no liquid-metal handling nor intermediate heat-transport loop. As a fast breeder, it was designed to utilize as much as possible the technologies developed for the LMFBR program, such as fuel and plant components. Since it is a gas-cooled reactor, it draws from the developed technology from related gas-cooled programs, particularly the steam-cycle HTGR, including the Fort St. Vrain reactor. This technology covers the basic technical aspects of the PCRV, helium circulator, steam generator, helium-purification system, CACS, containment systems, and a substantial portion of the plant systems. In addition, relevant information will be liberally exchanged according to the terms of an umbrella agreement and other agreements with a number of European organizations (e.g., the Gesellschaft fuer Kernforschung in the Federal Republic of Germany, the Swiss Federal Institute for Reactor Research). These organizations are actively involved in cooperative technological development programs for the PCRV reactor vessel and internals as well as in fuel, core, and safety-related studies.

Although an attempt is made to carry over as much technology from the LMFBR and the steam-cycle HTGR as possible, there are some features and components that are unique to the GCFR. These may be summarized as follows:

1. Core-support grid-plate internal shielding, material compatibility
2. Fuel-handling concept
3. Control and instrumentation at a high fast flux
4. Main circulator safety features
5. Central cavity closure
6. Thermal-hydraulic behavior of roughened fuel
7. Performance of vented fuel with pressure-equalization system
8. Fuel pellet-cladding interactions
9. Inspection and repair of primary system
10. Natural convection cooling with a single-phase gaseous coolant

These features are the principal components of the program for demonstration-plant development.

An evaluation of required development programs beyond those planned for the demonstration plant should include the following:

1. Prestressed-concrete reactor vessel (PCRV). Additional loops are required for the commercial-plant PCRV. It is necessary to verify the PCRV analytical

calculations with additional scale-model tests, although the basic technology for the demonstration plant is applicable to the commercial plant.

2. Core-support structure. The technology for the demonstration plant can probably be extended to the commercial plant. No additional development is expected.
3. Core locking and reactivity-control mechanism. The same mechanical principles are used for the demonstration and commercial plants, and no additional development is expected.
4. Core shielding. The neutron flux at the interface between blanket and shield is expected to be similar for the demonstration and for the commercial plants.
5. Main helium circulator. The pressures and temperatures for the demonstration and commercial plants are almost the same. However, the flow rates and associated horsepower ratings are significantly different, which may require extensive full-power testing and demonstration, depending on the degree of extrapolation necessary.
6. Steam generator. The development tasks for the demonstration plant are assessed to be adequate for commercial-plant application except for low-flow boiling stability and gas-flow distribution, which will require additional development.
7. Core auxiliary cooling system (CACS). The component technology for the demonstration plant is adequate for the commercial plant. The number of CACS loops in the commercial plant (three or six) has not been chosen yet, and the nature of additional development work depends on the choice of the number of loops. If a three-loop concept is chosen, larger components need to be developed; a revised PCRV design is necessary for a six-loop concept.
8. Other methodologies for physics calculations and safety evaluations are assessed to be the same for the demonstration and commercial plants. Radiological impacts of scaled-up configurations should be evaluated for the commercial plant.

Table 2-29 summarizes the requirements for technological advances in GCFR plant components.

Table 2-30 summarizes the GCFR research, development, and demonstration program. The table is organized according to issues identified by the NRC and the research, development, and demonstration program that is currently under way to resolve the issue with planned completion dates and cost estimates. Figure 2-11 presents the GCFR major milestone schedule.

Table 2-29. Technological advance requirements for the GCFR

| Plant component | Technological advance requirement: None | Modified configuration/ application | Modest improvements in performance or size | Modest improvement in performance or size and modified configuration/ application | Major improvement in performance or size and modified configuration/ application |
|---|---|---|---|---|---|
| Nuclear fuel | | | | X | |
| Reactivity-control systems | X | | | | |
| Reactor vessel | | | X | | |
| Core-support structure | | | X | | |
| Reactor-vessel internals, including shielding, ducting, control-rod guides, and baffles | | X | | | |
| Primary-coolant pumps and auxiliary systems | | | | X | |
| Primary-coolant chemistry/ radiochemistry control | | X | | | |
| Primary-system heat exchangers | | | X | | |
| Reactor instrumentation | | | | X | |
| Emergency core-cooling/safe-shutdown systems | | X | | | |
| Containment, containment-cleanup systems, and effluent-control systems | | X | | | |
| Other accident-mitigating systems, i.e., plant-protection systems | | X | | | X[a] |
| On-site fuel-handling, storage, and shipping equipment | | | | X | |
| Main turbine | | X | | | |
| Other critical components, if any | | | | | |
| Balance-of-plant components | X | | | | |

[a]Postaccident fuel-containment system only, if required.

Table 2-30. GCFR program research, development, and demonstration

| NRC issue | GCFR program to resolve | Planned completion date | Cost[a] |
|---|---|---|---|
| Power density and thermal margins | Core element development program (Ref. 22) | | |
| | Fast-flux fuel-irradiation tests in EBR-II (F-1, F-5) | FY 1985 | Medium |
| | High-pressure helium-loop fuel-irradiation tests in BR-2 reactor (HELM 2, 3, 4) | FY 1985 | Medium |
| | Out-of-reactor heated-rod tests to provide steady-state and transient rod-bundle data in core flow test loop at Oak Ridge National Laboratory (ORNL) (plant now in Title II design) | FY 1986 | Medium |
| | Supplementary program plan for further definition for verification of thermal limits in a DBDA and PSID Amendment 9 | FY 1979 | Small |
| Definition of depressurization accidents | Reexamination of system response to increase in design margins: (a) air ingress, (b) increased DBDA flow areas, (c) increased margins to cladding melting; documentation to the NRC as Amendment 7 to PSID | Completed FY 1976 | Small |
| Definition of core-disruptive accidents (CDAs) | GCFR safety program (Ref. 6) | | |
| | Complete flow blockage in subassembly | | |
| | Loss of flow with shutdown | | |
| | Loss of flow without shutdown | | |
| | Reactivity insertion without shutdown | FY 1987 | Medium |
| | GRIST-II in-reactor tests (at Idaho National Engineering Laboratory TREAT-upgrade reactor) (facility in preliminary design stage) | FY 1986 | Medium |
| | Low-power safety experiments (out-of-reactor tests at LASL) | | |

Note: See footnote at end of table.

Table 2-30. GCFR program research, development, and demonstration (continued)

| NRC issue | GCFR program to resolve | Planned completion date | Cost[a] |
|---|---|---|---|
| | Direct electric heating tests and thermite fuel-motion tests (out-of-reactor tests at Argonne National Laboratory (ANL)) | FY 1981 | Small |
| | PSID Amendment on CDAs | FY 1981 | Small |
| Diversity in reactor-shutdown system | Criteria under development | | |
| Adequacy of core cooling | GCFR safety program (Ref. 6) | | Medium |
| | Reliability analyses by probabilistic techniques | FY 1981 | |
| | Accumulation of engineering-reliability data bank | FY 1981 | |
| | Establishment of shutdown cooling criteria | FY 1979 | |
| | Residual heat removal PSID Amendment to the NRC | FY 1980 | |
| | Risk assessment in support of the PSAR | FY 1983 | |
| | Establish engineering reliability integration program | FY 1980 | Small |
| | Natural convection verification plan | Under development | |
| Containment system design | GCFR safety program (Ref. 6) | FY 1981 | Small |
| | Primary-containment-technology task: | | |
| | Postaccident fuel containment (PAFC) of PCRV | | |
| | Feasibility studies of alternative PAFC methods | | |
| | Response analysis of PCRV | | |
| | Secondary-containment technology task: | FY 1981 | Small |
| | Determine containment pressure/temperature response | | |
| | Analyze aerosol formation and behavior | | |
| | Assess radiological doses | | |

Note: See footnote at end of table.

Table 2-30. GCFR program research, development, and demonstration (continued)

| NRC issue | GCFR program to resolve | Planned completion date | Cost[a] |
|---|---|---|---|
| | Experimental test programs: PAFC experiments at ANL | FY 1981 | Small |
| | PSID Amendment on CDAs | FY 1981 | Small |
| Fuel design | Core element development program (Ref. 22) | | Medium |
| | Fuel-irradiation tests and out-of-reactor test programs: density and thermal margins | Ongoing | |
| | Heat-transfer tests in AGATHE-HEX text loop (Ref. 23) | | |
| | GB-10, 11 irradiation testing of pressure-equalization system | FY 1984 | |
| | Cladding development plan to: | | Small |
| | Assess type 316 stainless steel in prototypic GCFR environment | | |
| | Investigate behavior of advanced alloys for commercial GCFR | | |
| | Alternative fuel materials program: evaluate carbide fuels for vented-fuel applications in GCFR | Ongoing | Small |
| Nuclear design | Critical-experiment program (Ref. 24-25) | | Small |
| | Joint analysis by General Atomic and ANL | FY 1985 | |
| | Analysis of steam-ingress reactivity effects | Completed FY 1977 | |
| | Safety-related physics parameters (PSAR/FSAR) | FY 1978/ FY 1983 | |
| | Analysis of NRC-funded LMFBR slumped/voided criticals | FY 1980 | |

Note: See footnote at end of table.

Table 2-30. GCFR program research, development, and demonstration (continued)

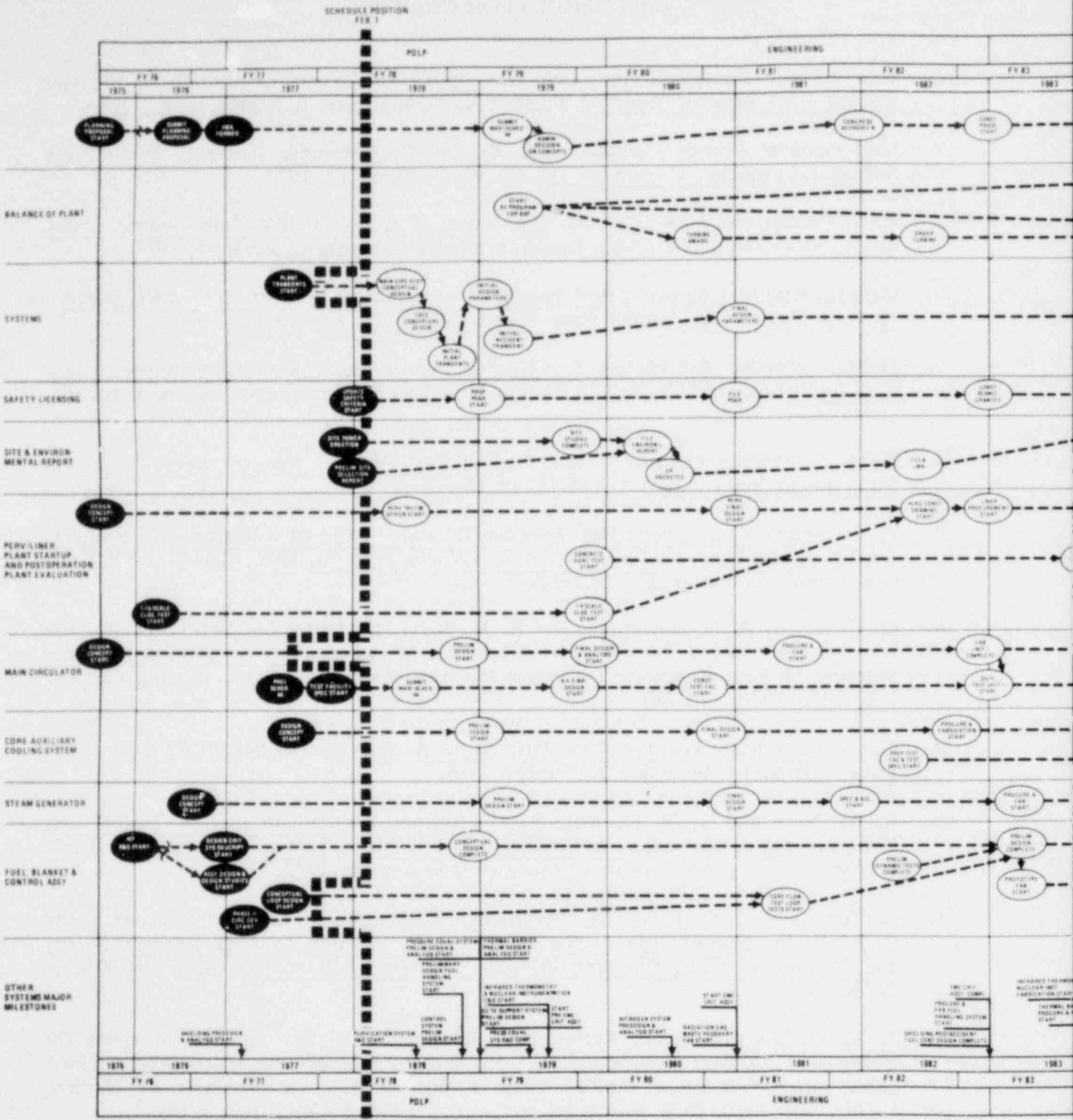
| NRC issue | GCFR program to resolve | Planned completion date | Cost[a] |
|---|---|---|---|
| | Experimental elements of program | Completed | Small |
| | Benchmark critical experiments at ZPR-9 (ANL) | FY 1976 | |
| | Preengineering mockup critical experiment | FY 1981 | |
| | Engineering mockup critical to set enrichments and parameters for FSAR submission | FY 1983 | |
| Prestressed concrete reactor vessel (PCRV) | PCRV development program (Ref. 26) | | Small |
| | PCRV structural overpressure-response test | | |
| | PCRV closure-response tests | | |
| | Closure primary holddown system tests | | |
| | Flow-restrictor tests | | |
| | Thermal barrier tests | | |
| | Experimental elements of program | | Small |
| | Closure model tests at ORNL | In progress | |
| | Pressure tests on scale models | | |
| | Dynamic response: Program to be developed pending definition of core-disruptive accident requirement | | |
| Generic scaleup | | | |
| Nuclear design area | Design improvements: Elimination of rod-ejection accident; allowable rod-worth now comparable to or less than that of the CRBR (commercial plants will have lower worth rods) | Ongoing | Small |
| | Steam-ingress experiments: Critical experiment scoped maximum reactivity worth of steam ingress | Completed FY 1978 | Small |

Note: See footnote at end of table.

Table 2-30. GCFR program research, development, and demonstration (continued)

| NRC issue | GCFR program to resolve | Planned completion date | Cost[a] |
|---|---|---|---|
| Core-disruptive accident analyses | Safety program: Analysis shows that unprotected loss-of-fluid accidents are relatively independent of plant size | Ongoing | Small |
| Primary-system components | Circulator test facility (Ref. 27): full-scale qualification testing of circulator, drive, primary loop-isolation valve (facility in preliminary design stage; construction time 2 years) | FY 1984 | Medium |
| Accident analysis studies | GCFR safety program task elements in place:<br>Probabilistic technology<br>Core-accident technology | | Medium |
| | ANL/GA joint accident studies: NRC presentation July 1977 | Ongoing | Small |
| In-service inspection | Definition of requirements: General Atomic participation in ASME Section XI, Div. 2 Subcommittee to define in-service inspection requirements for gas-cooled reactors | | Small |

[a]Small = less than $10 million.



SCHEDULE POSITION
FEB. 1
PDLP
ENGINEERING
FY 76
FY 77
FY 78
FY 79
FY 80
FY 81
FY 82
FY 83
1975
1976
1977
1978
1979
1980
1981
1982
1983
BALANCE OF PLANT
SYSTEMS
SAFETY LICENSING
SITE & ENVIRONMENTAL REPORT
PCRV/LINER
PLANT STARTUP
AND POSTOPERATION
PLANT EVALUATION
MAIN CIRCULATOR
CORE AUXILIARY
COOLING SYSTEM
STEAM GENERATOR
FUEL BLANKET &
CONTROL ASSY
OTHER
SYSTEMS MAJOR
MILESTONES

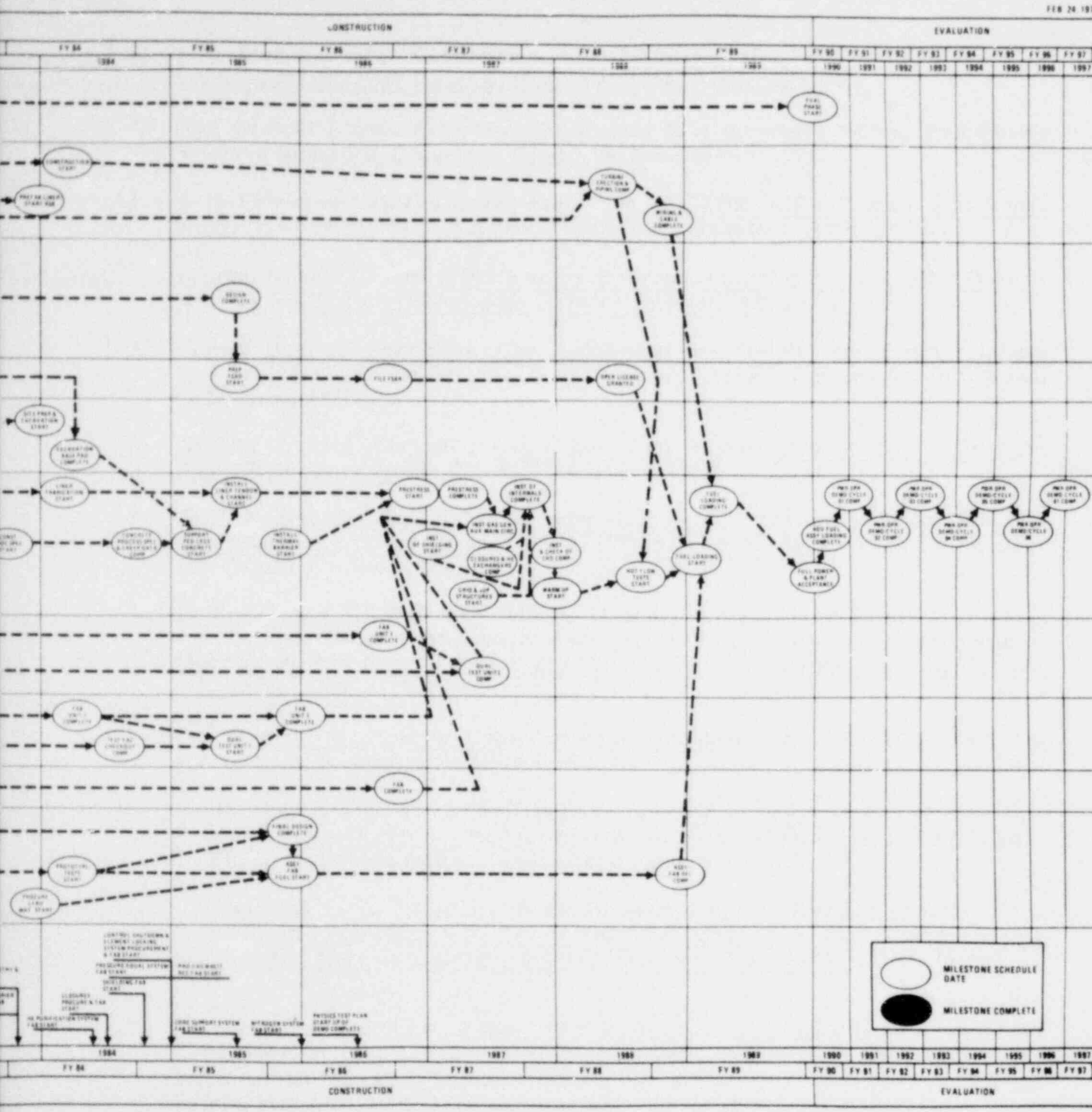



Figure 2-11. GCFR major milestones schedule.

## REFERENCES FOR CHAPTER 2


1. U.S. Atomic Energy Commission, Preapplication Safety Evaluation of the Gas Cooled Fast Breeder Reactor, Project No. 456, August 1, 1974.

2. Gulf General Atomic Company, Gas Cooled Fast Breeder Reactor, Preliminary Safety Information Document, GA-10298, February 15, 1971.

3. J. G. Connor and M. E. Giltrud, Evaluation of GCFR PCRV Containment Capability, NSWC/WOL/TR 77-42, Naval Ordinance Laboratory, April 15, 1977.

4. General Atomic Company, Preliminary Environmental Report, 300 MWe Demonstration Plant, Gas Cooled Fast Breeder Reactor, July 1974.

5. Helium Breeder Associates, Gas-Cooled Fast Breeder Reactor Commercialization Study prepared for the U.S. Energy Research and Development Administration, October 31, 1977.

6. General Atomic Company, Gas Cooled Fast Breeder Reactor Safety Program Plan (Preliminary Issue), GA-A-14749, March 1978.

7. A. P. Kelley, Gas Cooled Fast Breeder Reactor Accident Initiation and Progression Analysis, Progress Report July 1, 1975 to June 30, 1976, GA-A-14079, General Atomic Company, March 1977.

8. D. E. Emon, D. R. Buttemer, and R. H. Sevy, "Evaluation of Safety Test Needs for the Gas Cooled Fast Breeder Reactor," in Proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 5-8, 1976, CONF-761001.

9. Gas Cooled Fast Reactor Core Disruption Accidents, ANL/RAS/GCFR-76-1, Argonne National Laboratory, November 1976.

10. R. H. Sevy and T. Y. C. Wei, Post Transient Phenomena in Gas Cooled Fast Reactor Core Disruption Accidents--Loss of Flow Without Scram, ANL/RAS/GCFR-77-1, Argonne National Laboratory, September 1977.

11. A. Torri and T. C. Tompkins, "Accident Termination by Element Dropout in the GCFR," in Proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 5-8, 1976, CONF-761001, p. 1183.

12. T. E. Collins and A. P. Kelley, "An Application of Probabilistic Analysis for Comparison of Post Accident Fuel Containment Schemes for a 300 MWe GCFR," paper presented at a topical meeting on Probabilistic Analysis of Nuclear Safety, May 8-10, 1978, Los Angeles, Calif., ISBN:0-89448-101-0, p. U.8-1.

13. A. P. Kelley and T. Taniguchi, A Reliability Analysis of the Residual Heat Removal Systems for a 300 MWe GCFR, GA-A14653, General Atomic Company, January 1978.

14. J. T. Reilly, A Review of Methods for the Integration of Reliability and Design Engineering, GA-A14748, General Atomic Company.

15. G. W. Hannaman, GCR Reliability Data Bank Status Report, GA-A14839, General Atomic Company, July 1978.

16. G. W. Hannaman and A. P. Kelley, "Synthesis of Experience Data for Risk Assessment and Design Improvement of Gas-Cooled Reaction," paper presented at a Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, Topical Meeting, May 8-10, 1978, Los Angeles, Calif., ISBN:0-89338-101-0.

17. H. Manzel and C. Wang, Status of Post Accident Fuel Containment Studies for a 300 MWe GCFR Demonstration Plant, GA-A13634, General Atomic Company, 1975.

18. C. W. Wang and A. Torri, "Post Accident Fuel Containment in the Gas Cooled Fast Breeder Reactor, in Proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 5-8, 1976, CONF-761001, p. 2705.

19. M. Dalle et al., "Post Accident Heat Removal for Gas Cooled Fast Breeder Reactors," in Proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, October 5-8, 1976, CONF-761001, p. 2016.

20. C. S. Wang, A. Torri, and C. Yang, The Effect of Chemical Reactions on Post Accident Fuel Containment in GCFRs, GA-A14680, General Atomic Company, December 1977.

21. J. Conner and M. Gilfred, Evaluation of GCFR PCRV Containment Capability, NSWC/WOL/TR-77-42, Naval Surface Weapons Center, April 1977.

22. Core Element Development Program Plan for the GCFR Demonstration Plant, GA-A13312, General Atomic Company, May 1975.

23. Research and Development Studies in Gas-Cooled Reactor Core Thermodynamics at the Swiss Federal Institute for Reactor Research, EIR Report, September 1977.

24. An Experimental Study of the Neutronics of the First Gas-Cooled Fast Reactor Benchmark Assembly, ANL 76-36, Argonne National Laboratory, December 1976 (see also ANL 76-108).

25. Analysis of Full-Core Steam Flooding Experiments for the Phase II GCFR Critical Assembly, GA-A14834, General Atomic Company, May 1978.

26. Development Plan for the GCFR PCRV, GA-A13269, General Atomic Company, February 1975.

27. Conceptual Design for a Helium Circulator Test Facility, GA-A14999, General Atomic Company, June 1978.

# APPENDIX A

## U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

## BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

## NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

---

[a] $\geq$20% U-235 in uranium, $\geq$12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

| Fuel Material Type | Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6) | |
|---|---|---|
| | Mixed[a] | Mechanically attached[b] |
| $PuO_2$, $HEUO_2$ powder or pellets[c] | 1,000/kgHM | 10,000/kgHM |
| $PuO_2$-$UO_2$ and $HEUO_2$-$ThO_2$ powder or pellets[c] | 100/kgHM | 10,000/kgHM |
| LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels) | 10/assembly | 1,000/assembly |
| LMFBR or GCFR fuel assembly (including type b fuels) | 10/assembly | 1,000/assembly |

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

| Fuel material type | Minimum 2-year radiation level, (rem/hr at 1 m) | Process | Minimum initial radiation level, (rem/hr at 1 m) | References |
|---|---|---|---|---|
| $PuO_2$, $HEUO_2$ powder or pellets | 1,000/kgHM | Co-60 addition | 1,300/kgHM | 2, 3, 5, 6 |
| $PuO_2$-$UO_2$ and $HEUO_2$-$ThO_2$ powder or pellets | 100/kgHM | Co-60 addition | 130/kgHM | 2, 3, 5, 6 |
| | | Fission product addition (Ru-106) | 400/kgHM | 2, 3, 5, 6 |
| LWR, LWBR, or HTGR recycle fuel assembly | 10/assembly | Co-60 addition | 13/assembly | 2, 3, 5, 6 |
| | | Fission-product addition (Ru-106) | 40/assembly | 2, 3, 5, 6 |
| | | Pre-irradiation (40 MWd/MT) | 1,000 (30 day)/ assembly | 4 |
| LMFBR or GCFR fuel assembly | 10/assembly | Co-60 addition | 13/assembly | 2, 3, 5, 6 |
| | | Fission-product addition (Ru-106) | 40/assembly | 2, 3, 5, 6 |
| | | Pre-irradiation (40 MWd/MT) | 1,000 (30 day)/ assembly | 4 |

## REFERENCES FOR APPENDIX A


1. Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat, NUREG-0414, U.S. Nuclear Regulatory Commission, May 1978.

2. J. E. Selle, Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence, ORNL TM-6412, Oak Ridge National Laboratory, October 1978.

3. J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen, Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence, ORNL TM-6483, Oak Ridge National Laboratory, October 1978.

4. G. F. Pflasterer and N. A. Deane, Pre-Irradiation Concept Description and Cost Assessment, GEFR 00402.

5. Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use, IRT-378-R, (The Spiking of Special Nuclear Materials as a Safeguards Measure, Vol. 2, BNL File No. 5.9.1 for Vol. 1).

6. E. A. Straker, Material Radiation Criteria and Nonproliferation, SAI-01379-50765, Science Application, Inc., January 8, 1979.

# APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume V, Gas-Cooled Fast-Breeder Reactor

## Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

## RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

## RESPONSES TO SPECIFIC COMMENTS

Question 1

In the event that the upflow core-cooling design is adopted for the gas-cooled fast-breeder reactor (GCFR), it will be necessary for the U.S. Department of Energy (DOE) to re-describe the principal design features of the GCFR and provide an assessment of its safety characteristics in the prevention and mitigation of postulated accidents. This documentation should address all 13 of the safety considerations given on page 2-21 (page 2-20 of this volume) of the preliminary safety and environmental information document (PSEID), provide discussion of any additional safety considerations that the DOE considers necessary and appropriate, and address all of the comments and questions contained herein in the context of the upflow design. We foresee that the upflow design would not adversely affect our general conclusion given in the 1974 Preapplication Safety Evaluation Report and that some of our conditions and reservations regarding the adequacy of the present emergency core-cooling provisions might be positively addressed. Criteria related to the adequacy of thermal margins and fuel damage in the case of natural convection cooling would have to be developed in connection with the assessment of the adequacy of the use of natural convection for emergency core cooling.

Response

a. Acceptable Power-Density Levels and Thermal Margins

The core-power density in a GCFR is similar to that in a liquid-metal fast-breeder reactor (LMFBR). Fast-breeder reactors in general have higher core power densities than thermal reactors; thus, the GCFR and LMFBR both have core-power densities greater than light-water reactor (LWR) and high-temperature gas-cooled reactor (HTGR) cores. The distinguishing feature between the LMFBR and GCFR is coolant heat capacity, with the GCFR helium coolant having the smaller heat capacity. The heat capacity of the coolant residing in a fast-breeder reactor core at any time, if the coolant is stagnant, however, does not significantly delay core heating; thus, the coolant for all fast-breeder reactors must be continuously circulated if adequate core cooling is to be provided. An acceptable margin to a challenge of the core temperature limit is provided by maintaining adequate circulation of the coolant through the reactor core under all circumstances.

The GCFR design includes provisions for providing independent and diverse means of forced circulation cooling with the main cooling system, shutdown cooling system and the core auxiliary cooling system (CACS). In addition, the CACS is designed so that core cooling is provided by natural circulation. Sufficient elevation differences are planned between the reactor core, auxiliary heat exchanger, and ultimate heat sink to ensure circulation of sufficient coolant to cool the core adequately. Preliminary analyses indicate that the natural circulation flow developed is sufficient to maintain fuel-cladding temperatures well below damage limits following reactor scram and main-loop-circulator coastdown.

The GCFR program is currently establishing the core-faulted damage limit that will be presented to the U.S. Nuclear Regulatory Commission (NRC) as PSID Amendment No. 9. While the clad melting temperature had previously been selected as the damage limit, the GCFR program is now proposing a faulted limit of 1,260$^{o}$C (2,300$^{o}$F). It should be noted that the selected limit is not a physical threshold, but rather a selected threshold against which plant components are sized. Additional information on the core faulted damage limit is provided in response g.

b. Definition of Depressurization Accidents

In the Preapplication Safety Evaluation Report for the GCFR issued by the U.S. Atomic Energy Commission (AEC) Directorate of Licensing, three major areas of concern with regard to depressurization accidents were identified. These concerns were (a) the capability of cooling the core under laminar flow conditions, (b) the leak area selected for the design-basis depressurization accident, and (c) the effects of air ingress following the blowdown stage of a depressurization accident. These aspects are analyzed in detail and the results were documented in the GCFR PSID, Amendment 7 in February 1976. A brief summary of the study is given below.

1. Laminar Flow: The flow is expected to be laminar through the core when the core becomes depressurized. Accounting for heat transfer and friction characteristics of laminar flow, the core cooling was extensively studied in conjunction with depressurization-accident analysis. Adequacy of core cooling under laminar-flow conditions was ascertained in this analysis.

2. Leak Area: The subjects of design-basis depressurization-accident leak-area selection based on PCRV closure-failure probabilities and of penetration-flow-restrictor design have been under continuing review for both the GCFR and HTGR. Several discussions were held between General Atomic Company and the U.S. Nuclear Regulatory Commission (NRC) comparing the large HTGR prestressed-concrete reactor vessel (PCRV) penetration and closure design, materials, fabrication, in-service inspection, and failure considerations with those used for LWR pressure vessels. It is posited that PCRV closure failure is sufficiently improbable so that it need not be considered as a design-basis event for the HTGR. The same line of reasoning should apply to the GCFR PCRV penetrations and closures as well. However, in the spirit of what is believed to be the intent of the concern expressed in the safety evaluation report, the system response to depressurization accidents with much larger leak areas than the previously considered 25 square inches have been analyzed. Extensive analyses have been performed with the revised maximum leak area of 75 square inches in combination with the conservative analysis model which allows uncertainties for system parameters. The results of these analyses indicate that adequate core cooling can be achieved during and after the depressurization accidents involving this larger leak area.

3. Air Ingress: At the end of the depressurization blowdown, air from the containment atmosphere disperses into the primary coolant loop through the leak passage by means of thermally induced inhalation, natural convection, and eventually, molecular diffusion, resulting in air ingress into the primary-coolant loops.

The effect of air ingress has been analyzed extensively. Air ingress makes the coolant a better heat-transfer and heat-transport medium for a given volumetric flow. However, with the CACS motor characteristics having both maximum torque and maximum speed limitations, air ingress has a small effect on the overall core-cooling capability with the CACS operating. It has been shown that adequate cooling can be provided within the range between the minimum and maximum air-ingress rates.

4. Sensitivity Studies: Sensitivities of the core cooling under the depressurization accident to several system parameters were explored. Results of the

study suggest the basic conclusion that adequate core cooling can be provided with significant margin. The recent design reconfiguration to an upflow core, a revised lower core pressure drop and an anticipated higher containment back pressure further improve the core cooling under depressurization-accident conditions as compared to the results shown in the PSID, Amendment 7.

c. Definition of Core-Disruptive Accidents

During the GCFR Preapplication Safety Evaluation, General Atomic Company was not in a position to provide information on beyond-design-basis accidents that lead to fuel melting or core disruption. The NRC identified this class of accidents as an important safety consideration where future licensing reviews would require a discussion of relevant aspects of core behavior and consequences.

In response to this NRC position, the DOE established a GCFR safety program to investigate all aspects of core-damage accidents, including the identification of major accident classes, research into the phenomenological aspects of the accident progression, accident consequences, and the probabilistic aspects of core-damage accidents. Until the end of fiscal year 1978, a coordinated effort between General Atomic Company, Argonne National Laboratory, and Los Alamos Scientific Laboratory had investigated the basic characteristics of four accident classes in a cantilevered downflow core without lateral or bottom restraint. The four basic accident classes are defined as:

1. Total flow blockage in a single subassembly
2. Total loss-of-coolant circulation with reactor scram
3. Loss of forced circulation without reactor scram
4. Continued reactivity insertion without reactor scram

During fiscal year 1979, a preliminary investigation of these four beyond-design-basis accidents in a bottom supported upflow core design has been initiated. Only preliminary information on accident progression as it is influenced by the upflow design is available as of this date. The principal characteristics currently identified are as follows:

1. Total Flow Blockage in a Single Subassembly. The principal means for detection is provided by the assembly outlet thermocouple and by delayed neutron precursor monitoring in the outlet plenum. Assembly damage is initiated by cladding melting. Molten cladding is postulated to drain and refreeze in the lower axial blanket region.

Subsequent melting of fuel may lead to an accumulation of molten or refrozen fuel on the steel blockage with thermal attack of the still intact assembly wall. Prevention of a slow propagation of damage to the neighboring subassemblies is expected to be feasible by maintaining a residual coolant flow in the unblocked assemblies. Fuel-coolant interactions, energetic effects, or very rapid damage propagation do not appear as a principal concern in the high-pressure helium coolant environment of the GCFR.

2. Total Loss-of-Coolant Circulation with Reactor Scram. In the former downflow core design, prevention of this accident relied upon continued forced circulation following reactor scram. In the present upflow design, the capability is provided to remove the decay heat following reactor scram entirely by natural circulation of coolants to the ultimate heat sink in the core auxiliary cooling

loops. With this feature and with the added shutdown cooling system, reliability analyses for the total residual heat-removal systems have been performed and are providing confidence that an accident that postulates the loss of all decay-heat-removal capability is indeed a beyond-design-basis accident. Such an accident is nevertheless investigated because a series of common mode failures can be postulated that lead to a total loss of decay heat removal.

Core heatup initiated by a postulated series of common-mode failures leads to cladding melting in the upper half of the core region as supported by early experiments in the Steel Melting and Relocation Test Program at the Los Alamos Scientific Laboratory. Molten cladding drains and refreezes, possibly in several stages, but is expected eventually to form a solid blockage near the core lower axial blanket interface. Subsequent melting or crumbling of declad fuel rods may lead to a critical fuel accumulation on the steel blockage due to the reactivity effects of steel removal and fuel compaction. Very preliminary analysis of this accident phase indicates the possibility of vaporizing a few percent of the core fuel with an associated mechanical energy release potential well below the energy containment capability of the PCRV.

3. Loss of Forced Circulation Without Reactor Scram. The reference scenario for this accident postulates a total loss of main circulator drive power followed by a complete common-mode failure of both the main reactor scram system and of the backup shutdown system. No important differences have been identified in this accident sequence between upflow and a downflow configuration, and the consequences in terms of fuel vaporization and energy release are small and well within energy containment.

4. Continued Reactivity Insertion Without Reactor Scram. Differences between an upflow and a downflow core configuration for this accident class have been examined and no important differences have been identified, principally because at full flow, coolant drag forces on the fuel ejected from the cladding-failure location are much larger than the force of gravity. This class of accidents is not predicted to yield fuel vaporization.

Work is currently in progress to determine environmental consequences from these classes of accidents and to examine alternatives to mitigate the consequences.

d. Diversity in Reactor-Shutdown System

The basic plant safety criteria for which the reactor-shutdown system is being designed are as follows:

1. The control rod system and the backup shutdown system shall be independent, redundant, and diverse.

2. The control rod system trip shall be actuated automatically by the plant-protection system (PPS), and the control rods shall automatically drop into the core in the event of loss of power to the control system.

3. The backup shutdown system trip shall be actuated automatically PPS signals.

To meet these basic safety criteria, the design intent of the reactor trip systems is as follows:

There are two redundant protection systems (nominally referred to as primary and secondary reactor trip) to respond to the same design-basis events. Both protection systems result in different protective actions (by primary reactor trip activation of the control rods and secondary reactor trip activation of the safety rods) to accomplish the same safety functions (emergency negative reactivity insertion).

Conceptually, both protection systems utilize two-out-of-three logic systems. Inputs to each of the protection systems have been tentatively established and will be confirmed by future analysis as well as subsequent instrument sensor selection/design.

It is intended that both reactor trip protection systems independently will meet IEEE-279 (or IEEE-603).

The design of the control-rod-drive system for the GCFR utilizes an adaptation of the control-rod-drive mechanisms designed for the Clinch River LMFBR and the Fast Flux Test Facility (FFTF). These drives utilize the roller-nut principle for translating the driveline screw and releasing the driveline screw for a gravity trip insertion. The roller-nut drive mechanism principle has extensive reactor application history behind it, and is also employed in the nuclear submarine program. The backup shutdown system design has not been selected.

e. Adequacy of Core Cooling

The revised design of the current GCFR demonstration plant differs considerably from the design that formed the bases of the AEC Preapplication Safety Evaluation of the GCFR (issued August 1, 1974). Many of these design changes should influence Commission concerns regarding adequacy of the GCFR core-cooling systems. From a system design point of view, there are several major design changes that contribute to improved core-cooling capability in the GCFR:

1. Upflow Core and Natural Circulation Residual Heat Removal (RHR). Upward core flow direction allows the utilization of natural circulation RHR. It provides diversity to the forced circulation system and inherently passive and long-term RHR with minimum operator or powered action. The GCFR natural circulation concept utilizes an upflow core and the CACS. It is intended that the CACS design incorporate natural circulation capabilities on the helium, water, and air sides as a backup to normal forced circulation capabilities. Using the CACS, core decay heat is transported by the primary coolant helium to a high-pressure water in the core auxiliary heat exchanger (CAHE) which is elevated above the core. Heated water from the CAHE reaches the auxiliary loop cooler (ALC) located above the CAHE by natural circulation in the pressurized water loop. The heat from the ALC is ultimately rejected to the atmosphere by natural draft of air through a chimney. Based on detailed transient analysis, it is concluded that under pressurized conditions, natural circulation can safely cool the core and prevent core meltdown for an indefinite period under a total loss of forced circulation capabilities.

2. Inclusion of Shutdown Cooling System (SCS) The GCFR plant design has now two independent, diverse, and functionally redundant safety-class decay-heat-removal systems in addition to the normal main loop cooling system (MLCS). The

second residual heat removal system SCS shares the main circulator and steam generator with the MLCS. The SCS uses a safety-class pony motor to drive the main circulator with safety-grade power source. The SCS has its own safety-grade feedwater system and long-term ultimate heat sink. Adoption of the SCS as one of two long-term residual-heat-removal systems significantly increases the overall core-cooling reliability of the GCFR.

3. Electrically-Driven Radial Circulators. The change of the main helium circulator drive from steam turbine to electric motor further enhanced the simplicity of plant operation and control.

The use of electrically driven main circulators has significantly increased the core-cooling capability of GCFR. Some of the specific improvements are the following:

(a) Eliminating the concern to have adequate steam supply available and decoupling of the helium circulation from reactor heat source.

(b) Eliminating the requirement for steam generators to operate at low feed-flow, thereby avoiding potential low-flow boiling stability problems in the steam generator.

(c) Providing a higher stall and surge margin by use of a radial flow circulator well-suited for electrical drive, as compared to steam-driven axial flow circulators.

(d) Ensuring a longer circulator coastdown by means of a higher mechanical inertia of the electrical motor, thereby providing a greater time margin for starting up emergency power to safety-class backup-cooling systems following a loss of the main circulator power.

(e) Permitting testing of the main helium circulator system at full power at a test facility and additional preoperational testing in situ prior to plant start-up.

4. Elimination of Superheater. The elimination of the superheater resulted in an overall simplification of plant design, as well as improvements in its operation and control.

f. Containment System Design

The GCFR Program has adopted the Site Suitability Source Team that the NRC established for the Clinch River Breeder Reactor project including 1% of the core plutonium inventory as an aerosol. (See answer to Question 8 for more detail.) On the basis of this source term and the guidelines for dose-consequence calculations for a PSAR application, the GCFR Program has adopted a containment/confinement building with the interspace maintained at subatmospheric pressure by a filtered recirculation system with a filtered stack discharge of excess air. This configuration was adopted on the basis of its adequacy for a wide variety of sites. The basic design parameters such as primary containment leakage, direct bypass leakage, filtration rate, and efficiency remain to be determined but are expected to be within current state-of-the-art containment technology.

The following is a brief description of the reactor containment/confinement building design characteristics.

Containment. The containment building is a steel-lined prestressed-concrete structure similar to a standard LWR design.

Configuration of the building is a vertical cylinder with a hemispherical dome and a flat circular base, of which the lower portion is embedded in the concrete foundation (mat) that supports both the containment and confinement buildings.

Confinement. The confinement building that surrounds the reactor-containment building is a reinforced-concrete structure designed for a slightly negative pressure.

Sections of the building that house the ALCs (air-to-water heat exchangers) will be in compliance with Category I criteria.

g. Fuel Design

The major safety consideration identified by the NRC to be unacceptable in their 1974 review of the GCFR PSID in the fuel design area concerns the faulted-cladding temperature limit. The faulted-cladding temperature limit, referred to as the cladding-damage limit in the PSID, was stated to be 2,500°F. The NRC considered this limit to be unacceptable and predicted that the acceptable damage limit for the GCFR would come more in line with the 2,200°F limit acceptable for light-water reactors using stainless-steel clad. Subsequent to the NRC issuance of the Preapplication Safety Evaluation of the GCFR, the temperature criterion for stainless-steel-clad fuel in light-water reactors was evaluated (Ref. 1). In the evaluation, the acceptable temperature limit for LWR stainless-steel clad was found to be 2,300°F. The primary considerations leading to establishment of the 2,300°F temperature limit were clad ballooning and cladding oxidation. Neither of these, however, is considered to be a problem in the GCFR with the pressure-equalization system and the use of an inert gas.

The faulted-cladding temperature limit for application to the GCFR will be the subject of a PSID Amendment which is scheduled for submission to the NRC in December 1979. The amendment will seek NRC concurrence that the planned calculation models and supporting experimental verification programs are adequate for a GCFR faulted-cladding-temperature limit of 1,260°C (2,300°F).

A faulted-cladding temperature limit of 1,260°C has been tentatively selected for the GCFR on the basis that coolable core geometry is maintained if the cladding does not melt. The selected limit of 1,260°C is a temperature that has a finite margin to the cladding melting temperature (~1,400°C).

Relative to LWRs, the GCFR environment is not as aggressive in terms of oxidizing potential, nor is there any internal rod pressures to cause clad ballooning. Consequently, there is an inherently greater margin associated with the 1,260°C limit when applied to a GCFR than to a LWR.

The acceptability of the 1,260°C limit will not be based upon the LWR precedence but rather on a testing program that models the GCFR conditions. The planned PSID amendment will describe the program planned for verification of 1,260°C for the GCFR faulted-cladding temperature limit.

h. Nuclear Design

The validity of the nuclear analyses methods employed in the design of the GCFR core and for safety studies thereon is unaffected by the switch in coolant flow direction (downflow to upflow) because the internal regions, the subassembly designs for the core rod blankets, need not be altered to accommodate the change. The accompanying redesigns of the core support, assembly restraints, and core cavity will, of course, yield changes to outer region specifications; for example, more rows of radial blanket and/or shield elements can be accommodated and longer axial blankets may be utilized. Also, redesign of the assembly axial shield regions and the cavity shielding are anticipated. These changes are in some cases inversions or dimensional changes of outer regions in past designs.

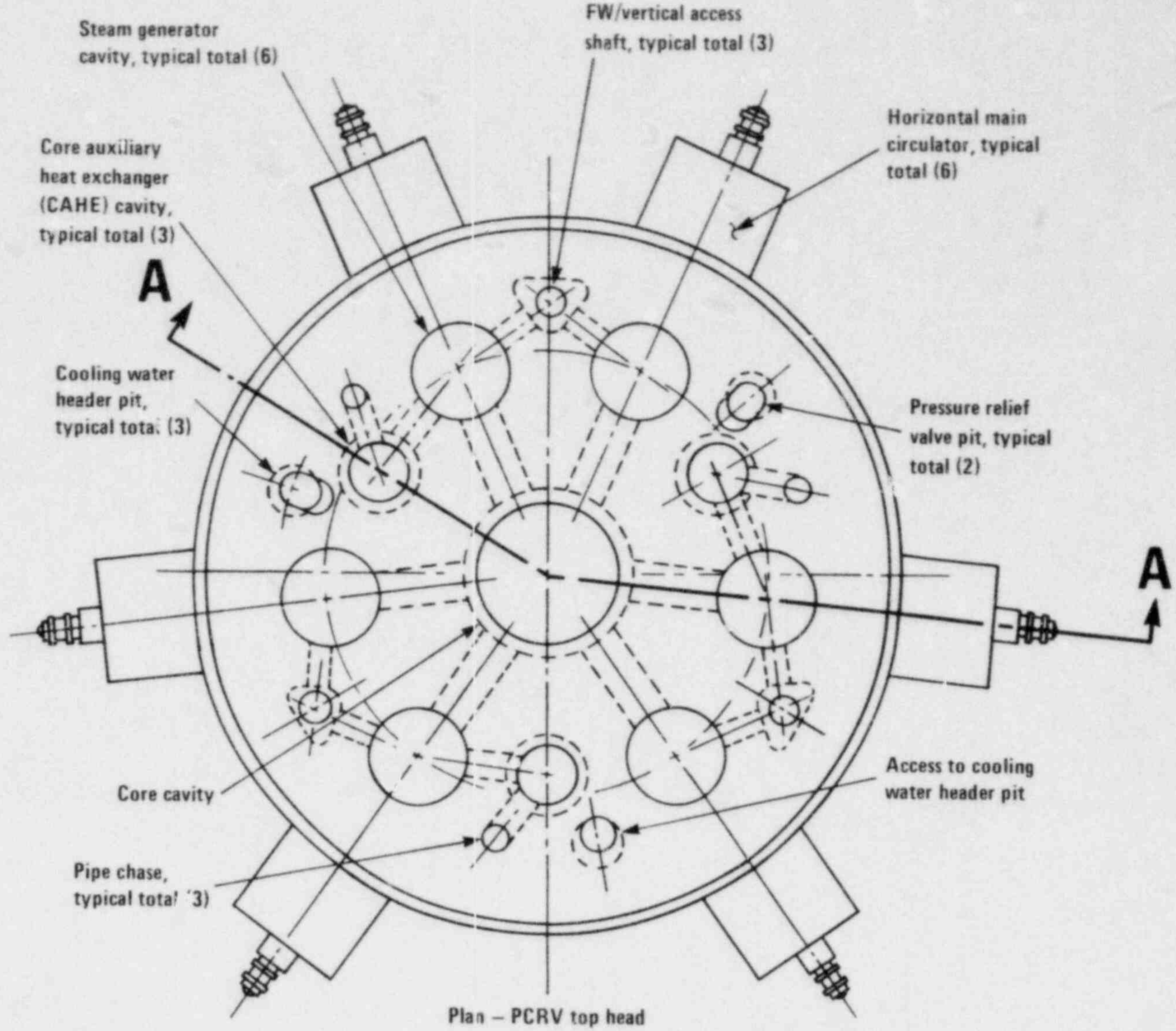
Neutronically, alterations of the outer region designs have insignificant effects on core physics and safety characteristics. For shielding considerations that pertain to neutron and gamma damage to PCRV internals, new analyses will be required; however, there appears no reason to suspect that the validity and uncertainties of the employed shielding methods are changed because of the redesign mandated by the upflow decision.

The safety issue of water ingress into the GCFR core coolant is considered to be resolved on the basis of the whole core "steam" flooding experiments conducted in the GCFR benchmark experiments. The steam-ingress worth was measured in a cold critical assembly with a core volume of only about 1,300 liters, representative of a GCFR power reactor core with a rating of 100 MWe. Although the experiments have a positive worth, the reactivity effect was well calculated by updated physics methods at General Atomic. Other measurements (also calculated) verified that the Doppler effect for uranium (as would be incurred in the heatup to power in a real GCFR core) has a substantial negative reactivity impact on the worth of steam ingress. Thus, the predictions of negative reactivity effect for realistic potential steam ingress into a GCFR core are well substantiated. The larger core designs for the demonstration and commercial plants will mean further negative steam worth, because of the lower leakage and lower accompanying positive component of the steam ingress reactivity effect.

Design changes involved in the reversal of coolant flow, therefore, will have no impact on the core steam ingress worth, and scale-up to larger cores is expected to produce a more negative reactivity coefficient for a given density of assumed steam flooding. The range of potential steam densities reasonably to be expected in accident situations may have to be reassessed in each overall system design, looking at each particular PCRV component with a water or steam loading and evaluating possible leak rates and maximum inventories.

i. Prestressed-Concrete Reactor Vessel

The PCRV configuration for a 1,200-MWe plant and the arrangement of the components in the six primary and three auxiliary coolant loops are shown in Figures B-1 and B-2. The central cavity contains the reactor core and shielding and is sealed by a concrete closure. The coolant flow for a primary loop (Figure B-2) is upward through the core into the core outlet plenum, then through the hot-gas duct to the steam generator cavity. From here the coolant flows downward across the steam generator tube bundles to the main helium circulator that pumps the helium to the core inlet plenum at the bottom of the central cavity. The helium circulators are located in horizontal penetrations in the bottom head of the PCRV.




Note:
Section A-A is shown on Figure 2.

Figure B-1. GCFR six-loop PCRV. Loop arrangement with top exhaust steam generator.

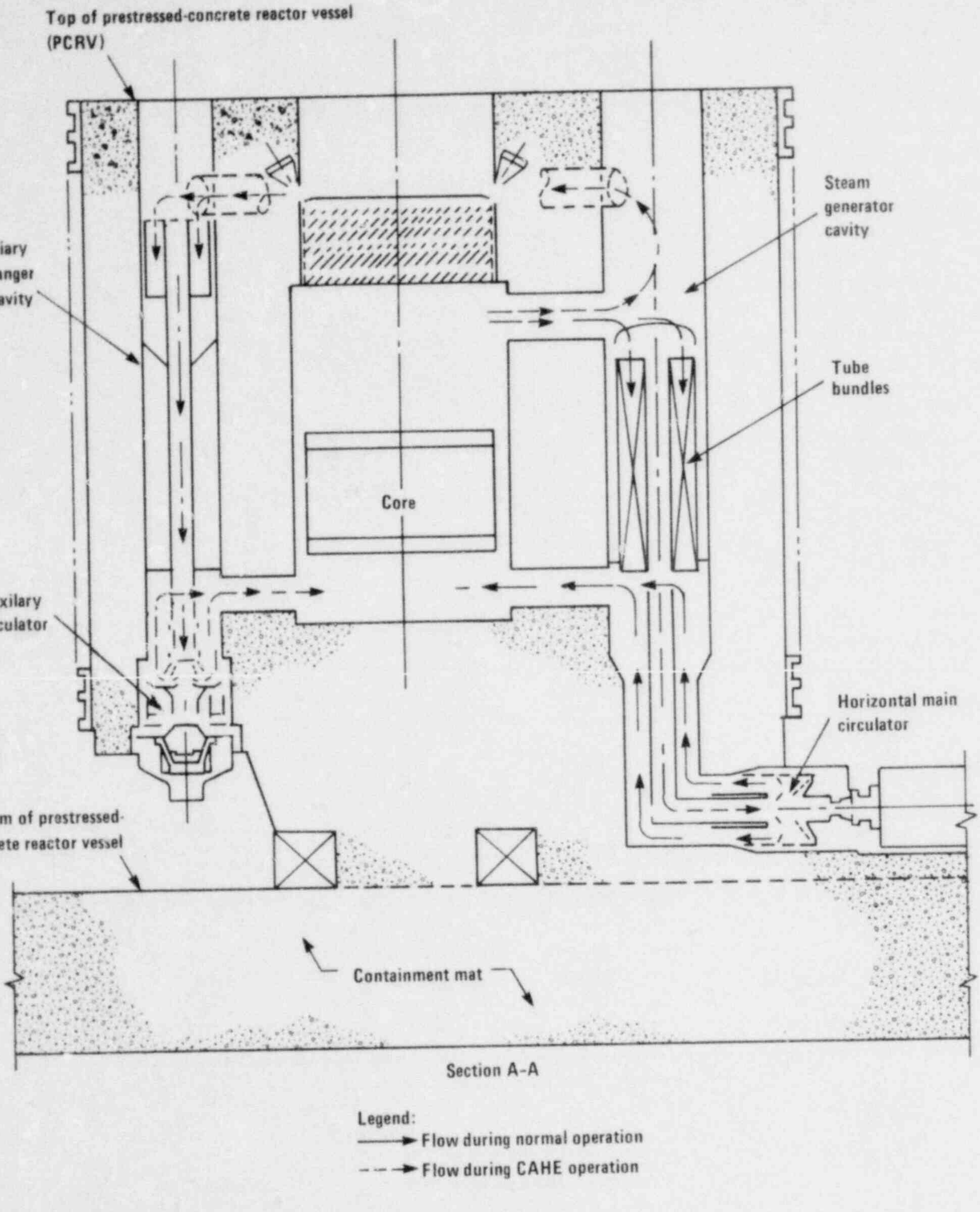



Figure B-2. GCFR six-loop PCRV flow path (top exhaust steam generator).

The PCRV contains a cavity and associated ducts for each of the three auxiliary-cooling-system loops. For auxiliary-cooling-loop operation, the core outlet gas passes into the upper region of three steam generator cavities, then through top head ducts to the upper CAHE cavities. From the CAHEs, the cooled helium continues downward to the auxiliary circulators to be pumped into the plenum below the reactor core.

The PCRV for the 1,200-MWe plant is constructed of high-strength concrete, reinforced with bonded reinforcing steel, and prestressed vertically by linear tendons and radially by circumferential wire winding. Horizontal prestressing in the region of the primary circulator penetrations is provided by linear tendons through the vessel. The previous structural design criteria for the PCRV have been superseded by the adoption of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments" (ACI Standard 359-74).

Changes are required in the configurations of the core cavity and upper steam generator cavity closures. While these closures will continue to be of concrete construction, different closure configurations and different methods of transferring the pressure loads from the closures to the PCRV will be employed. Changes in design criteria have resulted from the need for new closure designs as well as from the acceptance of the ASME Section III, Division 2, code. Specifically, failure of a penetration or closure designed to ASME Section III, Class 1 rules will not be a design-basis event, and flow restrictors will not be provided to limit the leakage flow area which would result from the failure of such a closure or penetration. Additionally, transfer of pressure loads from a closure or penetration to the PCRV concrete will be by redundant elements such as bolts, toggles, or shear anchors as applicable, rather than by primary and secondary systems as described in the PSID.

The criteria which will be used for the design of the PCRV liners, penetrations, and closures are as follows:

1. Cavity, cross duct, and penetration liners backed by structural concrete for load-carrying purposes will comply with the Subsection CB rules of the ASME Section III, Division 2, code.

2. Structural concrete, including its reinforcing and prestressing systems, and liners backed by structural concrete in concrete or composite steel and concrete closures will comply with the Subsection CB rules of the ASME Section III, Division 2, code. Metallic portions of these closures that are unbacked by concrete for pressure-resisting purposes, such as hold-down rings, bolt, toggles, and shear anchors, will comply with the Class 1 rules of the ASME Section III, Division 1, code.

3. Steel closures and penetration liners unbacked by concrete for load-carrying purposes, including shear anchorage elements, will be designed to the rules of the ASME Section III, Division 1, code. Class 1 rules will be followed for penetrations having a free-flow area greater than 10 square inches. Class 2 rules will be followed for small penetrations. Class 1 penetrations and closures will be designed and inspected to the same rules as LWR pressure vessels and use the same or similar materials. All pressure boundary welds will be full penetration, 100% radiographed during fabrication, and subject to volumetric examination during in-service inspection. Redundant shear anchorage elements will be used to transfer the pressure load from the penetration closure to the PCRV concrete. For these reasons, and on the basis of similarities between PCRV penetrations and LWR pressure vessels, failure of

a penetration or closure that is designed to the ASME Class 1 rules will not be a design-basis event. Further discussion of this position is contained in Reference 2.

4. The design-basis depressurization accident will be based on failure of either the largest pipe external to the PCRV which carries primary coolant, such as the pressure-relief pipe, or the largest penetration or closure which is not designed to ASME, Section III, Class 1 rules.

5. The allowable leakage of impure helium from the PCRV will be determined by analyses of the leakage rates that can be achieved and their impact on radiation doses at the exclusion area boundary and within the reactor building. The allowable leakage rate is expected to be greater than 3.65% of the PCRV helium inventory per year (.01% day).

6. Provisions for in-service inspection of penetrations and closures will comply with the rules of the ASME Section XI, Division 2, code, which is currently out for trial use and comment.

j. Generic Scaleup of Nuclear Design Areas and Analyses of Core Disruptive Accidents

The scaleup to a 1,200-MWe commercial plant involves significant changes in core subassembly specifications and, of course, total region volumes. In regard to the nuclear analyses, the principal concerns from the changed subassembly design (fuel-rod diameter and pitch, clad and duct dimensions, etc.) are (1) the different net volume fractions of fuel, steel, and coolant; (2) the changes in resonance shielding in the fuel rods and the regeneration of appropriate cross section sets; (3) the enrichment-zoning requirements for criticality and power flattening; and (4) the control rod loadings for adequate operational and shutdown control. Basically, the physics methods found adequate for the previous designs will be adequate for the larger systems.

However, it is not unexpected that different calculational biases, as for eigenvalues, rod worths, reactivity coefficients, and spatial power distributions, will be required for the scaled-up designs. For example, previous experiences in the fast-reactor community in the analysis of progressively larger LMFBR critical assemblies found systematic variations in the calculated-to-experimental (C/E) ratios for k-values and other parameters; generally, the changes in C/E discrepancies were within the assigned overall calculational uncertainties. As the GCFR concept evolves into the detailed design stages for actual construction, it is anticipated that full-scale mockup critical experiments will be performed to reestablish the pertinent calculational biases and uncertainties and to validate further the utilized physics methods.

The larger GCFR cores, with the lower average fuel enrichment and the lower net neutron leakage fractions, will have significantly different material reactivity coefficients pertaining to safety, mostly in the direction of enhanced safety properties; the uranium Doppler effect will be larger, and the effects of coolant loss and of cladding relocation should be lower. The enrichment and leakage changes also will assure a more negative steam ingress effect over a substantial range of steam density, if the fuel type remains as the current mixed oxide ($PuO_2$-$UO_2$) design.

The effect of neutron streaming through the "voided" fuel lattices in the GCFR remains unchanged for the larger designs. Although larger coolant passages may be adopted for larger pin diameter and pitch, the overall effect on reactivity is expected to stay at about 0.5% in k because of the reduced overall core leakage with the greater core volume. Additional theoretical studies, however, may be useful to validate the current methods used to derive the streaming correction.

The effect of core size on the consequences of core-disruptive accidents has been analyzed by Argonne National Laboratory (Ref. 3) for the downflow core design which indicated a significant degree of insensitivity to plant size for all types of whole-core accidents. Since no significant differences have been identified between an upflow core and a downflow core for accidents without scram, there is every expectation that the same degree of insensitivity will be maintained in the upflow-core design.

k. Primary System Components

1. Steam Generator

While the change from a downflow core to an upflow core had an effect on the ducting of primary coolant, it had a much smaller effect on the steam generator design and no impact on the safety features of the steam generator.

Other important design changes since the 1974 AEC review are as follows:

- The resuperheater section of the steam generator has been eliminated. The cost of the resuperheater could not be justified when analyzed against the enhancement of steam conditions to the turbine. This has resulted in simplifying the steam generator helical bundle design.

- There have been changes to the steam generator tubesheet locations and penetrations. The configuration now allows room for an access penetration into the steam generator PCRV cavity from the bottom. This will enable visual examinations to be made to the support structure of the steam generators. Access is provided at the penetration closure structural welds so ultrasonic volumetric examination of these welds can be made during in-service inspections.

- The main circulator, which was previously steam-driven, will now be powered by an electric motor. As a result of this change, the steam generator will not be required to operate at very low power levels, thereby eliminating the potential stability problems attendant with low-power-level operation.

- System parameters, primary coolant pressure, and temperature at the steam generator inlet have not significantly changed nor have steam-outlet pressure and temperature. The steam generator pressure boundaries and structure are to conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class A, and ASME Code, Section XI. Changes to these requirements between 1974 and this date will have no effect on the steam generator safety features.

2. Circulators and Loop Isolation Valves

Relative to the AEC Safety Evaluation Report (1974), a number of major design changes have occurred in the area of the main circulators. Most of the design changes affecting safety considerations have occurred due to adoption of electric drive for the main helium circulators. Other changes were due to the upflow core selection and general design evolution. The basic changes are as follows:

(a) Due to Upflow Core

The location of the main circulators has been changed from top-mounted vertical to side-mounted horizontal.

The location of the auxiliary circulators has been changed from top-mounted vertical to bottom-mounted vertical.

(b) Due to Design Evolution

The main helium loop isolation valve has been changed from a multiple-louver type valve to a half-circle flapper valve similar in design to that used in the Fort St. Vrain reactor with several additions. The valve in the main loop is designed to be a gravity and back-flow closing type with high-pressure helium jet assist, and supplied with fluidic position monitoring devices. The main loop isolation valve, which is fully open during normal plant operation, can be exercised with jet assist for partial stroking at any time during normal plant operation; therefore, its function can be readily verified. The valve assembly is removable through the circulator flange cavity.

The auxiliary-loop isolation valve is of a similar type, except that it is gravity opened and is back-flow closing. It is also supplied with double-acting, helium-jet assist so the valve can be "kicked" from either a closed or open position. The auxiliary-loop isolation valve has fluidic position monitoring, and its function can be verified during the normal plant operation.

(c) Due to Selection of Electric Drive

The main circulators have changed from 13,000 rpm, axial flow compressor-driven by a high-pressure series steam-turbine variable-speed drive to a 3,000-rpm centrifugal-flow compressor, utilizing synchronous motor variable-speed drive. The main effect of this change with regard to circulator operation is that the main circulators can be driven independently of the nuclear steam supply system operation. During the normal plant operation, however, the circulator speed is controlled according to helium flow requirements.

Other effects are in the following areas:

- Considerably lower stresses in the compressor disk and blades
- Elimination of large thrust loads occurring during the design-basis depressurization accident (DBDA) or downstream pipe-rupture accident

- Elimination of potential overspeed due to downstream pipe rupture and the need for development of a high-pressure/high-temperature rapidly closing steam valve

- Relative ease of conducting the full-power/full-flow prenuclear testing

The requirements for the water-bearing and seal-service system have been substantially simplified. With external electric motor drive, the circulator thrust load is taken by the electric motor oil-lubricated thrust bearing. Because of this, it was possible to develop a water-bearing concept utilizing a self-actuated pump mounted between the circulator journal bearings, thus eliminating the need for a high-pressure external pump. Elimination of the series turbine has done away with the need for a low-pressure separator. The self-actuated water-bearing pump system has eliminated the need for high-pressure accumulators, a high-pressure external water pump, and a backup-bearing water supply. It is estimated that approximately 60 to 70% of the water-bearing auxiliary-system components had been eliminated relative to the series turbine-drive system.

The speed probes and the circulator brake have been relocated from inside the circulator-bearing cartridge to outside of the primary closure and toward the electric motor allowing for easier inspection and maintenance.

The number of static seals on the circulator shaft isolating the circulator-service system from the reactor coolant has been changed from one to two, thus increasing the redundancy of the shutdown seal systems.

The speed control of the steam-driven circulator was achieved in a previous design by modulating the high-pressure/high-temperature throttle and bypass steam valves. In the case of the synchronous electric drive, the circulator speed is controlled by a solid-state variable-frequency controller utilizing thyristors. The variable-frequency control system employs a "self"-commutated frequency converter that is able to restart the circulator immediately following a loop trip.

A pony motor is provided outboard of the main motor. It is coupled directly to the main motor rotor. The pony motor and its power supply are safety-class components, and serve as a backup to the main motor during pressurized cooldown and refueling. (The function of auxiliary circulators to cover all modes of cooldown including the DBDA has not changed.)

### I. Accident-Analysis Studies

Incorporating the major design alternatives developed to date, key transient events under pressurized and depressurized coolant conditions have been analyzed.

The GCFR demonstration plant reference design now has the following three independent systems for forced convection core cooling:

1. the main loop cooling system (MLCS) - non-safety clas
2. the shutdown cooling system (SCS) - safety class
3. the core auxiliary cooling system (CACS) - safety class

In addition, diverse and passive core cooling by natural circulation using the CACS is available.

Pressurized Cooldown. Application to a number of accident events under pressurized conditions indicates that any one of the above RHR systems, can adequately cool the core. It is intended that the licensing criteria on core cooling under total loss of station ac power for 2 hours can be met by the natural circulation mode.

Depressurization Accidents. The DBDA is an extremely low-probability event. The CACS is specifically designed to perform the RHR function under DBDA conditions (details are discussed in PSID Amendment 7). Following slow depressurization accidents, the MLCS with the main circulators can also perform adequate core cooling.

Natural Circulation RHR. Recently, natural circulation RHR in the GCFR has been studied extensively. Various schemes and scenarios of natural circulation in the primary coolant loops and the secondary coolant loops have been found to be feasible. One of the feasible systems uses the CACS, as discussed under response to this question.

m. In-Service Inspection

In-service inspection is discussed under Question 6.

Question 2:

It will be necessary to establish explicit licensing criteria for the GCFR as a portion of its construction permit review. The objective of these criteria will be to assure that at least a comparable level of safety is achieved in comparison with other commercial reactors. Means for establishing such criteria, in descending order of desirability, are (a) direct adoption of existing criteria (e.g., IEEE criteria and many Regulatory Guides), (b) adoption of existing criteria where necessary discrepancies can be justified, and (c) the development of new criteria to meet the unique aspects of the design. Preliminary criteria development during the preapplication review phase is desirable to guide the conceptual and preliminary design activities and to anticipate areas that will need to receive close attention during the construction permit review stage. We appreciate that General Atomic has been active in this area in the recent past.

One aspect that has not yet been explored is the contribution to criteria development available from the several European governments cooperating in the development of the GCFR. We are generally aware of some of the differences in criteria between the Federal Republic of Germany and the United States, but have not considered how such differences might be manifested in either the design of the GCFR or in its licensing criteria. We are interested in a discussion of the potential effects of these differences with particular regard to in-service inspection and testing, seismic design, and requirements for redundancy and diversity of engineered safety features. Please discuss how you expect these criteria differences to influence the design and licensing criteria of the GCFR in the United States. If there are other criteria differences you believe are significantly different, please discuss these also (e.g., design-basis accidents, containment-system design bases, and primary-system integrity).

Response

Licensing criteria for the GCFRs have not been established by the German licensing authorities. Our German counterparts have used the SNR-300 licensing requirements and their own interpretation of the German licensing situation for fast reactors to derive guidance for the principal design features they expect to be required in Germany.

Under the United States/German Umbrella Agreement for Gas-Cooled Reactors, there is a joint task defined to identify differences in criteria, codes and standards in the United States and in Germany, and to interpret the differences as they might affect the GCFR design. This activity has not as yet been funded in Germany.

To initiate the establishment of GCFR licensing criteria, a revision to Amendment 8 to the GCFR PSID on General Design Criteria was submitted to the NRC as part of the planned preapplication review in July 1979. The objective of this document is to obtain NRC concurrence with recommended changes in the General Design Criteria, which are worded specifically for the GCFR.

Question 3

At the meeting held on February 26, 1979, the main body of the information provided was for the 300-MWe design although the conceptual characteristics of the 1,200-MWe plant were outlined. Please resolve from the standpoint of the desired approach to the DOE safety review of the 1,200-MWe plant which course you will follow to satisfy additional information on (a) establish the scale-up feasibility of the 300-MWe design to the 1,200-MWe size; (b) provide information in greater depth for the 1,200-MWe size with reference to features of the 300-MWe plant that demonstrate feasibility of the 1,200-MWe design; or (c) identify some alternative plan that will satisfy the NASAP objectives.

Response

An on-going active task has been established in the GCFR Program to define a 1,200-MWe plant; therefore, "information in greater depth," specifically for the 1,200-MWe plant, will be developed in the near future.

In addition, it is currently being considered that the demonstration plant use components that are as prototypical as possible of the 1,200-MWe plant. For a six-loop large plant of 200 MWe per loop, a two-loop, 400-MWe demonstration plant is implied. As a minimum, the main circulator and its drive, the steam generator, and operating conditions are to be prototypical, but the auxiliary circulator, auxiliary heat exchanger, refueling equipment, and, to the extent practical, all equipment also may be prototypical. Therefore, "reference to features of the (demonstration) plant" will be provided to further establish the safety and feasibility of the 1,200-MWe plant.

Question 4

What, in addition to that provided in the PSEID, can be said about occupational doses for the GCFR relative to LWRs and the other NASAP reactor designs? Consider normal operation, refueling, in-service inspection, and decommissioning plans.

Response

Occupational doses for the GCFR relative to LWRs and other NASAP reactor designs have not been explicitly addressed in the GCFR Program. It is expected, however, that the clean primary helium coolant system, together with the vented fuel rod system, will result in total occupational exposures far less than the 400-1,000 man-rems actually experienced at LWR plants (Ref. 4).

Operating experience at the Fort St. Vrain HTGR plant has, for example, shown no activation of the helium circulators with the result that hands-on maintenance can be performed routinely when the plant is shut down. Similarly, entry into the reactor building during plant operation does not require protection from activated coolant or crud deposition. There is every reason to believe this will be true for the GCFR helium systems and turbomachinery as well.

Exposures during refueling operations would be expected to be roughly comparable to the same LMFBR operation.

Question 5

What equilibrium fraction of noble gases, iodines, and other volatile fission products is resident in the GCFR fuel rods in comparison to nonvented fuel rods? Also, how is the comparative decay heat level of the reactor altered by continuous venting of the volatile fission products? Do accident studies consider this lower inventory of fission products in the GCFR core?

Response

The equilibrium fractions of radioactive noble gases, iodine, and other volatile fission products resident in GCFR fuel rods are nearly the same as those in nonvented fuel rods. The stable noble gases and long-lived Kr-85 (10.7y) reach equilibrium fractions by venting at rates equal to the fuel-release rates. Tritium (12.3y) is released from the solid fuel (release rate/birth rate (R/B)≥100%) and reaches equilibrium residual fraction (approaching 1 to 2%) by venting and by permeation of the cladding. The residual fractions of the other radioactive noble gases remaining in the rods approaches 100% except for Xe-131m (12d) and Xe-133 (5.3d). Venting rate birth rates (V/B) for Xe-131m and Xe-133 were measured in the HELM 3 test up to burnups of 28 MWd/kg (Ref. 5) at the GCFR linear heating rate of 45 kW/m maximum and 40 kW/m average over 12 rods, 700° C maximum cladding mid-wall temperature, and 5.8 MPa He pressure. The V/Bs were found to be maxima of 10 and 2%, respectively. Thus, the residual fuel-rod fractions (1-V/B), even for these intermediate-lived noble gases, are greater than 90%. There has been no measurable venting of iodine. All other volatiles, except those resulting from venting of precursor gas (such as Cs), are contained in the fuel subassemblies.

The decay-heat level of the reactor is negligibly altered by continuous venting. Kr-85 and tritium fission products generate negligible amounts of decay heat because of their slow decay rate and low energy radiations. The 2- to 12-day Xe radioisotopes remove only a negligible amount of decay heat by venting (~2 kW(t)) total. The volatile fission products are not vented at measurable rates.

Accident studies done to date at General Atomic conservatively assume that all of the decay heat is generated in the GCFR core. Reduction of the inventory of fission products in the GCFR core is considered negligible for accident studies and is ignored (even in depressurization accidents).

Question 6

What are the specific criteria and requirements for in-service inspection and how will these be integrated into the preliminary design? What role will the ASME Section XI committee play in these decisions?

Response

Proposed ASME Section XI, Division 2, "Rules for Inspection and Testing of Components of Gas-Cooled Plants," applicable to HTGRs, will be expanded at some future date to include components unique to the GCFR.

Inspection Requirements. Table B-1 identifies those reactor internals for which specific inspection requirements exist, or will be included, in the current proposed GCR code. The applicability of these requirements to the GCFR can be assumed under a recent change to the charter of the ASME code body responsible for gas-cooled-reactor system rules of inspection and testing, items 1 through 6. Additional requirements can be anticipated for those components and component functions unique to the GCFR. These have been identified by items 7 through 9 of Table B-1.

Access Provisions. Designs will provide access, including means of material surveillance specimen placement and retrieval, for those components determined to be subject to the inspection requirements of the code. Where possible, existing penetrations, such as for control rods and instrumentation, will be utilized for inspection access. Where special ISI penetrations are required, configurations will provide for viewing (4-inch internal diameter) and material surveillance (6-inch internal diameter). Access to regions above and below the core grid plate is necessary for thermal barrier and core support structure inspections.

Volumetric examination of tubing in helical coil design heat exchangers is an ongoing development; where the requirement to inspect is determined, access to tube sheets will be provided for tube probe examination by the method selected. It should be noted that note 2 of Table B-1 provides in the code an alternative to the requirement to inspect and hence the necessity for a means of component access. Redundant support systems for heat exchanger tube bundles is a case in point. Furthermore, where degradation can be detected for those items identified in the tabulation by means of instrumentation or other in-place monitoring systems, the necessity for in-core surveillance specimens may be negated. This exemption will be included in the next issue of the GCFR code under a new subsection on surveillance of nonmetallic materials. Thermal barrier insulation would be specifically applicable to this exempted category.

Question 7

How will development programs for the GCFR primary system components be affected if the development of HTGR technology is not carried out in the United States substantially beyond the Fort St. Vrain reactor?

Response

The termination of the development of HTGR technology in the United States would impact the development of GCFR primary system components in two areas: the reactor vessel and the steam generator.

Table B-1. In-service inspection requirements--reactor internals

| Item | Component, part | Inspection method | Area/material to be inspected |
|---|---|---|---|
| 1 | Thermal barriers | Visual | Exposed and accessible areas |
| | | Material surveillance | Elevated-temperature structural metals |
| | | (Note 1) | Nonmetallic materials--fibrous blanket and ceramic block insulation |
| 2 | Core support structures | Visual | Exposed and accessible areas |
| | | Material surveillance (Note 1) | Elevated-temperature structural metals |
| 3 | Core lateral restraints | Material surveillance (Note 1) | Elevated-temperature and other structural metals |
| 4 | Liner | | |
| 5 | Core auxiliary heat exchanger support structures | Visual (Note 2) material surveillance (Note 1) | Elevated-temperature structural metals |
| 6 | Core auxiliary circulator support components | Visual (Note 3) | |
| 7 | Steam generator tube bundle support structures (Note 4) | Visual (Note 2) | |
| | | Material surveillance (Note 1) | Elevated-temperature structural metal |
| 8 | Steam generator circulator support components (Note 4) | Visual (Note 3) | |
| 9 | Heat exchanger tubing (Note 4) | Volumetric | |
| | | Material surveillance (Note 1) | Elevated-temperature structural metal |

Notes:

1. Specimens and/or complete components when removed for other reasons.
2. Suitable alternative method with ability to detect failure of each individual load path is acceptable.
3. When withdrawn, disassembled, or made accessible for other reasons.
4. Exempt from inspection requirements when component function is not utilized for shutdown heat-removal operations.

HTGR reactor vessel development, which is applicable to the GCFR Program, includes the following ongoing and planned activities.

PCRV Load Monitor Testing. Testing to develop the design of the load monitor system for the circumferential prestressing system and testing to develop and verify concrete material creep and failure modes for 3-D PCRV analysis.

PCRV Liner Development. Development of an analytical approach to satisfy fracture toughness requirements for Class 1 steel penetrations and closures and development of design criteria for anchorage systems, cooling tubes, and flow restrictors.

Development and testing of liner and adjacent concrete for ability to withstand abnormally elevated temperatures.

PCRV Thermal Barrier Development. Tests to determine if chatter (intermittent slip-stick at sliding surfaces) exists and, if so, parametric tests to determine effects of insulation, compression, helium impurities, etc., on chatter. These tests are to be followed by long-term thermal cycling tests.

Cold vibration tests on a full-scale thermal barrier assembly to determine seismic response of cover plates. Analysis of these response data to be followed by high-temperature long-term resiliency testing with vibration. The HTGR vibration testing also includes tests on Class A and Class B thermal barrier assemblies to verify resistance to flow-induced and acoustic vibrations at reactor operating temperatures.

Tests to establish the integrity of seal members at critical thermal barrier junctions when subjected to thermal cycling.

Thermal properties tests on low- and high-temperature insulation materials, including the effects of a helium environment.

HTGR steam generator development which is applicable to the GCFR Program includes the following ongoing and planned activities:

Wear Protection Test. Tests designed to determine what protection is required for steam generator tubes in order to prevent excessive wear due to thermal cycling and flow-induced vibration.

Superheat Tubesheet Large Forging and Welding Test. Tests for evaluation of large alloy 800H forgings, development of welding techniques for various materials and combinations thereof, and manufacturing inspection, with emphasis on the ultrasonic inspection of alloy 800H.

Tubing In-service Inspection Test. Tests to develop reliable methods for transporting tubing in-service inspection monitoring unit and to develop techniques to perform volumetric inspection.

Bimetallic Weld Test. Tests to produce a failure in steam generator tube dissimilar metal weldments in a reasonably short time. The test cycle, although accelerated, will produce failure that is typical of those experienced in service. The materials being considered are 2-1/4 Cr-1 Mo and alloy 800H.

The termination of HTGR technology development would require that those activities described above be performed and financed by the GCFR Program. The development of the GCFR circulator and circulator-drive system would not be affected by termination of HTGR development. The GCFR Program calls for the construction of a circulator test facility which will allow full-scale development testing and qualification of the circulator and its drive system. Development of these components, therefore, does not rely on future HTGR technology development.

Question 8

We understand from the February 26th meeting that General Atomic is now considering core-disruptive accidents (CDAs) and core melting as containment design bases, and has patterned its reactor siting source term and its containment configuration after the Clinch River design safety approach articulated by the staff in a May 6, 1976 letter to ERDA. Please provide the following:

1. Documentation confirming or correcting relevant material presented to the NRC on February 26, 1976

2. A discussion of why the Clinch River containment design and siting source term are considered appropriate to the GCFR

3. A description of experimental research programs planned to confirm assumptions used in the CDA analysis and the containment system design

Response

1. The GCFR Program is considering the effect of CDAs and core-melting accidents on the maintenance of containment integrity for a period of time yet to be determined.

2. The GCFR Program has tentatively adopted the Site Suitability Source Term, which the NRC has mandated for the CRBR project including the release of 1% of the core plutonium as aerosol into the containment. This Site Suitability Source Term is being interpreted as a generic source term for fast reactors in the same sense as the LWR Site Suitability Source Term is generic to PWRs and BWRs. There is no current basis on which to justify a GCFR source term substantially different from the LMFBR source term.

With respect to CDA releases, there may be differences in the fuel-vaporization fraction and energy release between GCFRs and LMFBRs; these differences are likely to be small compared to the difference in limiting plutonium vaporization at the core level and the plutonium contained in the source term. Furthermore, the ability to contain fuel aerosols within the primary coolant boundary may be quite comparable between GCFRs and LMFBRs. In the LMFBR, there is an efficient means to transmit work energy to the primary vessel by the coolant which can create a release path.

However, the coolant also can reduce the activity available for release from the vessel by coolant washout. In the GCFR, the efficiency of transmitting energy to the reactor vessel is greatly reduced and the ability of the vessel to absorb energy is substantially increased. The coolant washout effect in the LMFBR is replaced by plateout on internal structures in the GCFR.

3. The experimental GCFR safety research program considered necessary to support CDA analyses is currently being reviewed in light of the upflow decision. The following basic experimental programs are in place and are expected to continue, possibly with some redirection of emphasis:

(a) The out-of-pile Duct Melting and Fallaway Test program at Los Alamos Scientific Laboratory has been reduced in scope and renamed Steel Melting and Relocation Test (SMART) program. The objective of these experiments is to provide detailed information on the initiating phase (prior to fuel relocation) during a total loss-of-coolant-circulation with scram accident. These tests simulate a full-size fuel assembly with partial guard assemblies to represent as accurately as possible the power distribution and environment to investigate internal natural circulation, duct wall heat transport as well as cladding melting, relocation, and refreezing. These experiments are principally aimed at guiding methods development and verifying the analysis methods for this accident class. Two full-length, 37-rod experiments have been completed to date.

(b) A direct electric heating facility (out-of-pile) for the GCFR experiments is in the operational checkout phase at the Argonne National Laboratory (ANL). This facility is capable of testing sections of the GCFR fuel rods in an 85-atm flowing-helium environment to simulate full-power/full-flow conditions, as well as transient conditions simulating protected and unprotected power and flow transients. This facility will be used to study a wide variety of physical phenomena related to fuel behavior under high-pressure helium flow conditions, such as the strength of pellet-to-pellet bonds, fuel-swelling effects due to high-pressure helium absorption, fuel fragmentation and sweepout, etc.

(c) The GCFR Program is participating in the Thermite Test Program at ANL to study out-of-pile the penetration characteristics of molten fuel into an axial blanket structure both under pressure-driven-injection and under gravity-drainage conditions. The objective is to determine the feasibility of molten fuel ejection through the axial blankets to remove sufficient fuel from the core to yield permanent subcriticality.

(d) The GCFR Program is participating in the Post-Accident Heat Removal Test Program (out-of-pile) at ANL. This program is developing experimental evidence on molten-pool heat-transfer correlations, on the behavior of molten pools, and on the interaction of other materials with molten $UO_2$. The GCFR Program intends to use this type of experiment to study molten-fuel penetration rates through a steel blockage inside a subassembly as well as into steel structures in the central PCRV cavity floor.

(e) Pre-GRIST Irradiated Fuel Testing. Evidence developed to date indicates that differences in fuel behavior between a GCFR and an LMFBR are almost exclusively due to the coolant differences in the damage and disruption phase and not due to effects built into the fuel during normal operation. Therefore, it may be justifiable to use LMFBR irradiated fuel for in-pile transient testing in the GRIST-2 facility (see Item 6). A combined in-pile/out-of-pile test program is currently under consideration and development to demonstrate that fuel behavior under CDA conditions is not significantly influenced by the preirradiation cooling environment to justify the use of LMFBR fuel for GRIST-2 testing and to make available early data on CDA fuel behavior.

(f) Gas Reactor In-Pile Safety Test (GRIST-2) Program. The GRIST-2 test facility is designed to test up to 37 nearly full-length fuel rods, including axial blankets, under prototypic GCFR flow conditions in the TREAT Upgrade reactor facility at Idaho. The facility, designed by EG&G, Idaho, is in the preliminary design phase. Argonne National Laboratory is the designated experimenter and has responsibility for integrating the GRIST program into the SAREF program. General Atomic will provide test fuel and define end-user experiment needs. Helium Breeder Associates is responsible for program management.

(g) Fuel Aerosol and Energy Release Experiments. No experimental programs in these areas are currently planned. The information developed under the LMFBR aerosol program is considered largely applicable to the GCFR. The need for experiments in these areas will be reconsidered, however, when a more complete assessment of accident consequences in the upflow core design is obtained.

## REFERENCES FOR APPENDIX B


1. Franklin D. Coffman, Jr., LOCA Temperature Criterion for Stainless Steel Clad Fuel, NOREF-0065, June 1976.

2. P. C. Ricardella, Design Basis for HTGR Penetrations and Closures, GA-A13832, General Atomic Company, May 7, 1976.

3. Gas Cooled Reactor Core Disruption Accidents, ANL/RAS/GCFR-76-1, Project Staff, Argonne National Laboratory, November 1976.

4. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low As is Reasonably Achievable (ALARA)," March 1979.

5. "Part 2. Irradiation of GCFR Test Fuel Bundles in the BR2 Helium Loop-MOL," Topical Meeting Proceedings, International Conference on Fast Breeder Reactor Fuel Performance, ISBN-89448-105-3, Monterey, California, March 5-8, 1979.


18645 *U.S. GOVERNMENT PRINTING OFFICE : 1980 0-310-912/172