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GENERAL ATOMIC

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CONCEPTUAL DESIGN OF THE BLANKET AND POWER CONVERSION SYSTEM FOR A MIRROR HYBRID FUSION-FISSION REACTOR

12-MONTH PROGRESS REPORT
FOR THE PERIOD JULY 1, 1975 THROUGH JUNE 30, 1976

by
K. R. SCHULTZ, C. B. BAXI, R. RAO,
A. R. VECA, G. A. BACKUS, J. B. DEE,
E. A. ESTRINE, M. T. SIMNAD, C. C. BAKER,
F. L. OPENSHAW, and H. J. SNYDER, JR.

Prepared under
Contract No. E(04-3)-167
Project Agreement No. 38
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U.S. Energy Research and Development Administration

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report is the result of a conceptual design study done at General Atomic Company during FY-1976 to investigate the feasibility of applying gas-cooled reactor technology to the Mirror Hybrid Fusion-Fission Reactor. The study was done in conjunction with mirror hybrid reactor design efforts at Lawrence Livermore Laboratory and with guidance, support and encouragement from the staff at LLL.

The concept of a hybrid reactor combines a fusing plasma that produces neutrons with a surrounding fissile or fertile blanket that uses these neutrons. Hybrid reactors have been discussed since the earliest days of the controlled thermonuclear research program in the early 1950s. Two excellent review papers have recently been published that summarize much of the work that has been done on hybrid systems (Refs. 2-1 and -2).

Two of the potential features of hybrid reactors that are of particular interest for the study reported here are:

1. The potential of using a multiplying blanket to enhance the energy production of the reactor. This will allow net power production from a sub-breakeven fusion reactor.
2. The potential to produce fissile material which may be subsequently used in fission reactors. This may help to improve the net economics of the fusion reactor and provide a potential source of fissile material for the fission reactor economy.

By allowing improvement in both the energy production and the net economics of a fusion reactor while producing fissile fuel, the hybrid concept may make it possible to achieve a commercially viable fusion-fission power system.

The hybrid power reactor concept devised by Lawrence Livermore Laboratories, described in Ref. 1-3, optimized in Ref. 1-4, and studied during the work reported here makes use of both these potential hybrid characteristics. The hybrid system utilizes a fusion machine that was optimized for neutron production and that requires 1035 MWe of input power to produce 470 MWt of fusion power. The uranium blanket boosts the nuclear power to 4014 MWt with a resulting net electrical power output of 889 MWe. In addition, the blanket produces enough plutonium to sustain six 1000 MWe LWR power plants.

The work reported here was specifically directed at design and analysis of the blanket and power conversion system of the Mirror Hybrid Reactor. The work was a cooperative effort that was closely coordinated with Lawrence Livermore Laboratory. Within General Atomic Company further interdisciplinary cooperation was emphasized. The contract was supervised and coordinated by the Fusion Division. The blanket fuel and module design was done by the Gas-Cooled Fast Reactor Division. The power conversion system design and analysis was done by the Advanced Concepts Division. Advice and data on fuel cycle matters was provided by the HTGR Fuel Engineering Division.

1.2 SUMMARY

This report presents the conceptual design and preliminary feasibility assessment for the hybrid blanket and power conversion system of the Mirror Hybrid Fusion-Fission Reactor. Existing gas-cooled fission reactor technology is directly applicable to the Mirror Hybrid Reactor. There are a number of aspects of the present conceptual design that require further design and analysis effort. The blanket and power conversion system operating parameters have not been optimized. The method of supporting the blanket modules and the interface between these modules and the primary loop helium ducting will require further design work. The means of support and containment of the primary loop components must be studied. Nevertheless, in general, the conceptual design appears quite feasible. Some important parameters are summarized on Table 1.2-1.

There appear to be no inherent features in the hybrid concept that could compromise the safety aspects of the reactor. Further work is needed, however, in the areas of fuel performance analysis and the helium ducting design, support, and containment to assure that the reactor will be adequately cooled under all circumstances. The apparent total absence of potential accidents that could cause the system to become supercritical does offer the hybrid reactor some advantages compared to fission reactor systems. If the maintenance of adequate cooling can be demonstrated, the hybrid reactor could be licensed under the current Nuclear Regulatory Commission fission reactor regulations.

TABLE 1.2-1
MIRROR HYBRID FUSION FISSION REACTOR SUMMARY

Power

Fusion power	470 MWt
Total nuclear power	4015 MWt
Blanket power	3920 MWt
Total thermal power	4382 MWt
Direct converter electrical output	368 MWe
Turbine generator output	1574 MWe
Total electrical power	1942 MWe
Net electrical output	889 MWe
Reactor thermal efficiency	22.1 %
Average plutonium output	2260 kg/yr

Blanket

Blanket modules	16
Fertile material	Uranium silicide (U_3Si)
Fuel configuration	Inconel 718-clad rods
Structural material	Inconel 718
Primary coolant	Helium

Helium Loop

Average pressure	6.08 MPa (60 atm.)
Blanket outlet temperature	530°C (986°F)
Blanket inlet temperature	282°C (539°F)
Helium flowrate	3401 Kg/S (27.0×10^6 lbm/hr)

Helium Circulators

8 steam turbine driven, 13.2 MW (17,700 HP) each

Steam Generators

16 helically-wound, axial flow, once-through

Steam Loop

Pressure: superheater outlet	8.3 MPa (1200 psia)
resuperheater outlet	5.4 MPa (778 psia)
feedwater inlet	10.1 MPa (1460 psia)
Temperature: superheater outlet	445°C (831°F)
resuperheater outlet	504°C (940°F)
feedwater inlet	171°C (340°F)
Steam flow rate	1612 Kg/s (12.8×10^6 lbm/hr)

TABLE 1.2-1 (Continued)

<u>Auxiliary Circulators</u>	8 electric motor driven 2.75 MW (3700 HP) each
<u>Auxiliary Heat Exchangers</u>	8 helically-wound axial flow
<u>Steam Turbine-Generator Set</u>	Tandem-compound 1576 MWe
<u>Approximate NSSS Cost</u> [*]	\$295M (\$332/KWe)

^{*}Excluding nuclear fusion island.

REFERENCES

- 1-1. Leonard, Jr., B. R., "A Review of Fusion-Fission (Hybrid) Concepts," Nucl. Tech. 20, December 1973, pp. 161-178.
- 1-2. Lidsky, L. M., "Fission-Fusion Systems: Hybrid, Symbiotic, and Aegean," Nucl. Fusion 15, February 1975, pp. 151-173.
- 1-3. Moir, R. W., et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," UCRL-51797, June 25, 1975.
- 1-4. Lee, J. D., D. J. Bender, and R. W. Moir, "Optimizing the Mirror (Fusion-Fission) Hybrid Reactor for Plutonium Production," UCRL-76986, November 17, 1975.

2. SCOPE AND GOALS OF STUDY

2.1 SCOPE OF STUDY

The general scope of this study was to carry out engineering design studies related to mirror reactor concepts under the general direction of the Lawrence Livermore Laboratory. Those areas in which General Atomic Company possesses particular experience and expertise were to be emphasized, especially in the design of high temperature gas-cooled blankets and structures. In particular, GA was to perform a detailed feasibility study of the blanket design and the power conversion system design of the Mirror Hybrid Fusion-Fission Reactor. The initial conceptual design presented by Lawrence Livermore Laboratory in Ref. 2-1 and optimized in Ref. 2-2 was to be evaluated and any necessary design modifications were to be made to obtain a final conceptual design of the blanket and power conversion system.

The blanket design aspect of this study was to entail a detailed feasibility study of Lawrence Livermore Laboratory's blanket module design. Consideration was to be given to heat transfer and fluid flow within the entire module. Potential materials problems were to be reviewed as was the mechanical design of the module and sub-module. Specific design advice was to be provided in the areas of fission product retention and cladding surface preparation (roughening). Evaluation of fuel material choices was to be made.

The power conversion system primary, auxiliary and secondary loops were to be laid out. The performance of this system was to be evaluated. Special consideration was to be given to the requirements imposed by safety

and licensing concerns and by startup and control procedures. Recommendations on blanket and power conversion system design modifications and improvements were to be made. Areas requiring further study were to be delineated. Cost estimates for the blanket and nuclear steam supply system were to be provided.

2.2 GOALS OF THE STUDY

After consultation with Lawrence Livermore Laboratory, it was agreed that the conceptual design should use conventional gas-cooled reactor technology as much as possible. Extrapolation of the state-of-the-art in design of the various components was to be kept to a minimum. The goal of the study was to obtain a conceptual design that would have potential for near-term construction with a minimum requirement for large scale development programs.

Many of the blanket and power conversion system design and operating parameters were specified by Lawrence Livermore Laboratory in the initial conceptual design (Refs. 2-1 and 2-2). Nevertheless, a goal of this study was to assess alternatives and to make appropriate changes where such changes were desirable.

A primary goal of this study was to maximize the inherent and designed safety features of the hybrid reactor system. Although tradeoff studies were not made, cost considerations were to be of secondary concern compared to the safety and licensing aspects of the reactor system.

The goals of the study, then, can be summarized as the design of a mirror hybrid reactor blanket and power conversion system that is technically feasible. The system should be constructable using existing gas cooled reactor technology and should be licensable under existing Nuclear Regulatory Commission regulations.

REFERENCES

- 2-1. Moir, R. W., et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," UCRL-51797, June 25, 1975.
- 2-2. Lee, J. D., D. J. Bender, and R. W. Moir, "Optimizing the Mirror (Fusion-Fission) Hybrid Reactor for Plutonium Production," UCRL-76986, November 17, 1975.

3. SYSTEM PARAMETER CHOICES

3.1 TEMPERATURE AND PRESSURE

The system parameter choices, in general, influence the loop component design and the plant economics. In particular, the helium temperature and pressure parameters have a significant bearing on the design feasibility, fabricability and power conversion loop sizes of such components as the helium circulators and the coolant piping. A preliminary analysis was performed to select these two parameters.

3.1.1 Temperature Parameter

Helium gas inlet temperature at the blanket ($T_{\text{Blk-in}}$) and the outlet temperature ($T_{\text{Blk-out}}$) together determine the helium mass flow rate ($\dot{m}$) required to transfer the design blanket fission heat rate ($\dot{Q}_f$), since the specific heat (C_p) of helium remains fairly constant at the conditions considered. The heat transfer equation describing these parameters can be expressed as

$$\dot{m} c_p (T_{\text{Blk-out}} - T_{\text{Blk-in}}) = \dot{Q}_f$$

Thus, for a given design $\dot{Q}_f$ it is preferred to maximize ($T_{\text{Blk-out}} - T_{\text{Blk-in}}$), within permissible limits, to lower $\dot{m}$, which in turn minimizes the coolant piping size, circuit pressure drop, and the circulator pumping power.

$T_{\text{Blk-out}}$, which represents the bulk helium temperature as it leaves the blanket, is limited by the fuel temperature characteristics. As explained in Chapter 4, the maximum $T_{\text{Blk-out}}$ obtainable is limited to about 803°K (986°F). The blanket helium inlet temperature, $T_{\text{Blk-in}}$, could be minimized

to the extent the steam generator performance and cost would not be seriously compromised. A lower $T_{\text{Blk-in}}$ results in a larger steam generator heat transfer surface area. After a few power conversion loop design iterations, it was found that a reasonable $T_{\text{Blk-in}}$ value will be about 280°C (540°F).

It should be noted that the secondary coolant terminal temperatures in the steam generator [505°C (940°F) steam outlet, and 171°C (340°F) water inlet] are closely coupled to the helium temperatures and they are also limited by the 'pinch-point' temperature difference. The pinch-point temperature difference is the difference between the helium temperature and the water temperature at the inlet to the evaporator zone of the steam generator. This is the minimum differential between helium and water temperatures and should be kept in the range of 40 to 55°C.

In summary, the following helium gas temperatures were selected for the current phase of work.

Helium bulk temperature leaving blanket	530°C (986°F)
Helium temperature leaving steam generator	270°C (520°F)
Helium temperature entering the blanket or exiting the circulator	280°C (540°F)

3.1.2 Pressure Parameter

Helium pressure in the system has an impact on the coolant piping size and circulator design. A lower system pressure, such as 2.03 MPa (20 atm) recommended by Ref. 3-1 is neutronically desirable as a thin blanket first wall may be used. A low system pressure for the power conversion loop, however, presents problems. Two preliminary conceptual design specifications were made for the power conversion system, one at 6.08 MPa (60 atm) and one

at 2.03 MPa (20 atm). Both designs produced 1574 MWe. The results of these preliminary designs are shown on Table 3.1-1. These data are preliminary and may not agree completely with the conceptual design presented in Chapters 4 and 5. Nevertheless, these preliminary data allow comparison of the 6.08 MPa (60 atm.) design and the 2.03 MPa (20 atm.) design.

The low gas densities associated with low system pressures require large size piping to maintain a practical flow dynamic head. In addition, the parameter $\Delta p/p$ (ratio of circulator pressure rise to pressure) characterizes the selection of a circulator design with an associated flow rate. For ratios of $\Delta p/p \geq 0.03$ multistage circulators are required. The design of these large multistage circulators would involve additional developmental studies.

In the case of a 2.03 MPa (20 atm) helium system pressure, pipe sizes in the range of 1.72-1.83 m (68-72 in.) are required with 3500 Pa (0.5 psi) coolant dynamic head. Connection of such piping to a blanket module having a maximum width of 2.16 m (85 in.) presents practical difficulties. In addition, the design of a piping system of this size would present severe problems in items such as seismic constraints and thermal expansion provisions. Above all, a value of $\Delta p/p \approx 0.1$ computed with this pressure would require a 4 or 5 stage circulator and the feasibility of such a design might well be a question. Pumping power in the order of 500 MW, about one-third of the desired power conversion system output, would be required.

There are no safety or feasibility problems that would be exacerbated by use of the 6.08 MPa (60 atm) system pressure. Hence a higher system

TABLE 3.1-1
MHFFR POWER CONVERSION SYSTEM PARAMETERS
(Comparison of Preliminary Results for 20 and 60 Atm Pressure)

Description	Unit	20 Atm Case	60 Atm Case
<u>Power Distribution</u>			
1. Thermal power into steam generators (BLKT + DC)	MW	4746	4422
2. Power conversion loop output	MW	1573.6	1573.6
3. Helium circulator power required	MW	530.3	134.0
4. Power conversion system efficiency	%	33.14	35.56
<u>Helium Data</u>			
5. Total flow rate into steam generators	kg/hr (Lbm/hr)	15.5×10^6 (34.2×10^6)	12.4×10^6 (27.3×10^6)
6. Flow rate from blanket	kg/hr (Lbm/hr)	13.7×10^6 (30.1×10^6)	10.9×10^6 (24.0×10^6)
7. Flow rate from direct converter	kg/hr (Lbm/hr)	1.9×10^6 (4.1×10^6)	1.48×10^6 (3.3×10^6)
8. Flow rate/module (16 modules)	kg/hr (Lbm/hr)	0.86×10^6 (1.9×10^6)	0.68×10^6 (1.5×10^6)
9. Temperature at steam generator inlet	°C (°F)	550.0 (1022.0)	530.0 (986.0)
10. Temperature at signal outlet	°C (°F)	313.3 (596.0)	273.3 (524.0)
11. Temperature at circulator outlet	°C (°F)	337.7 (640.0)	282.2 (540.0)
12. Average helium pressure	MPa (psia) (atm)	2.03 (294) (20)	6.08 (882) (60)
13. Pressure drop			
Blanket	MPa	0.083 (12.0)	0.083 (12.0)
Ducts	(psid)	0.028 (4.0)	0.028 (4.0)
Steam Generator		0.055 (8.0)	0.055 (8.0)
Total ΔP		0.165 (24.0)	0.165 (24.0)

TABLE 3.1-1 (Continued)

Description	Unit	20 Atm Case	60 Atm Case
<u>Steam/Water Data</u>			
Steam generator inlet conditions			
14. Pressure	MPa (psia)	21.9 (3175)	10.2 (1475)
15. Temperature	°C (°F)	211 (412)	171 (340)
16. Enthalpy	Joules/kg Btu/lb	0.91 x 10 ⁶ 391	0.73 x 10 ⁶ 313
17. Flow rate	kg/hr (Lbm/hr)	6.7 x 10 ⁶ (14.7 x 10 ⁶)	5.9 x 10 ⁶ (12.9 x 10 ⁶)
Steam generator outlet conditions			
18. Pressure	MPa (psia)	3.67 (533)	4.95 (719)
19. Temperature	°C (°F)	510 (950)	504 (940)
20. Total steam/water pressure drop	MPa (psid)	2.39 (347)	2.14 (310)
<u>Steam Generator Data</u>			
21. Number of units		24	16
22. Approximate size	meters (ft)	3.51 (11.5) diam X 20.7 (68) high	3.20 (10.5) diam X 15.2 (50) high
23. Tubing (OD)			
Re-superheater	cm	2.54 (1.00)	2.54 (1.00)
Superheater	(in.)	2.54 (1.00)	2.54 (1.00)
Evaporator		1.91 (0.75)	1.91 (0.75)
Economizer		1.91 (0.75)	1.91 (0.75)
<u>Helium Circulator Data</u>			
24. Number of units		40 (approx)	8
25. Power rating per circulator	MWatt (hp)	≈15 ≈20,000	≈13 ≈17,400
26. Type		Multistage	Single stage
27. ΔP/P		≈0.08	0.027

pressure was preferred and for the first iteration a system pressure of 6.08 MPa (60 atm) was selected. With this pressure piping sizes were found to be reasonable [diameters $\leq$ 1.22 m (48 in.)]. A single-stage circulator will be adequate and the pumping power is also relatively modest, about 9% of the power conversion system output.

It appears that a higher system pressure in the order of 6.08 MPa (60 atm) is preferable and is used for the present study. Optimization of the system pressure has not yet been done. The tradeoffs are of an economic nature. Higher system pressures result in increased system efficiency and decreased component size. Low pressure, on the other hand would allow use of less structural material which could save on component costs and will also improve the neutronics performance of the blanket. This optimization should be pursued in the future. The entire set of power conversion system operating parameters -- pressure, pressure drop, temperature and temperature rise -- should be optimized for the specific characteristics of the mirror hybrid fusion-fission reactor.

REFERENCE

1. R. W. Moir, et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," UCRL-51797, June 1975.

4. BLANKET DESIGN

4.1 FUEL DESIGN

4.1.1 Materials Selection

4.1.1.1 Introduction. The candidate fuel materials that have been evaluated for use in the MHFFR include uranium metal, uranium alloys, uranium oxide, uranium carbide, and uranium nitride. A comparative evaluation was made in terms of the important parameters, such as fuel density, neutron economy, physical, mechanical and corrosion properties, irradiation effects, compatibility with cladding and with impurities in the helium coolant (particularly tritium), and fabrication cost.

The operational conditions tend to favor the use of metallic uranium alloys, in particular uranium silicide (U_3Si) fuel which has been under development in Canada for the CANDU reactor program during the past decade. The characteristics of the metallic and alloyed uranium fuels will be summarized in this section. Uranium oxide is less desirable because of the significantly lower fuel density, lower neutron economy, and higher fabrication cost compared to uranium silicide. Uranium carbide and nitride also are less desirable because of similar reasons. However, uranium carbide may be a suitable alternative fuel if the silicide should not qualify because of temperature limitations. Table 4.1-1 compares some of the pertinent properties of candidate fuel materials.

TABLE 4.1-1
PROPERTIES OF FUELS

	U* (modified)	U-10 Mo	U-5Fs** (EBR-II Fuel)	U ₃ Si	UC	UO ₂
U-density (g/cm ³)	19.04	17.12	18	15.5	12.97	9.65
Melting point	1133°C	1150°C	1002°C	930°C	2400°C	2800°C
Thermal conductivity cal-sec-cm ² -°C-cm	0.095(600°C) 0.08(350°C)	0.07(600°C) 0.05(350°C)	0.08(600°C) 0.06(350°C)	0.05 (RT to	0.06(700°C)	0.007(1000°C) 0.005(2000°C)
Thermal Expansion x 10 ⁻⁶ /°C	19(650°C)	12.3(400°C)		14(600°C)	11.6(1000°C)	10.1(1000°C)
Max. operating temperature	610°C	700°C	660°C	900°C	2000°C	2400°C
Irradiation swelling at 1% bu	3 v/o into axial hole	6 v/o at >7x10 ¹³ f/cc/ sec. Higher swelling at lower f/cc/sec	6 v/o, or 4 v/o with 400 ppm Si	ΔD = 1.2% with axial hole of 12 v/o	6 v/o at <1000°C 1 v/o at <900°C	0.4 v/o at <1500°C

*U containing 360 ppm Fe + 1000 ppm Al _ 30 ppm Si _ 900 ppm C.

**U containing 5 % Fissium (0.22 Zr + 2.5% Mo + 1.5% Ru + 0.3% Rh + 0.5% Pd)

4.1.1.2 Metallic Fuels. Uranium alloys have been used in the fuel elements in the carbon dioxide cooled first generation nuclear power reactors in Britain (Calder Hall or Magnox Reactors) and France, and in fast breeder reactor prototypes (Dounreay, EBR-I and -II and Fermi) (Ref. 4.-1). Uranium fuel rods are produced by casting, rolling and extrusion, followed by machining. Uranium undergoes three phase changes up to its melting point, namely, the alpha (orthorhombic) up to 666°C, beta (tetragonal) from 666 to 771°C, and the gamma (body centered cubic) from 771 to 1130°C (melting point). Anisotropic alpha-uranium is subject to dimensional changes under both thermal cycling and nuclear irradiation. The changes are governed by the structure and composition as well as temperature and burnup.

Metallic uranium fuels are generally limited to operation below approximately 600°C maximum temperature, and to relatively low burnups of about 5000 MWD/MTU (0.5% burnup) because of irradiation damage. Swelling and growth become excessive primarily because of fission gas bubbles at high temperatures and the formation of lattice defects (vacancies, interstitials, dislocation loops, etc.) at low temperatures. Irradiation-creep is also a problem at low temperatures. There is little swelling below about 400°C. The growth reaches a maximum in the range 400 to 600°C. Above approximately 700°C fission gas swelling predominates. The Magnox reactors used uranium adjusted with iron (260 ppm), aluminum (650 ppm), carbon (800 ppm), silicon (20 ppm) and nickel (50 ppm), and the French EDF reactors used U-1% Mo in EDF-1, 2, 3 and 4, and "Sicral" alloy, uranium containing Al (700 ppm), Fe (300 ppm), Si (120 ppm), and Cr (80 ppm) in EDF-5. These minor alloying elements result in grain size refinement and

very finely divided precipitates and the swelling diminishes by several orders of magnitude. These additions modify the $\alpha - \beta$ transformation and favor grain refinement and absence of preferred orientation upon quenching these alloys from temperatures in the beta range. Thus, heat treatment minimizes distortion of fuel elements due either to thermal cycling or irradiation growth, since induced intergranular stresses and strains are reduced, a typical grain size effect. These fuels were clad with magnesium alloys (Mg containing 0.8% Al, 0.002 to 0.05% Be, 0.008% Ca and 0.006% Fe in the U.K., and Mg containing 0.6% Zr in France), with maximum clad temperature of 480°C.

The performance of modified metallic uranium fuels has been greatly improved by providing voidage with an axial hole (Ref. 4.-2). Test elements with 5% void have been irradiated to 18,000 MWD/MTU (2.2 at. % burnup) at fuel temperatures of up to 610°C. The available voidage in the fuel was consumed at a rate of 3% per atom % burnup. No irradiation tests have been reported on the void-containing uranium at elevated temperatures above the phase transformations. The density changes that accompany the phase changes could lead to severe distortions of the fuel pins. It is not known whether cladding restraint and the axial hole would accommodate the volume changes that accompany phase changes, although the metal fuel (U-5 Fissium) in EBR-II has operated at temperatures where the fuel undergoes a phase change from the alpha phase to the gamma phase. The density change is also reflected in a significant effect on the power coefficient (Ref. 4.-3).

Uranium alloyed with 10% by weight Mo has a metastable single phase body centered cubic structure, with higher strength, solidus temperature, and irradiation stability (about 10% diametral increase at 1% burnup) that is maintained if the fission rate is above a threshold (3.5×10^{13} f/cc/sec). It swells rapidly at temperatures above approximately 800°C. It was used as fuel in the Dounreay, Hallam, and in the Fermi fast breeder demonstration reactor. However, its irradiation stability was degraded by eutectoid decomposition of the γ -phase in the lower flux of Fermi (fuel maximum 600°C, clad 570°C), compared to its behavior in the high neutron flux test reactors in capsule irradiation tests. The molybdenum degrades the neutron economy and increases the fabrication cost significantly (about twice the cost of UO_2).

Metallic fuels for breeder reactors have been developed and studied at Argonne National Laboratory, namely, uranium-5% fissium and an alloy of U-15% Pu-10% Zr (Ref. 4.-4). The latter fuel has been irradiated to 6 atom % burnup when adequate void space was provided between the fuel and cladding to accommodate 25 to 30% swelling and a plenum for fission gases. The fuel was sodium-bonded to the cladding and had a smear density of 75%. It attained a power rating of 15 kW/ft and expanded to touch, but not strain, the cladding. The swelling rate was 2-1/2% per atom % burnup (the solid fission products accounted for 10%, the fission gas bubble swelling for 22% to 66%, and 60 to 70% of the fission gases were released via cracks and fissures). There was a marked effect of external pressure upon swelling. For example, the swelling per one atom percent burnup was 50% at 1 atm, and 10% at 67 atm. The cast alloy U-5 w/o fissium* has been tested as a fuel

* A typical fission composition in wt. % is 49.2% Mo, 40.2% Ru, 6% Rh, 3.6% Pd and 1% Zr.

in EBR-II. At 600°C, U-5 wt % Fs alloy shows a swelling rate of about 8% volume increase per at. % burnup. It begins to swell significantly and to release fission gases at approximately 700°C.

The maximum fuel temperature in EBR-II was 660°C, with maximum clad temperature of 584°C. The swelling rate was about 6% $\Delta V/V$ per at. % burnup. The U-5% Fs alloy with 400 ppm silicon decreased the swelling rate and was good to about 1.2% burnup. Significant decrease in density occurred when this alloy was thermally cycled between the alpha and the alpha-plus-gamma temperature. The alloy swelled most rapidly in the range 575-650°C when post-irradiation annealed. Relatively thin niobium cladding (0.009-in. thick) restrained the swelling effectively. Fuel burnups as high as 5.8 at. % were achieved at center fuel temperatures up to 700°C (650°C cladding temperature). Large reductions (as much as 30%) in the thermal conductivity of the irradiation fuel were measured.

4.1.1.3 Uranium Silicide, U_3Si . During the past decade extensive work has been carried out at Chalk River, AECL, on the development of uranium silicide (U_3Si has 96.2 wt. % U and 3.8 wt. % Si) fuel for the CANDU reactors (Ref. 4.-5). The interest in U_3Si as a fuel arose from its unique combination of high uranium density (15.58 g/cm³) with low parasitic absorption, excellent corrosion resistance (especially with 1.5% Al additions), and high thermal conductivity (0.2 W/cm°C). A peritectoid phase change takes place in U_3Si at 930°C which is taken as the upper limit of operation. This allows a maximum heat rating ($\int \lambda d\theta$) of 86 W/cm in a U_3Si rod, with fuel surface temperature of 500°C (compared to 72 W/cm for a UO_2 fuel rod). The U_3Si fuel was estimated to have a cost advantage of 10 to 20% over UO_2 fuel, because of lower fabrication costs and higher U density.

The irradiation behavior of U_3Si has been studied extensively (Refs. 4.-6, 4.-7, and 4.-8). The results indicate that this fuel has good irradiation stability to reasonably high burnups up to about 900°C. The swelling can be very effectively restrained by the cladding and coolant pressure that forces the fuel to swell into axial or peripheral voidage machined into the fuel rods. The maximum swelling occurs at 500°C and decreases both above and below this temperature. However, above 900°C rapid swelling occurs during post-irradiation annealing. U_3Si elements irradiated at linear power outputs of 78 kW/M to burnups in excess of 19,000 MWd/tonne U (455 MWh/KgU) swelled only 1.2% diametral increase. The $U_3Si + 1.5\% Al$ alloy showed even less swelling, and has the ability to survive large step increases in power. U_3Si-Al elements with 15 volume % central axial void have operated at high power (75 kW/M) at maximum fuel temperature of about 660°C to burnups of at least 25,000 MWD/Tonne U (500 MWh/KgU or 2% burnup) with diameter increases of less than 1.2%.

4.1.1.4 Fuel Material Choice. From the discussions presented above, three fuel material candidates were selected as viable candidates for final evaluation: Uranium-molybdenum alloy, uranium silicide and uranium carbide. Since any of these materials would be an acceptable choice for the MHFRR, the final selection was based on cost estimates for the system economics that each alternative would dictate.

The economics of the various fuels is affected by four factors:

1. Basic fuel cost,
2. Fabrication cost,
3. Revenue from power produced, and
4. Revenue from fissile production.

Compared to UO_2 cost as a base, it is estimated that U-Moly will be thrice as expensive (due to stringent Q/A requirements), UC will be twice as expensive and U_3Si cost will be on par with UO_2 . The fabrication cost of blanket rods is a function of number of rods in the reactor. Due to lower peak temperature permissible in the U-Moly, the allowable rod diameter is smallest amongst the three fuels. The allowable rod size for UC and U_3Si is approximately the same as the cladding temperature is the limiting parameter. The estimated fabrication costs (including basic fuel costs) are summarized in Table 4.1-2 as are some of the important physical characteristics of the rod designs. These estimates are based on rough first approximations of \$7/rod for hardware cost and \$6/rod for pelletizing cost.

Revenues from power and fissile production are roughly proportional to the Uranium density in the fertile material. The estimates of changes in power and fissile production with U-Moly as reference, are shown in Table 4.1-2. These estimates are based on data provided by Lawrence Livermore Laboratory for U-Moly and UC and by density estimates for U_3Si .

Finally, taking U-Moly as the reference case, the economics of UC and U_3Si were evaluated. As shown in Table 4.1-2, UC represents a loss of 24.6 M\$/year and U_3Si a benefit of 7.1 M\$/year compared to U-Moly.

Recommendation

The economics of U_3Si are somewhat better than U-Moly, and much better than UC. However, use of U_3Si also represents the following advantages over U-Moly:

1. The quality assurance requirements during fuel fabrication for U_3Si will be less stringent than for U-Moly.

TABLE 4.1-2
ECONOMICS OF CANDIDATE FERTILE MATERIALS FOR MHFFR

	U-Moly	UC	U ₃ Si
Uranium density (g/cc)	17.12	12.97	15.5
Maximum operating temperature (°C)	700	2000	900
Approximate rod diameter (mm)	10	18	17
Number of rods	2.6×10^6	1.2×10^6	1.3×10^6
Basic fuel cost (\$)	70×10^6	47×10^6	23×10^6
Blanket rod hardware cost (\$)	18×10^6	8×10^6	9×10^6
Pelletizing cost (\$)	16×10^6	7×10^6	8×10^6
Total blanket rod cost (\$)	104×10^6	62×10^6	40×10^6
Blanket assembly cost (M\$)	154	112	94
Change in electrical power produced with U-Moly as reference	--	- 154 MW(e)	- 71 MW(e)
Change in fissile production with U-Moly as reference	--	- 770 kg/yr	- 289 kg/yr
Change in capital cost and reloads (M\$/year)	--	+ 17.1	+ 24.1
Change in revenue (M\$/year)	--	- 41.7	- 17.0
Net benefit (+) or loss (-)	--	- 24.6	+ 7.1

2. Due to higher rod diameter, the flow Reynolds number for U_3Si will be higher and away from the undesirable laminar flow range,
3. For U_3Si the silicon content is only 3.8% compared to 8 to 10% Molybdenum in the U-Moly alloys. This will be beneficial for neutronics, and
4. Due to larger rod diameter for U_3Si fuel, for the same volume fraction, the rod-to-rod clearance will be larger. This is an advantage for mechanical design of the assembly.

In summary, U_3Si represents a number of advantages compared to all other candidate materials. It is recommended as the fertile material for the MHFFR.

4.1.2 Fuel Configuration

The following configurations were evaluated for the blanket fuel of the MHFFR:

1. Honeycomb,
2. Plates,
3. Tangential rods with cross flow,
4. Radial rods with solid pellets, and
5. Radial rods with annular pellets.

Evaluation of these configurations (Table 4.1-3) from thermal-hydraulic as well as manufacturing considerations indicated that for U_3Si as the blanket fuel, radial rods with annular pellets represents the best configuration.

For the reactor pressure of 6.08 MPa (60 atm), the rods will be sealed and prepressured to 83×10^4 Pa (120 psi). Due to the small burnup to which the fuel is subjected (1% peak and 0.56% average) and the highly

TABLE 4.1-3
EVALUATION OF MHFFR BLANKET FUEL CONFIGURATIONS

Shape	Remarks
Honeycomb	<ol style="list-style-type: none"> 1. Small holes required due to small heat transfer coefficient. 2. Technology for fabrication and canning needs development.
Plates	<ol style="list-style-type: none"> 1. Bonding between fuel and clad necessary. 2. Difficult to bond Inconel.
Tangential rods	<ol style="list-style-type: none"> 1. Large variation of heat transfer coefficient around the periphery. 2. No advantage in average heat transfer coefficient.
Radial rods with solid pellets	<ol style="list-style-type: none"> 1. No problems except possible excessive swelling.
Radial rods with annular pellets	<ol style="list-style-type: none"> 1. Represents ideal shape from thermal considerations as the maximum clad and fuel temperatures will be constant. 2. Fabrication technology is developed. 3. With 8% axial hole, diametral swelling for U_3Si fuel will be limited to 1.2% (References 4.-5 and 4.-6).

retentive metallic fuel, fission product release will be negligible and will not contribute to any increase in rod plenum pressure.

The cladding material is Inconel 718. A clad thickness to rod diameter ratio of 0.01 is sufficient to limit the clad stresses within the allowable 296 MPa (43,000 psi) stress limit for a clad hot spot temperature of 800°C. This thickness is conservative in that no internal pressure was assumed.

4.1.3 Fuel Design and Performance

The objective of this task was to determine a fuel design (within the economic constraints of MHFFR) which did not exceed any safe operating limits of temperatures or stresses.

The material selected for clad and blanket structure is Inconel 718. The fuel material is Uranium Silicide (U_3Si). The temperature limits were fixed at:

Maximum hot spot clad temperature	800°C
Maximum hot spot fuel temperature	900°C
Maximum pressure vessel temperature	500°C

The inlet and outlet coolant temperature to the blanket module were fixed at 280°C and 530°C. These temperatures enable a safe design of the blanket fuel while achieving a reasonable thermal efficiency in the steam generator. No effort was made to optimize these temperatures.

Fuel design was based on hot spot temperatures obtained by applying the hot spot factors to the nominal temperatures calculated in an average channel.

Hot Spot Factors

The effect of theoretical and experimental analysis uncertainties, instrumentation accuracy, manufacturing tolerances, physical properties and correlation uncertainties must be considered in thermal-hydraulic analysis of MHFFR blanket so that the design limits are not exceeded during operation. The various uncertainties are accounted for in hot spot factors. The hot channel is defined as the channel affected by the simultaneous adverse occurrence of all uncertainties and the predicted hot spot temperatures in the channel are compared with the required limit.

A semi-statistical method (Reference 4.-9) adopted for FFTF, CRBRP, and GCFR analyses was used in the present analysis. Using this method, hot spot factors for MHFFR blanket were evaluated. These are summarized in Table 4.1-4.

The hot spot factors in Table 4.1-4 are used to calculate the hot spot temperatures as follows:

$$T'_c = T_I + \Delta T_B * F_b + \Delta T_f * \overset{*}{F}_f + \Delta T_c * F_c \quad (1)$$

and

$$T'_f = T_I + \Delta T_B * F_b + \Delta T_f * \overset{+}{F}_f + 2\Delta T_c * F_c \\ + \Delta T_g * F_g + \Delta T_f * F_R \quad (2)$$

where,

T'_c = mid-wall hot spot cladding temperature,

T_I = nominal inlet temperature of the coolant

ΔT_B = nominal bulk temperature rise

F_b = hot spot factor for bulk coolant rise

ΔT_f = nominal film drop

$\overset{*}{F}_f$ = hot spot factor for film drop (clad)

TABLE 4.1-4
HOT SPOT FACTORS FOR MHFFR BLANKET

F_b , hot spot factor for coolant temperature rise	1.3
F_f , hot spot factor for film ΔT	1.25^+ (1.8^*)
F_c , hot spot factor for cladding ΔT	1.25
F_g , hot spot factor for ΔT between cladding ID and fuel surface	1.5
F_R , hot spot factor for fuel ΔT uncertainty	1.2

* For hot spot cladding temperature.

+ For fuel temperature calculations.

ΔT_c = nominal temperature drop through half the clad thickness

F_c = hot spot factor for cladding

T'_F = hot spot fuel temperature

F_f = hot spot factor for film drop (fuel)

ΔT_g = nominal temperature drop through the clad pellet gap

F_g = hot spot factor for gap temperature drop

ΔT_F = nominal temperature drop in the fuel

F_R = hot spot factor for fuel ΔT uncertainty

Thermal Analysis

Thermal analysis was performed for an average flow channel of the rod bundle to obtain the nominal temperatures. Hot spot factors were applied to these temperatures to obtain the hot spot temperatures.

The distribution of power generation rate (approximated at the end-of-life) in the blanket is shown in Fig. 4.1-1. The coolant flow was treated as a one dimensional incompressible flow. The heat transfer coefficient for wire wrapped bundle is 10 to 20% larger than the standard turbulent-flow correlation (Ref. 4.-10). As a conservative approach it was assumed that the heat transfer coefficient in MHFFR blanket is 10% larger than turbulent flow in a tube.

The first-wall of MHFFR blanket with the baffle is designed so that the velocity of flow remains constant as the flow turns and part of the flow goes through the baffle. This is achieved by providing holes in the baffles with increasing diameter towards the center. The maximum first-wall temperature could be limited to 468°C with this arrangement.

Fig. 4.1-1. Power distribution in MHFFR blanket at end-of-life

Figure 4.1-2 shows the maximum hot spot cladding and hot spot fuel temperatures as a function of rod diameter. In order to limit the hot spot cladding temperature to 800°C, a rod diameter of 17 mm was selected. The peak hot spot fuel temperature for rod diameter of 17 mm is 904°C, which is within the calculational uncertainty of the 900°C limit.

The axial distribution of nominal temperatures for 17 mm rod diameter is plotted in Fig. 4.1-3. The cladding and the fuel temperatures are almost uniform except for the first few centimeters where the entrance effect is dominant. The hot spot temperatures for the recommended rod diameter are plotted in Fig. 4.1-4.

Regeneration effects due to heat transfer between hot and cold helium in the blanket module are quite small due to small heat transfer areas available and large mass flow in helium. The regeneration effect was estimated to be less than 5°C.

Pressure Drop

Since the Mach number of flow in the MHFFR blanket is much smaller than 1, the flow was treated as incompressible. The following relations were used to calculate the pressure drop through the MHFFR blanket module.

$$\Delta P_f = \frac{fL}{D_e} \frac{\rho V^2}{2g_c} \quad (3)$$

$$\Delta P_l = K \frac{\rho V^2}{2g_c} \quad (4)$$

$$\Delta P_w = M \frac{fL}{D_e} \frac{\rho V^2}{2g_c} \quad (5)$$

Fig. 4.1-2. Hot spot temperatures as function of rod diameter for MHFRR blanket

Fig. 4.1-3. Nominal temperatures in the MHFFR blanket for a rod diameter of 17 mm

Fig. 4.1-4. Hot spot temperatures in the MHFFR blanket

where,

ΔP_f = pressure drop for flow through passages

ΔP_l = pressure drop due to losses (e.g., turning, expansion, contraction, etc.)

ΔP_w = pressure drop in wire wrapped bundle

f = friction factor

L = length of passage

D_e = hydraulic diameter

ρ = average density of coolant

V = velocity of flow

g_c = gravitational constant

K = loss coefficient

M = friction factor multiplier for wire wrapped bundle

$$= \left[\frac{1.034}{(P/D_e)^{0.124}} + \frac{29.7 (P/D_e)^{6.94} Re^{0.086}}{(M/D_e)^{2.239}} \right]^{0.885}$$

P = rod-to-rod pitch

H = wire wrap pitch

Re = Reynolds number of flow

The friction factor correlations and loss coefficients in Eqs. (3) to (5) were obtained from Ref. 4.-11 and the friction factor multiplier in Eq. (5) is a Novendstern correlation (Ref. 4.-12).

The blanket submodule was modeled as a flow resistance network. Using Eqs. (3) to (5) the following pressure drops were calculated.

Total pressure drop in module	3.9×10^4 Pa (5.7 psi)
Pressure drop in blanket region	1.2×10^4 Pa (1.8 psi)
Pressure drop in module during shutdown (depressurized) cooling	0.2×10^4 Pa (0.3 psi)

Summary of Blanket Data

The significant dimensions and thermal-hydraulic parameters for MHFFR blanket are summarized in Table 4.1-5.

4.2 MODULE DESIGN

4.2.1 Material Selection

The initial design studies for the MHFFR blanket module were carried out using 316 stainless steel as the reference cladding and structural material. This material was initially chosen because it is currently the reference material for the core component of most fast breeder reactor concepts, both here in the U.S. and abroad. Its properties, therefore, are well characterized under the influence of high temperature and fast neutron fluence. When applied to the MHFFR blanket, however, the 316 SS does not have the high strength capability needed, especially for the first wall which is highly stressed but which must be kept thin. Consequently, the resulting module design is very bulky which adversely effects the neutron economy of the plant. Upon further consideration, it was also determined that although 316 SS is well characterized in fast reactor environment, there is no experience at all with this material in the expected fission-fusion environment which can have first wall peak neutron energies of 14 MeV and temperatures of 500°C. A material testing program, similar to the one currently being carried out to qualify 316 SS for fast reactor core components, will have to be undertaken for the selected MHFFR blanket material. Since such a testing plan is needed, no matter what material is chosen, it was decided to consider other materials that might exhibit better properties at the higher temperatures and fluences.

TABLE 4.1-5
MHFFR BLANKET DATA

Blanket Rod Dimensions (mm)

Rod od	17.00
Clad thickness	0.175
Rod id	16.65
Pellet od	16.55
Annular hole	4.70
Rod pitch	17.85
Wire-wrap pitch	125.00

Rod Plenum

Pressure at assembly	83×10^4 Pa (120 psi)
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Temperatures (°C)

Inlet to module	280
Outlet from module	530
Maximum hot spot cladding temperature	797
Maximum hot spot fuel temperature	904
Maximum first wall temperature	468

Flow

Average Reynolds number in blanket	9300
Average velocity of flow, m/sec	19.8
Heat transfer coefficient, watts/cm ² °C	2180
	(384 Btu/hr-ft ² -°F)

Pressure Drop

In the blanket	1.24×10^4 Pa (1.8 psi)
Total module	3.9×10^4 Pa (5.7 psi)

Burnup

Peak	~1%
Average	~0.56%

A material with better high temperature properties would allow for a reduction in structural material in the blanket module and possibly improve the economy of the overall plant.

After a limited survey of some of the "super alloys", Inconel 718 was chosen as the reference material for both the cladding and structure of the MHFFR blanket. Three reasons lead to the selection of this material: (1) it has excellent high temperature properties (Figs. 4.2-1 and 4.2-2), (2) it is already a qualified bolting material in the high temperature code case, 1592, of Section III of the ASME Boiler and Pressure Vessel code, and (3) there is enough experience in the fabrication and welding of this material that no new processes are required.

4.2.2 Module Design

The design concept for the blanket design was specified by Lawrence Livermore Laboratory in Ref. 4-13. The MHFFR blanket design consists of 16 "orange peel" modules arranged into a spherical shell which lies within the superconducting magnet coils and surrounds the plasma. Each "orange peel", Fig. 4.2-3, is a separate pressure vessel consisting of a number of submodules which contain the blanket assemblies. Each submodule is 25 cm wide, 100 cm in depth with a variable length depending on the position of the submodule from the equator of the "orange peel". Using the 25 cm width for the submodules a first wall thickness of 0.5 cm was calculated. This was based on a primary coolant pressure of 6.08 MPa (60 atmospheres) and a first wall temperature of 500°C.

The 6.08 MPa (60 atmosphere) pressure required to improve the heat removal characteristics of the helium coolant causes the external walls of

Alloy 718 Sheet and Plate Annealed at 1950 F and Aged

Tensile Strength

Fig. 4.2-1. Alloy 718 tensile strength

.2% Yield Strength

Fig. 4.2-2. Alloy 718 0.2% yield strength

the module to experience very high loads. For this reason a reentrant coolant flow system was chosen for the module design. Such a system reduces the amount of structural material required per module by maintaining the external module walls near the inlet coolant temperature of 280°C.

4.2.2.1 Blanket Rod Mechanical Design. The blanket rod is shown in Fig. 4.2-4. As previously mentioned the material chosen for the blanket is Uranium Silicide (U_3Si) and for the cladding is Inconel 718. A cladding thickness of 0.175 mm was chosen based on simple pressure stress calculations. No creep-collapse analysis of the cladding was performed; instead the rod was pre-pressurized to 0.83 MPa (120 psi) (cold). A fuel to clad gap of 0.1 mm was included in the design. This clearance is adequate for the fabrication of the rod and also to accommodate the differential thermal expansion and swelling between clad and fuel during service without excessively straining the cladding (less than 1% total cladding strain at the end of life). A special feature of the fuel rod design is the thin, dished end plug. This design will allow adequate strength and sealing but is quite thin to avoid degrading the energy of the incident fusion neutrons.

4.2.2.2 Blanket Assembly Design. The blanket rods have been grouped into standardized fuel assemblies to ease the handling of the large number of fuel rods and to simplify the fabrication of the modules. Two different types of assemblies are required. First, a rectangular cross sectional assembly is needed to fill the central positions of the submodules, Fig. 4.2-5. Second, a trapezoidal assembly is required to fill in the edges of each submodule as much as possible (Fig. 4.2-6). This drawing,

which is a cross section through a typical submodule, shows the positioning of the two types of assemblies. The difference between a central submodule at the equator of the "orange peel" and an end submodule near the pole is the number of rectangular assemblies which can be accommodated by the submodule. A central submodule has 8 rectangular and 2 trapezoidal assemblies, whereas the smallest submodule at the pole has 1 central and 2 trapezoidal assemblies. This is shown in Figs. 4.2-6 and 4.2-7. Intermediate submodules would have 2 trapezoidal assemblies and between 1 and 8 rectangular assemblies.

Both assemblies are axially divided into two zones. A fast fissioning blanket zone at the end nearest the plasma and a tritium breeding zone. The design responsibility for this latter part has been left to Lawrence Livermore Laboratory and the discussion here will be centered on the fissioning blanket zone. To maximize the amount of blanket material in each assembly the blanket rods are set on a triangular pitch. This, however, causes some discontinuities at the edges of the rectangular assemblies. Shaping the inner box wall of these assemblies assures that the coolant flow is properly distributed throughout the assembly thus providing adequate cooling of all the rods.

The fuel rods are axially positioned within the assembly by cantilevering them from a base plate which is supported by tie-rods at the four corner positions. The spacing between the rods is maintained by means of a wire wrap spacer system. Because the rods are cantilevered, two wire wrap pitches were included in the relatively short 279 mm rod to give support over the entire length and avoid possible problems with the sagging or bowing of the blanket rods. A flow baffle is included at the inlet end of the assembly

SECT A-A SCALE: 2/1

ITEM		PART NO.	DESCRIPTION	MATL/MATL SPEC
PARTS LIST				
REQD/ASSEMBLY		MHFFR DESIGN		DIST.
<div> GENERAL ATOMIC COMPANY SAN DIEGO, CALIFORNIA</div>				
TITLE BLANKET ROD				
SIZE B	CODE IDENT. NO. 32334	DWG. NO. 3235-868-005		ISSUE
SCALE NONE		QAL		SHEET 1 OF 1

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
NO BURRS OR SHARP EDGES
DIMENSIONS PER ANSI-Y-14.5
TOLERANCE:
DECIMAL FRACTION ANGLE
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Fig. 4.2-4. Blanket rod
4-31

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TITLE			BLANKET SUBMODULE CROSS SECTION (CENTRAL SUBMODULE)	
SIZE			CODE IDENT NO.	DWG. NO.
D			32334	3235-868-006
SCALE: NONE			QAL	SHEET OF

Fig. 4.2-6. Blanket submodule cross section (central submodule)
4-35

REQD/ASSEMBLY		ITEM	PART NO.	DESCRIPTION	MATL/MATL SPEC
PARTS LIST					
MHFFR DESIGN				DIST.	GENERAL ATOMIC COMPANY <small>SAN DIEGO, CALIFORNIA</small>
UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES NO BURRS OR SHARP EDGES DIMENSIONS PER ANSI-Y-14.5 TOLERANCE: DECIMAL FRACTION ANGLE .XX ± ± ± .XX ± ± ± ALL MACHINED SURFACES DO NOT SCALE PRINT				PROPRIETARY INFORMATION THIS DOCUMENT IS THE PROPERTY OF GENERAL ATOMIC CO AND WILL BE RETURNED UPON REQUEST OR WHEN NO LONGER NEEDED BY THE RECIPIENT. INFORMATION CONTAINED HEREIN MAY NOT BE COMMUNICATED TO OTHERS NOR MAY THE DOCUMENT BE COPIED IN WHOLE OR IN PART WITHOUT THE EXPRESS CONSENT OF GENERAL ATOMIC CO.	
TITLE BLANKET SUBMODULE CROSS-SECTION (END SUBMODULE)				SIZE	ISSUE
CODE IDENT NO 32334				DWG. NO.	
SCALE NONE QAL				3235-868-004 SHEET 1 OF 1	

Fig. 4.2-7. Blanket submodule cross section (end submodule)
4-37

to assure that the entire first wall surface is cooled by the inlet helium; this will be explained in more detail in the next section.

4.2.2.3 Blanket Submodule Design. Figure 4.2-8 shows the design for the central submodule. The combination of rectangular and trapezoidal elements are bolted to flanges which are provided at the bottoms of the submodule's sidewall. These sidewalls which are extensions of the first wall support the blanket assemblies and also serve as the load carrying members from the first wall to the structural back wall.

Coolant gas enters through the inlet duct and completely fills the bottom plenum below the blanket assemblies. From this lower plenum the gas passes through the space provided between the submodule's side walls and the blanket assemblies. At the first wall the flow is reversed and directed into the blanket region by the flow baffle which is designed to assure that the entire surface of the first wall is cooled by the reversing flow. The coolant then passes through the fission zone and the tritium breeding zone and is collected in a chamber at the bottom of the assembly. From this chamber the coolant is discharged through two venturies into the main outlet duct. The venturies are included to limit the forces on the blanket in the event of a depressurization accident as explained in Chapter 7.

4.2.2.4 Blanket Module Design. The blanket module design is shown in Fig. 4.2-3. It is basically a collection of 45 submodules, similar to those described in the previous sections, arranged in the "orange peel" configuration. The adjacent submodules have a common sidewall thus eliminating pressure stresses on these large flat surfaces.

The thin first wall has been reduced to a series of pressure vessels, one for each submodule. Each of these vessels is cylindrically shaped to reduce the pressure stresses and tied to the structural backwall by means of the sidewalls and tie plates.

Special effort was made to insure that the module design is fabricable. All welds are external and require no special equipment. The construction of the module is done in a sequence to insure this fabricability. Construction begins at one end of the "orange peel" base plate after the helium outlet plenum has been installed but before the side covers are welded on. The helium outlet ducting for the first submodule is installed. The first submodule wall is welded into place and the blanket assemblies are bolted in place on the support structure. The other submodule wall is then welded on and the "gull-wing" shaped top piece that forms the first wall of the plasma chamber is welded on to seal the submodule. The procedure is then repeated for the next submodule: outlet ducting and support structure is installed, blanket assemblies are bolted into place, the wall is welded in and the gull-wing first wall segment is welded on to seal the submodule. This process continues on down the module until the other end is reached. The module base side covers are then welded into place to complete construction. Each of the welds is external at the time it is made. This means that standard welding techniques may be used. Further, each weld may be inspected before moving on to the next step in the construction process. These steps may be visualized by examining the direction of the welds on Fig. 4.2-8.

Sixteen modules are arranged to form a hollow spherical blanket structure within the Yin-Yang magnet coils. The fusion plasma region is

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	BLANKET SUBMODULE (CENTRAL)	
GENERAL ATOMIC COMPANY SAN DIEGO, CALIFORNIA		DRAWING NUMBER APR 78 3235868 008
SCALE 1/2		ISSUE

Fig. 4.2-8. Blanket submodule (central)

in the center of the spherical blanket cavity. The minor fusion reactor requires neutral beam injectors and direct convertors. Ports for these devices must be provided in the blanket structure. Detailed design of the modules to allow for beam ports and leakage ports has not been done. It is expected that these ports can be included at the edges of the modules without any major perturbation to the module design.

The coolant inlet and outlet ducts are shown coming out of the module approximately tangential to the equator. This was necessitated by the large sizes of the ducts (1.11 m and 1.22 m inlet and outlet ducts, respectively) and the space limitation at the rear of the module. It is felt, however, that this design is not optimized and will need some further work to determine the best configuration for the helium inlet and outlet arrangement. It is also felt that taking some extra pressure drop by necking the ducts to reasonable sizes until the magnet is cleared is probably a worthwhile design approach. This change should not greatly affect the performance of the plant yet it could simplify the ducting and module interface design.

As is discussed more fully in Chapter 5, several concepts are being considered for connection of the helium ducts to the module. The final choice will depend on the outcome of tradeoffs between the module design, the ducting design and the design of the Yin-Yang magnet and its support structure. This decision will receive further attention in the future.

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5. POWER CONVERSION LOOP DESIGN

5.1 PURPOSE AND SCOPE

The primary and secondary loop design for the mirror hybrid fission-fusion reactor (MHFFR) is presented in this section. The purpose of the studies performed was to develop a conceptual design for the primary/secondary loops using gas-cooled fission reactor technology. The present scope of work on the power conversion loop design included the following:

1. Selection of system pressure and temperature parameters for full load and accident conditions.
2. Design and performance evaluation at full load and accident conditions of
 - a. The primary loop.
 - b. The auxiliary loop.
 - c. The secondary loop.

Design improvements, alternate schemes and further studies for follow-on work are suggested at appropriate places in this section. This study includes primarily steady-state operation at full power and a suggested design basis accident case. Plant transient effects will be considered in future studies.

A conceptual layout of the primary and auxiliary loops is included in this section.

5.2 PRIMARY COOLANT SYSTEM

5.2.1 Reactor and Primary Loop Interface

A design concept of the MHFRR was developed in Ref. 5-1. A detailed description of the reactor operation and the heat removal from the fuel blanket and blanket walls was also given. The power conversion loop (PCL) receives heat from two sources; viz., from the blanket and from a direct converter attached to the reactor. Helium gas is used as the primary coolant in the reactor to transport the heat for power conversion. The reactor conceptual arrangement was further optimized in Ref. 5-2, and that configuration was used for this study, as shown in Fig. 5.2-1. Figure 5.2-2 schematically represents the power flow for the reactor.

The reactor blanket consists of sixteen (16) modules, each module resembling an "orange peel." Two pipe connections are provided at the back panel of each module for helium gas entry and return. Helium circulation inside the blanket fuel submodules was described in Chapter 4. The primary loop design discussed here covers aspects of the loops external to the blanket module.

5.2.2 Reactor Coolant System

The reactor coolant system consists of four independent primary loops and four independent auxiliary loops. The primary loops are used for normal power operation and for shutdown or depressurized cooling; the auxiliary loops may be used for reactor decay-heat removal following normal or emergency shutdown of the primary loops. The auxiliary loop design is discussed separately in Section 5.3.

Conceptual arrangement of a primary loop is shown in Fig. 5.2-3. The principal components of each primary loop are the steam generators, the

Fig. 5.2-1. Mirror hybrid fusion-fission reactor — Lawrence Livermore Laboratory design concept

Fig. 5.2-2. Mirror fusion-fission reactor - power flow diagram

Fig. 5.2-3. MHFFR - conceptual primary/auxiliary loop arrangement

main helium circulators and helium pipes and valves. A discussion of these components is given in greater detail in this report, including design bases, descriptions and, where applicable, design evaluations for the particular component or system involved.

5.2.3 Design Bases for Conceptual Study

The principal function of the reactor coolant system is to transfer heat from the blanket fuel submodules and blanket walls to the steam generators through the primary loop. The steam produced in the steam generators is used to drive the plant turbine generator system. Helium, the reactor coolant, is circulated at a flow rate, temperature and pressure consistent with the functions of the fission blanket, steam generators, and the helium circulators. Under full load normal operating conditions, the heat transferred from the reactor is approximately 4400 MW(t). The reactor coolant system is required to operate at full-load power and also under decay-heat removal conditions following a reactor shutdown. It will also be capable of confining any radioactive material and limit its accidental release within acceptable limits.

The pressure-retaining boundary of the reactor coolant system and its associated components could be either structural elements and/or a steel liner kept in compression by prestressed concrete. The structural supports considered for the present study are only conceptual. Actual recommendations would be made following an in-depth study, when a detailed configuration of the hybrid reactor machine with its support structure is available.

Since the current scope of work on the MHFFR is conceptual in nature, the design requirements considered for the reactor coolant system were primarily based on the normal steady-state performance and a postulated worst

accident condition caused by depressurization. The effects due to plant transients, such as part load operation, startup and normal shut down, were not considered in detail.

5.2.4 Primary Loop Conceptual Design

Each primary loop consists of four modules, which are interconnected by ducts. Each loop consists of four steam generators, one per module, and two helium turbocirculators arranged in two sub-loops as shown in Fig.

5.2-3.

The steam generator unit consists of two sections; viz., an economizer-evaporator-superheater (EES) section and a separate resuperheater section (RSH). Each steam generator unit is of monotube helical-coil design with upward boiling for stability. Hot helium enters at the top of the steam generator and passes downward in a single pass over the resuperheater and EES bundle.

Steam from the superheater tubesheet headers drives the single-stage axial flow compressor. Upon exiting the turbine, the steam reenters the steam generator at the resuperheater unit. The resuperheated steam is then used to drive the main power turbine.

At full power operation all four primary loops are in operation. In the event of failure of any one main loop, the fusion power plasma will be shut down although operation at 50% power or less could be continued using one sub-loop if desired. Each main loop has its own redundant operable part to continue to cool all the four modules following a reactor shutdown. Details of decay-heat removal procedures are discussed in Section 5.3.

The main helium circulator at full-load conditions provides a pressure rise of approximately 1.24×10^5 Pa (18 psi). The circulator outlet pressure

at the circulator diffuser is about 62×10^5 Pa (900 psi) at full power under pressurized conditions.

The approximate pressure losses around the helium circuit, based on the helium flow rates of the preliminary heat balance for normal full-load conditions are:

<u>Location</u>	<u>Pressure Loss Pa (psi)</u>
Blanket	3.97×10^4 (5.77)
Steam generator	
EES section	4.14×10^4 (6.00)
RSH section	1.38×10^4 (2.00)
Helium ducts, hot and cold including bends, expansion and contractions, valves	2.75×10^4 (4.00)
Total	12.24×10^4 (17.77)

5.2.5 Primary Loop Functional Evaluation

The primary loop not only provides the cooling during full-load normal operating conditions, but also provides the "first line of defense" cooling during shutdowns and following a depressurization accident. All four primary loops are independent and thus the isolation of one loop will not affect the performance of the other three loops and the modules attached to them. In particular, if one of the four primary loops failed by the occurrence of a steam generator tube leak or loss of helium pressure, the three unaffected loops would remain in operation efficiently cooling the modules connected to them. Thus, in the event the auxiliary cooling system (ACS) is required due to multiple failures in a primary loop, only the ACS for the affected loop will be required to operate.

The inherent redundancy contained in a primary loop design is shown in Fig. 5-3. Two steam generators and one turbocirculator form one subloop and thus each primary loop has two subloops. The occurrence of a leak from a steam generator tube from any one unit or the failure of one turbocirculator affects only one subloop.

The interconnecting pipes insure that the remaining operable circulator forming the second subloop could adequately cool all four modules attached to that loop at up to 50% of full power.

5.2.6 Helium Piping Design - Pressure Trade-Offs

Reference 5-1 suggested a low system pressure of 20 atmospheres in order to have a thin first wall for the submodule. A thin wall was preferred to increase the plutonium and tritium breeding in the blanket. Such a low system pressure, though neutronically desirable, poses problems with the power conversion loop design. Lower gas density and higher volumetric flow rate require large size ducts in the range of 1.72 m - 1.83 m (68 in. to 72 in.). Connections of such ducts to the back panel of each module present practical difficulties. In addition, the design and performance of the multistage (4-5 stage) turbocirculator needed for a pressure rise to inlet pressure ratio ($\Delta P/P$) of 0.08 and associated large flow rate would require a circulator development program. The pumping power is also quite high (500 MW).

For these reasons a higher system pressure was preferred and for the first iteration a pressure of 6.078 MPa (60 atmospheres) was selected with LLL concurrence. Duct sizes calculated with this high pressure were found to be reasonable [1.22 m (48 in.) diameter hot duct and 1.11 m (44 in.) diameter cold duct]. Hot helium ducts will be designed to code case 1592.

Duct interior surfaces may be coated with a tritium barrier, and outside surfaces will be insulated. The insulation may have a small gap kept under a light vacuum, if necessary, to contain any possible tritium release from the helium stream.

The primary loop layout scheme is significantly dependent upon the spatial arrangement of the "Yin-Yang" magnets and modules (see Loop Layout, Section 5.6). Structural massiveness and interferences make interconnection of all 16 modules difficult. Due to symmetry, four quadrants were used, each composed of four modules integrated with a compact ducting arrangement. In the present conceptual design, it is considered to be feasible to reinforce and support the helium ducts, and also the steam generator units and circulators by means of structural elements. Such a scheme results in easing detachment of the ducts and removal of the module during refueling.

Details concerning a viable method of supporting the primary loop components will be considered in the next phase of the work. A PCRV type of support structure, either to totally enclose the reactor or partly contain the major loop components, has several desirable features and may be feasible but this needs to be studied further.

It is expected that a full double-ended offset rupture of the helium ducts will have to be guarded against. If structural elements are used for supports, they reinforce the ducts both longitudinally and circumferentially with suitable spans such that any possible rupture area will be constrained within allowable limits.

In order to keep the parasitic pressure drop to a minimum in the ducts, a dynamic head of 3447 Pa (1/2 psi) was selected. The net pressure drop in the primary helium ducts is estimated to be approximately 27,576 Pa (4 psi),

accounting for bends, contraction/expansion losses, valves and Tee-connections. Table 5.2-1 shows the principal design data for the preliminary power conversion loop design.

5.2.7 Main Helium Circulator

5.2.7.1 Design Bases. The function of the main circulators is to provide helium flow through the reactor. Each of the four main loops is equipped with two main circulators. Each circulator unit consists of a single-stage axial flow compressor and a single-stage steam turbine driven by superheated steam before it is reheated and passed to the main turbine. These units are similar to those designed for GCFR-HTGR plants.

5.2.7.2 Operating Requirements. The circulators will be designed to operate under various normal and abnormal conditions, including (1) normal plant operation at full load, (2) plant startup, (3) routine plant shutdown for refueling or maintenance, and (4) postulated accidents, including the design basis depressurization accident. The circulator operating parameters for pressurized conditions are given in Table 5.2-2.

5.2.7.3 Design Requirements. The helium circulators will be designed to meet the requirements shown in Table 5.2-2 under 100% load condition and will be capable of operating without surge or surge instability at all operating conditions.

Normal circulator control will be achieved by means of an inlet throttle valve. A steam bypass will be provided to obtain balance between the main loops. The circulator will be provided with a service system to supply high-pressure water for lubricating the bearings and to supply

TABLE 5.2-1
POWER CONVERSION LOOP PRINCIPAL DESIGN DATA

<u>Power data</u>	
Net PCL output power, MW(e)	1573.6
Net power input to PCL, MW(t)	4381.0
Power input from blanket, MW(t)	3919.5
Power input from direct converter, MW(t)	461.5
Helium circulator pumping power, MW(t)	105.5
PCL efficiency, %	35.92
Net station output power, MW(e)	888.9
Total nuclear power - fusion plus fission, MW(t)	4014.6
Net station efficiency, %	22.14
<u>Primary coolant (helium) data</u>	
Average pressure, Pa (atm)	60.78×10^5 (60)
$\Delta P/P$, %	2.04
Total flow rate, kg/s (lbm/hr)	3401 (27.02×10^6)
Inlet temperature at steam generator, °C (°F)	530 (986)
Outlet temperature at steam generator, °C (°F)	275 (527)
Temperature rise through circulator, °C (°F)	7.2 (13)
<u>Secondary coolant (steam/water)</u>	
Feedwater inlet pressure, Pa (psia)	10×10^6 (1460)
Net flow rate, kg/s (lbm/hr)	1611.5 (12.8×10^6)
Superheater outlet pressure, Pa (psia)	8.272 (1200.0)
Resuperheater inlet pressure, Pa (psia)	5.61×10^6 (814.0)
Resuperheater outlet pressure, Pa (psia)	5.36×10^6 (778.0)
Feedwater inlet temperature, °C, (°F)	171 (339)
Resuperheater outlet temperature, °C, (°F)	504 (939)
Net pressure drop in steam generator, Pa (psid)	2.11×10^6 (307)

TABLE 5.2-2
CIRCULATOR OPERATING PARAMETERS (PRELIMINARY) AT FULL POWER

<u>Compressor type</u>	Single stage
He flow rate, kg/s (lb/sec)	425 (937.5)
He inlet pressure, Pa (psia)	6.08×10^6 (882.0)
He inlet temperature, °C (°F)	277 (530)
He pressure rise, Pa (psi)	1.22×10^5 (17.7)
He outlet pressure, Pa (psia)	8.62×10^6 (899.7)
Speed, rpm	6100
Blade tip diameter, m (in.)	1.22 (48)
<u>Turbine type</u>	Single stage
Steam flow rate, kg/s (lb/sec)	201 (444.0)
Steam inlet temperature, °C, (°F)	438 (821)
Steam inlet pressure, Pa (psia)	7.445×10^6 (1080.0)
Steam outlet pressure, Pa (psia)	5.772×10^6 (830.0)
Speed, rpm	6100

purified buffer helium for preventing inleakage of bearing water into the reactor coolant loop or outleakage of the helium coolant.

5.2.7.4 Design and Operation Description. The steam turbine for each main helium circulator is located between the superheater outlet and the resuperheater inlet. Thus the full flow of steam generated in the plant drives the circulator turbines before passing to the main steam turbine. In this flow arrangement, the steam-driven helium circulators tend to be inherently self-regulating since the helium cooling requirements of the reactor are directly related to the supply of steam produced. Analysis performed during this study indicates that the best location for the circulator turbine is between the superheater and the resuperheater. Elimination of the resuperheater reduces the plant efficiency by about 2 points (36% to 34) and therefore affects the plant economics. The principal design data and performance characteristics of the main helium circulator and its steam turbine are given in Table 5.2-2.

5.2.8 Steam Generators

5.2.8.1 Design Bases.

5.2.8.1.1 Functional Requirements. The purpose of the steam generator is to transfer the heat from the reactor coolant to the secondary coolant.

High-temperature [530°C (986°F)] helium from the blanket and direct converter enters the steam generators and gives up its heat to the resuperheater section and main steam section to produce superheated steam at 444°C (830°F) and resuperheated steam at 505°C (940°F).

5.2.8.1.2 Design and Operating Requirements. The steam generators will be designed for continuous operation and also for conditions during reactor shutdown when the main loops are used for decay-heat removal.

The steam generators will be designed to accommodate, without operational impairment, all normal and emergency operating conditions. The design bases are, in general, similar to those for HTGR-GCFR plants and include the following.

Allowance will be made in sizing the steam generator to permit full-load operation with a certain number of tubes plugged in addition to a certain degree of internal tube fouling.

The allowances on the heat-transfer bundle tubing will include provision for imbalances expected in the circuit flow due to tube plugging, gas-flow imbalances, fabrication tolerances, heat-transfer-coefficient uncertainties, and friction-factor uncertainties.

The steam generators and their components will be designed so that no damage or malfunction is caused by internally generated vibrations, flow-induced vibrations, or environmental vibrations. The design, material, fabrication and inspection of the steam generators will be in accordance with applicable ASME codes.

5.2.8.2 Description and Operation. Each primary loop consists of four steam generators and the plant has sixteen (16) in total. Each steam generator module is about 3.2 m (10 ft 6 in.) in diameter and approximately 14.63 m (48 ft) high. The module is an axial-flow heat exchanger that is made up of two separate helically wound tube bundles; these are the resuperheater tube bundle and the main steam-generator tube bundle, which consists of the combined economizer, evaporator, and superheater sections. In the

steam-generator main bundle, the incoming high-pressure feedwater is converted into superheated steam that is then used to drive the main helium-circulator series turbine. The steam is subsequently returned to the steam generator and resuperheated before it flows to the main turbine-generator set of the power plant. The water-steam flow on the tube side is upward; the helium flows downward across the tube bundles. The major steam-generator design parameters are summarized in Table 5.2-3.

Helium coolant gas carries the reactor heat via ducts from the blanket and direct converter to the steam-generator. The hot gas flows downward over the resuperheater, superheater, evaporator, and economizer tube bundles. After passing through the steam-generator bundle, the cooled gas then passes into the main circulator inlet. Gas leaving the circulator passes through the diffuser, and reenters the blanket and direct converter, completing the main loop circuit.

The feedwater for each steam-generator module enters through feedwater-tubesheets and then through lead-in tubes connecting the tubesheets to the bottom of the economizer section.

The economizer and evaporator sections have 0.02 m (0.75-in.)-diameter tubes, whereas the superheater section of the steam-generator tube bundle has 0.0254 m (1.0-in.)-diameter tubes. Variation in the tube diameter in the economizer, evaporator, and superheater sections will provide pressure drop in the system and, if necessary, orifices will be installed in each feedwater tube to assure stability at low flow. Tube plugging to isolate defective tubes is done at the tubesheets, which are readily accessible.

TABLE 5.2-3(a)
STEAM-GENERATOR DESIGN DATA IN SI UNITS
(Data are for each unit)

	Resuperheater Section	Steam-Generator Sections		
		Superheater	Evaporator	Economizer
Steam generator diameter, m	3.2	3.2	3.2	3.2
Heat duty, W	24.8×10^6	86.3×10^6	51.1×10^6	117×10^6
Logarithmic mean temperature difference, °C	55.8	91.1	98.2	88.5
Overall heat-transfer coefficient, W/m ² -deg-K	1110	1691	1959	2022.6
Heat-transfer surface area, m ²	461	471.2	334	1330
Helium side				
Flow rate, kg/s	207	207	207	207
Inlet temp., °C	530	508	428	382
Outlet temp., °C	507	428	382	275
Average pressure, MPa	6.02	6	5.99	5.98
Pressure drop, MPa	0.013	0.016	0.005	0.019
Water-steam side				
Flow rate, kg/s	100	100	100	100
Inlet temp., °C	403	308	302	171
Outlet temp., °C	505	445	308	302
Average pressure, MPa	5.6	8.62	9.26	10.06
Pressure drop, MPa	0.103	0.41	0.852	0.747
Tubes				
Number of tubes	510	221	221	221
Tube size, m	0.0254×0.0254	0.0254×0.0018	0.019×0.0016	0.019×0.0016
Tube bundle height, m	2.16	2.07	1.62	6.34

TABLE 5.2-3(b)
STEAM-GENERATOR DESIGN DATA IN U.S. UNITS
(Data are for each unit)

	Resuperheater Section	Steam-Generator Sections		
		Superheater	Evaporator	Economizer
Steam generator diameter, ft	10.5	10.5	10.5	10.5
Heat duty, Btu/hr	84.66×10^6	294.5×10^6	174.4×10^6	399.2×10^6
Logarithmic mean temperature differ- ence, °F	100.37	164	176.7	159.25
Overall heat-transfer coefficient, Btu/hr- ft ² -°F	195.41	297.74	344.9	356.2
Heat-transfer surface area, ft ²	4960	5072	3596	14,315
Helium side				
Flow rate, lb/hr	1.65×10^6	1.65×10^6	1.65×10^6	1.65×10^6
Inlet temp., °F	986	945	802	719
Outlet temp., °F	944.7	802	719	526.6
Average pressure, psia	873	871	869	867
Pressure drop, psi	2.0	2.39	0.8	2.8
Water-steam side				
Flow rate, lb/hr	0.799×10^6	0.799×10^6	0.799×10^6	0.799×10^6
Inlet temp., °F	756	587	575	340
Outlet temp., °F	940	831.5	587	575
Average pressure, psia	813	1251	1343.3	1460
Pressure drop, psi	15	59.41	123.7	108.55
Tubes				
Number of tubes	510	221	221	221
Tube size, in.	1.0 x 0.10	1.0 x 0.070	0.75 x 0.062	0.75 x 0.062
Tube bundle, height, ft	7.1	6.8	5.3	20.8

5.3 AUXILIARY COOLING SYSTEM

5.3.1 Design Bases: Functional Requirements

The auxiliary cooling system provides an independent means of cooling the shutdown reactor and removing the decay heat produced by the blanket. The system will be used (1) when the main cooling loops are not available, or (2) when it is desired not to perform decay-heat removal with the main cooling loops. The auxiliary cooling system must provide adequate cooling and prevent the temperatures of the fuel, the cladding, and the reactor coolant from exceeding prescribed limits so that safe cool-down of the reactor and the reactor cooling system is ensured after any credible system failure.

The auxiliary cooling system is designed to perform the engineered safety function of providing adequate emergency reactor cooling after shutdown as an independent backup to the main cooling loops. Figure 5.2-3 schematically shows the conceptual arrangement of the auxiliary cooling system (ACS).

Each primary loop is provided with an independent auxiliary cooling loop resulting in four separate auxiliary loops for the entire reactor. Such an arrangement has two significant advantages: first, in the event of a depressurization accident in one of the four loops and multiple failures in the steam-generators/circulators of that loop, only the auxiliary loop associated with the affected loop will be required to operate. The offsite or diesel power requirements will then be quite modest. Second, the practical difficulties arising from large piping configurations with complicated routing required for integration of all four quadrants are avoided.

Efficient cooling will be available for the rest of the reactor with the three unaffected loops under pressurized conditions.

Each auxiliary loop consists of two heat exchangers and two circulators. The loop circuitry is so arranged that if one circulator or one heat exchanger failed, the remaining operable units would provide adequate cooling.

The auxiliary loops will be designed to provide cooling in either a pressurized or a depressurized condition.

In addition to the engineered safety function, the auxiliary cooling loops provide decay-heat removal and cooling during refueling. For these uses of the auxiliary cooling loops, the transfer from the main cooling loops to the auxiliary loops will be done sufficiently long after shutdown to minimize the auxiliary power requirements. Long-term cooling by the auxiliary core-cooling system is described in Section 5.8.

In addition to the above functions, the auxiliary cooling system must also be capable of providing safe cooldown of the reactor with a mixture of gases in the reactor coolant system, such as could result from accidents involving steam or air inleakage.

5.3.2 Design Criteria

To ensure that the auxiliary cooling system adequately meets the performance objectives under all normal operating or credible combinations of accident conditions, the general design criteria for the system will be in accordance with the criteria presented for the reactor coolant system and with the specific requirements applicable to the essential system components; viz., the auxiliary circulator and the auxiliary heat exchanger.

The auxiliary loop design performance discussed in this report does not include effects imposed by plant transients. The transient conditions and effects will be considered in the future.

5.3.3 Description

The auxiliary cooling system consists of four separate and independent cooling loops (see Fig. 5.2-3). Each loop contains two auxiliary heat exchangers and two auxiliary circulators, and a cooling-water system for supplying the proper quantity of pressurized cooling water to the heat exchangers. The auxiliary cooling system is designed as a Class I system and as an engineered safeguard. It will be capable of cooling the reactor with the reactor coolant system either pressurized or depressurized and with one of the heat-exchanger/circulator systems out of service. Each auxiliary circulator is electrically driven by a motor supplied by an independent power supply.

The auxiliary heat exchangers employ helically coiled tubes with a single-pass helium cross-flow passing over the tubes. The water side of the heat exchanger is divided into parallel subsections.

In the external part of each auxiliary cooling system there are two circulating pumps in parallel. One is a low-capacity pump that is used to circulate water through the heat exchangers during normal station operation when the auxiliary circulators are not in operation. The larger pump is used to provide the higher flow rates required for auxiliary cooling. Pumping power is available from multiple sources, including the essential power system. The closed-loop cooling system rejects heat to the atmosphere through an air-cooled heat exchanger.

Operation of the heat exchanger will be continuous at full water-side temperature differential [82°C (180°F) inlet, 204°C (400°F) outlet] except for two conditions: (1) in the depressurized mode, the heat exchanger outlet temperature will be held at 260°C (500°F) and (2) during the latter stages of decay-heat removal or during refueling, the outlet temperature will decrease because of decreased heat-removal requirements. Operation of the auxiliary circulator service system will be continuous during plant lifetime, but at a reduced heat load during the periods in which auxiliary circulator operation is not required.

The following principles govern actions upon startup of an auxiliary core-cooling loop:

1. The auxiliary circulator drive motor is energized to produce a flow corresponding to that of the reactor coolant, and establish coolant flow through the blanket and the auxiliary heat exchanger.
2. Cooling-water flow in the auxiliary heat-dump system is switched to the large-capacity circulating pump by administrative controls.
3. The power supply, and thus the auxiliary circulator speed, is automatically adjusted until the set point of the helium temperature at the blanket inlet is achieved. This provides for the increase in speed needed for cooling when the reactor is depressurized or for the decrease in speed that should accompany repressurization.

The auxiliary loops will be designed and instrumented to facilitate periodic testing of operability while the plant is on load. The heat exchanger and related heat-removal system will be in continuous service with circulation of cooling water.

5.4 AUXILIARY CIRCULATORS

5.4.1 Design Bases

5.4.1.1 Functional Requirements. The auxiliary circulators are part of the auxiliary cooling system and provide helium flow for decay-heat removal. Two circulators are provided for each primary loop and will supply the necessary flow in the most severe operating condition. Each auxiliary circulator unit will consist of an electric-motor-driven centrifugal compressor and diffuser. The auxiliary circulators are to be operable at all pressure levels from full helium inventory down to refueling (depressurized) status.

The normal use of the auxiliary circulators will be during the following conditions:

1. Plant shutdown.
2. Refueling (approximately atmospheric pressure).
3. Functional checkout with the reactor at full load.

The emergency-cooling mode of auxiliary circulator operation occurs in the event of trouble in any one main loop. Because of the wide range of required operating conditions, the electric-motor drives will be capable of variable speeds and each motor will be operated by an independent control system and be supplied from an essential power supply.

5.4.1.2 Performance Requirements. The design of each auxiliary loop is based on a heat-removal capacity of 5% of the normal thermal reactor power. The helium-side design conditions, required for the emergency cooling mode of operation, are for a 1.52×10^5 Pa (1.5 atm) pressure and blanket inlet

and outlet temperatures of 100°C (212°F) and 500°C (932°F) respectively.

These conditions require the following circulator characteristics.

For each auxiliary loop

Mass flow rate	23.44 kg/s	186,212 lb/hr
Circulator outlet pressure	1.714×10^5 Pa	24.87 psia
Compressor pressure rise	0.193×10^5 Pa	≈2.8 psia
Compressor inlet temperature	100°C	212°F

5.4.1.3 General Design Requirements. Equipment, piping, and the electrical wiring associated with the operation of the auxiliary circulators will be designed to operate under environmental conditions resulting from the design basis depressurization accident.

The auxiliary circulator will be capable of startup or shutdown against normal system pressure and temperature and will have a provision for functional checkout with the main circulators operating.

The normal speed control of the circulator will be by means of the variable-frequency power available to the drive motor. The control system and power supply for each circulator will be independent of the others. Each circulator will be driven by its own power supply and will be controlled through its own control system. The motors will be capable of restarting following any voltage interruption.

The single-failure criterion will be met in that failure in any one control system or power supply will not cause a failure or inhibit operation of the other auxiliary cooling subloop.

5.4.2 Description

The design of the auxiliary circulator is conceptual and only a preliminary design has been made to date.

The preliminary design data for the auxiliary helium circulator is summarized in Table 5.4-1.

The speed required for the auxiliary circulator is a strong function of the primary circuit pressure; quite low speed is sufficient with the reactor coolant pressurized, whereas high speed is required in the depressurized condition. Each circulator is driven by a squirrel-cage induction motor supplied with variable-frequency power from independent essential power supplies for each machine.

5.4.3 Design Evaluation

An electric-motor drive is provided for the auxiliary circulators as a different motive source from the steam turbine drive used on the circulators for the main loops. The design of the auxiliary circulators is based on well-established technology. Due to its fundamental simplicity, the electric motor-driven centrifugal compressor of moderate size and power poses no particular new problems or development requirements.

Two circulators are provided where one is adequate to meet performance requirements in the event of a depressurization accident. Each circulator is self-contained and failure of one cannot cause failure in the other. The service systems will be based on the modular concept for each unit for essential services.

Electric power for the motors may be obtained from either the normal or the emergency power sources. Availability of the auxiliary circulators can be assured in the time available in the event of loss of a main loop.

TABLE 5.4-1
AUXILIARY HELIUM CIRCULATOR PRELIMINARY DESIGN DATA

Type	Centrifugal
Drive.	Electric motor
Fluid.	Helium
Speed, rpm	5000
Inlet temperature, °C (°F)	100 (212)
Inlet pressure, Pa (psia).	1.52×10^5 (22.07)
Outlet pressure, Pa (psia)	1.71×10^5 (24.87)
Mass flow, kg/s (lb/sec)	23.44 (51.72) per circulator
Efficiency, %.	82
Power, MW (hp)	2.75 (3700)

5.5 AUXILIARY HEAT EXCHANGERS

5.5.1 Design Bases

5.5.1.1 Functional Requirements. The purpose of the auxiliary heat exchangers is to remove heat from the reactor helium coolant when the main cooling loops are shut down and unavailable for this purpose. Helium flows through the heat exchanger, where heat is transferred to the cooling water inside the tubes. Cooled helium leaving the top of the heat exchanger enters the auxiliary circulator and is circulated through the blanket modules to repeat the cycle.

5.5.1.2 Design and Operating Requirements. The auxiliary heat exchangers will be designed to provide cooling for both pressurized and depressurized conditions. Each heat exchanger will have a heat-removal capacity of 5% of the normal reactor power. For the depressurized design case, the required auxiliary loop will provide adequate cooling following a reactor scram. At present, transfer of reactor cooling to the auxiliary loop (if ACS service is required), is expected in about 85 seconds, worst case following plasma shutdown, based on GCFR plant practice. Within a few seconds or almost instantaneously after the reactor shutdown the auxiliary circulator could be brought to speed if there is not a loss of offsite power. The parameters for the depressurized design conditions are summarized in Table 5.5-1; they are based on the operation of one auxiliary loop.

Boiling in the auxiliary heat exchangers will be prevented by maintaining the water under a pressure sufficient to assure approximately 28°C (50°F) subcooling at the maximum expected operation temperature.

TABLE 5.5-1
DESIGN DATA FOR AN AUXILIARY HEAT-EXCHANGER

General

Helium frontal flow area, m ² (ft ²)	3.48 (37.5)
Heat duty, MW (Btu/hr)	48.71 x 10 ⁶ (166.39 x 10 ⁶)
Logarithmic mean temperature difference, °C (°F)	68 (154.52)
Overall heat transfer coefficient, W/m ² -K (Btu/hr-ft ² -°F)	549 (96.7)
Heat transfer surface area, m ² (ft ²)	1034.5 (11,136)

Helium side

Flow rate, kg/s (lb/hr)	23.44 (186,212)
Inlet temperature, °C (°F)	500 (932)
Outlet temperature, °C (°F)	100 (212)
Average pressure, Pa (psia)	1.52 x 10 ⁵ (22.07)
Pressure drop, Pa (psi)	13,788 (2.0)

Water side

Flow rate, kg/s (lb/hr)	62.63 (497,500)
Inlet temperature, K (°F)	355 (180)
Outlet temperature, K (°F)	533 (500)
Average pressure, Pa (psia)	14.48 x 10 ⁶ (2,100)
Pressure drop, Pa (psi)	2.068 x 10 ⁵ (30)

Tubes

Tube size, m (in.)	0.019 x 2.1 x 10 ⁻³ (0.75 x 0.083)
Longitudinal tube pitch, m (in.)	0.025 (1.0)
Transverse tube pitch, m (in.)	0.031 (1.246)
Tube bundle height, m (ft)	3.048 (10)

Safety relief valves will be provided for the auxiliary heat exchangers to protect them from overpressurization when an exchanger is isolated on the water side.

Since the depressurized condition limits the design of the auxiliary heat exchanger, the pressurized condition performance was assumed to be satisfactory.

Other considerations that govern the design of the heat exchangers are discussed below.

The tubes in the heat-exchanger tube bundle will be segregated into subgroups, each with approximately the same number of tubes. Separate cooling-water supply (and return) lines will connect each of these tube subgroups to the cooling-water headers. The arrangement of the tubes will be such as to minimize the effect of inleakage from a tube failure.

The heat exchangers will be designed so that they will not be damaged or caused to malfunction either by internally generated vibrations, such as flow-induced vibrations, or by environmental vibrations.

The design, fabrication, inspection, and materials specified for the component parts of the auxiliary heat exchangers shall be in accordance with the requirements of the appropriate approved codes where applicable.

5.5.2 Description

The auxiliary-loop coolant is pressurized water. Each heat exchanger is a helically wound, axial-flow tube bundle with an integral shroud. The tube bundle is approximately 2.43 m (8 ft) o.d. and 3.05 m (10 ft) high and is made of carbon or low-alloy steel. Preliminary design data for the auxiliary heat exchanger are summarized in Table 5.5-1.

The helium flow path from the blanket through the auxiliary cooling loop is similar to that through the main cooling loop. During normal reactor plant operation, there is a small bleed of helium from the blanket through the closed auxiliary loop. A small flow of cooling water through the heat exchangers maintains the auxiliary cooling loops at close to the cold helium temperature. This reduces thermal transients on loop startup and minimizes parasitic heat losses. During use for shutdown cooling (depressurized), the design inlet water temperature is 180°F and the outlet water temperature to the heat-dump system is 260°C (500°F).

5.5.3 Design Evaluation

The auxiliary heat exchanger is conventional as to design, material, and duty. Dynamic flow instabilities are not anticipated, since no boiling occurs. The performance of the heat exchanger under transient conditions will be evaluated later.

5.6 GENERAL EVALUATION OF PRIMARY/AUXILIARY SYSTEM DESIGN

The design evaluation of the various components making up the several primary/auxiliary cooling loops has been covered in previous sections. The objective of this section, therefore, is to discuss the reactor coolant system as a whole and its capabilities during normal and abnormal operation of the plant with regard to its system function of transferring heat from the reactor to the secondary (water, steam) coolant system.

5.6.1 Safety Factors

5.6.1.1 Cooling Loops and Helium Circulators. Normal full-load operation of the plant is accomplished with all circulators operating at or near rated speed. If a failure of one of the two main circulators of a primary loop

should occur, this will lead to automatic power shutdown of the reactor. Normal decay-heat removal is continued as demanded by the system.

In order to extend normal main-loop cooling after shutdown, a set of bypass lines and valves provides a means of bypassing circulator turbine discharge steam flow around the resuperheater and passing it to the condenser, thus making more heat transfer possible from the helium into the main sections of the steam generators. With this system, steam production is maintained at a level sufficiently high to keep the circulators operating, and the rate of utilization of reactor heat can be controlled, so that there is sufficient time to bring auxiliary boilers into service to provide the circulator steam supply necessary for long-term decay-heat removal by the main loops.

To improve the reliability of shutdown cooling, four completely independent auxiliary cooling loops, each with two separate helium circulators and two heat exchangers have been provided. The design and operation of these auxiliary loops was described in a previous section.

5.6.1.2 Steam Generators and Water Supply. The steam generators provide the normal means of heat removal from the reactor coolant both during plant operation and during shutdown periods. Redundancy is provided by the design of the cooling loop, each having two subloops and each subloop with its own two steam generators. The heat-removal capability of a steam generator loop is a function of the amount of feedwater and steam flow in the loop; the plant control system is designed to regulate the amount of reactor coolant flow and reactor power to match the feedwater flow and to maintain rated steam temperatures from the operating steam generators.

5.7 SECONDARY COOLANT AND POWER CONVERSION SYSTEM

5.7.1 Steam-Cycle

The reference steam-cycle for full-load operation, which is the basis of the design described in this report, is shown in Fig. 5.7-1. The major equipment in the cycle consists of sixteen once-through steam generators, including resuperheating sections, the turbine generator, steam-turbine drives for the helium circulators, condensers, feedpumps, feedwater heaters, and associated piping.

Feedwater enters the economizer section of the steam generators at a net total flow rate of 1611 kg/s (12.8 million lb/hr) at 10.16×10^6 Pa (1475 psia) and 171°C (340°F) and is converted to superheated steam at 8.27×10^6 Pa (1200 psi) and 443°C (830°F). The steam from each steam generator is directed to its helium-circulator series steam turbine, where it leaves at 5.71×10^6 Pa (830 psia) and 403°C (757°F). To reduce the moisture content in the last stages of the main turbine, the steam is reheated to 505°C (940°F) at 5.48×10^6 Pa (795 psia) and then it proceeds to the high-pressure turbine. The extraction steam for the high-pressure feedwater heater is taken from the crossover between the high-pressure and intermediate-pressure turbines and is used to heat the feedwater to its final temperature of 171°C (340°F). Extraction steam at approximately 6.2×10^5 Pa (90 psia) drives the boiler feedpumps. Additional extractions are made from the low-pressure turbine and are directed to the low pressure feedwater heaters. The remaining steam flow is exhausted to the main turbine condenser. Condensate from the hot well of the main condenser is pumped at low pressure through three stages of feedwater heating.

Fig. 5.7-1. Secondary coolant (steam) loop arrangement

Gross electrical output of this plant is 1573.6 MW. Net heat rate is 10×10^6 J/kWh (9482 Btu/kWh), which corresponds to a power conversion system thermal efficiency of 35.92. The heat balance was based on a helium-loop design pressure drop of 1.22×10^5 Pa (17.7 psi), circulator and circulator turbine overall efficiency of 80%, and reactor coolant inlet and outlet temperatures of 282°C (540°F) and 500°C (932°F) respectively. The steam is reheated after it passes through the series circulator turbines, thereby providing throttle conditions of 5.13×10^6 Pa (744 psia) and 503°C (937°F), which are acceptable to conventional nonreheat steam turbines. The moisture content of the exhaust steam is low.

5.7.2 System Design

5.7.2.1 Turbine Generator. The turbine selected is a tandem-compound machine comprising a high-pressure section, an intermediate-pressure section, and an eight-flow low-pressure section exhausting downward to the condenser. Inlet steam conditions are 5.13 MPa (744 psia) and 503°C (937°F). The turbine generator unit has a design rating of 1576 MW with the throttle valve wide open, corresponding to a backpressure of 3 in. Hg absolute.

The entire turbine is steam-sealed and a bypass arrangement from the main steam line to the main condenser is provided for startup and shutdown.

5.7.2.2 Condenser and Circulating-Water System. The main steam condenser will be capable of rejecting heat to the circulating-water system from the main turbine, low-pressure heater drains, and the auxiliary condenser. In addition, during startup and plant upset conditions, the main condenser will provide the heat sink for secondary coolant from the flash tank.

An auxiliary condenser provides a heat sink for the boiler feedpump turbines and the circulator bearing-water pump turbines. Water from this condenser is returned to the condensate system through the main condenser.

5.7.2.3 Startup and Shutdown Auxiliaries. Details and sequential procedures concerning the plant startup and shutdown controls are discussed in Section 6. The following summarizes the principal operations.

Each main cooling loop has its own auxiliary boiler to provide steam for the circulator turbine and circulator bearing-water-pump turbine during initial startup and during decay-heat-removal operations. Each boiler will be sized to provide the steam required for the operation with the reactor coolant system depressurized.

An auxiliary boiler feedpump will be provided for each auxiliary boiler to supply feedwater to it from the emergency water supply system. These pumps will be electrically driven and sized to accommodate the auxiliary boilers.

During startup, shutdown, and decay-heat-removal operations, the feedwater flow to the steam generators is provided by electrically driven shutdown boiler feedpumps. These pumps supply high-pressure feedwater to the steam generators from the emergency water supply system. They will be sized to maintain sufficient water level in the steam generators and to supply feedwater to the desuperheaters during shutdown and decay-heat-removal operations.

The bypass line from the circulators to the main condenser will also contain a flash tank that will supply a source of steam, which can be used to drive the main boiler feedpumps when extraction steam is not available from the main turbine and can also be used for feedwater heating. During

normal operation, the flash tank will be pressurized with extraction steam from the main steam turbine.

5.7.3 System Operation

5.7.3.1 Normal Power Operation. During normal operation there are eight steam loops operating in parallel, each consisting of two resuperheaters, two circulator turbines, two EES sections, and high-pressure heaters connected to a feedpump header. The resuperheater outlets are joined in a common main steam pipe leading to the high-pressure turbine. Steam is extracted in each loop prior to the circulator turbines to drive the circulator bearing-water pump turbines and is exhausted to the auxiliary condenser along with steam from the main boiler feedpump turbines. Condensate along with steam from the main boiler feedpump turbines. Condensate from the auxiliary condenser flows to the main condenser and is recycled with the main condensate flow.

5.7.3.2 Normal Shutdown and Reactor Trip. The control features concerning the normal shutdown are covered separately in Section 6. Normal shutdown or reactor trip refers to the plant shutdown operations required to get the plant load decreased to decay-heat level. Removal of decay heat is described in Section 5.7.3.3.

For a programmed shutdown, the power level will be decreased gradually down to a minimum load with the main turbine on line. At this load, the main turbine will be tripped and the reactor power further decreased. The main steam will be bypassed to the desuperheater and flash tank, where steam will be extracted to continue to drive the main boiler feedpumps until the load decreases to about 6%. At that time, the shutdown boiler

feedpumps will be started and the main boiler feedpumps shut down. The load will then continue to be decreased to the decay-heat level. During a reactor trip, the intermediate step of driving the main feedpumps with steam from the flash tank may be optional.

The shutdown boiler feedpumps supply feedwater for low flows during all shutdown operation. The inventory of the steam generators is sufficiently large to supply steam to the main circulators during startup of these feedpumps or during temporary loss of electric power to the pumps without significantly degrading the steam conditions.

During normal shutdown or reactor trip with all equipment available, the bypassed steam will be desuperheated and sent to the main condenser via the flash tank. The water from the condenser is then pumped and heated with steam from the flash tank. It then flows to the shutdown boiler feedpumps, completing the heat-sink portion of the cycle. The principal safety-related characteristics of this system are that

1. The transition from normal operation to shutdown requires only a few, simple control actions (see Plant Controls section).
2. The system can operate for sufficient time after loss of any single component to initiate the decay-heat removal phase with either main or auxiliary loops.
3. The water inventory in the steam generators is sufficient so that the shutdown boiler feedpumps do not have to be started rapidly and could be interrupted to allow changeover to emergency electric power supply.

5.7.3.3 Decay-Heat-Removal Operation. The control features governing this operation are given separately in Section 6. The shutdown procedures

described in Section 5.7.3.2 will be capable of providing adequate cooling as the plant load is reduced to about 2%. Thereafter, additional operating procedures will be required to continue the use of the main circulators, or a switch to the auxiliary loops will be made.

As the decay-heat decreases further, the quantity of steam produced by the steam generator will be insufficient to supply the circulators with enough power to maintain blanket flow stability. In order to maintain the proper steam flow to the main circulator turbines, auxiliary steam will be supplied by auxiliary boilers. The steam from the steam generators will be bypassed around the circulator turbines to a desuperheater and then to the main condenser. The auxiliary boilers will be started at the beginning of shutdown.

Steam from the auxiliary boilers will flow through the main circulators and then bypass the resuperheaters to the desuperheater. From there it flows through the flash tank to the main condenser. As in normal shutdown, water from the condenser is then pumped and heated with the steam from the flash tank. It then flows to the shutdown boiler feedpumps, completing the heat-sink portion of the cycle.

Each main loop has connections to the auxiliary boiler system to provide steam for the circulator turbine and the circulator bearing-water-pump turbine during decay-heat-removal operation.

Emergency water can be supplied to the shutdown boiler feedpumps from the condensate storage system in case the main condenser is lost. This system thus gives a means for long-term decay-heat removal completely independent of the auxiliary core-cooling loops. Much of the equipment is used for normal operation and thus is already in operation.

5.8 ACS RESPONSE AND DEPRESSURIZATION ACCIDENT EFFECTS

5.8.1 Functional Aspects

The general response of the blanket during a depressurization accident is that the maximum fuel temperature is reached in about 120 sec (see Section 7) after depressurization is complete and no coolant flow is established. The main circulators have considerable power and speed capability so the depressurized conditions or the effects of air ingress do not impose undue requirements. Should a depressurization accident occur, the main cooling loops would continue to provide shutdown cooling and the auxiliary cooling system (ACS) would provide an independent and diverse backup if the main loops should fail. Presently the ACS is designed to operate immediately following signals from a main loop failure to function. The depressurized condition provides the design basis for the ACS circulator and its electrical drive motor. If the main loops fail and the cooling function is transferred to the ACS the initial ACS water temperatures and circulator speed are dependent upon the timing of main loop failure.

5.8.2 Auxiliary Loop Cooler

The auxiliary loop cooler transfers heat to the ultimate heat sink which is air. Its required performance is defined by the loop heat duty and the maximum auxiliary heat exchanger water inlet and outlet temperatures, but the actual design is not yet definitely specified at this conceptual stage. The auxiliary loop cooler is a counter cross flow heat exchanger with water on the tube side and air on the shell side.

5.8.3 Auxiliary Cooling System Response and Performance Following Depressurization

The auxiliary cooling system is made up of the auxiliary heat exchanger (AHE), the auxiliary loop cooler (ALC), the auxiliary circulator and the pressurized water loop with pumps and pressurizer. A schematic of this system is shown in Fig. 5.8-1. Heat is transferred from the reactor coolant to the flowing water system in the AHE and is rejected from the flowing water system to the ambient air in the ALC. The auxiliary circulators are of centrifugal compressor design and each is driven by a squirrel cage induction motor which derives variable frequency power from separate essential buses. The design of the auxiliary circulator is conceptual and only a preliminary design has been made. The torque required of the circulator drive motor is directly related to the system pressure drop. The circulators are nearly constant volume flow machines implying that the mass flow is nearly proportional to the product of the motor speed and the coolant density. For a low system pressure drop characteristic, as would be expected for full depressurization with pure helium as the reactor coolant, the maximum speed limit would be encountered. If significant air ingress occurs in a depressurized reactor, the system density increases and the motor torque limit may be reached and the motor speed would be reduced.

During normal plant operation, the ACS loops are in a standby mode with the auxiliary circulators shut down and the auxiliary-loop isolation valves closed, and a small leakage flow from the blanket inlet manifold will expose the auxiliary heat exchanger to a helium temperature of 282°C (540°F). In order to prevent boiling in the auxiliary heat exchangers should the water flow stop, the system will be pressurized to 14.67 mPa (2100 psia), which is the saturation pressure at 340°C (643°F). The main

Fig. 5.8-1. Auxiliary cooling system schematic

circulating pumps and the large air fans are shut down. Only small auxiliary circulating pumps operate in each loop to provide a small flow through the coolant system to maintain the equipment near operating temperature and prevent thermal shock to the system when the main pump starts. The auxiliary pump flow is small enough so that it does not constitute a major thermal loss for the reactor. The initial water temperature prior to the depressurization accident can be set by the helium leakage flow, the auxiliary pump flow and the small air flow in the ALC.

When a reactor trip or a scram signal is generated, the ACS controls are activated for starting the large air fans and the main circulating pumps. Speed controls of the auxiliary circulator will respond to the main cooling loop performance and the auxiliary circulator is in a ready condition to transfer cooling to the ACS, in case of main loop functional problems.

It should also be noted that the ACS controls, while switching over the cooling function from the main loops to the auxiliary loops, will not impose serious thermal transient effects on the loop components.

With the piping geometry considered, a depressurization accident will cause the helium inventory from one quadrant of the reactor core to be mixed with the air in the secondary containment. The ACS design and its performance are based on this phenomena, since depressurization of more than one quadrant of the blanket leads to a more favorable condition for cooling with an increase in the back pressure. In the final equilibrium condition, preliminary characteristics of the helium/air mixture were estimated as follows:

Approximate equilibrium pressure following depressurization accident	$\approx 1.52 \times 10^5 \text{ Pa}$ (1.4 atm)
--	--

Approximate mass of air in the mixture	$\approx 2.74 \times 10^5 \text{ kg}$ ($6.04 \times 10^5 \text{ lbm}$)
Approximate mass of helium in the mixture	$\approx 18,370 \text{ kg}$ ($40,500 \text{ lbm}$)
Mole fraction of air in the mixture	$= 0.673$
Mole fraction of helium in the mixture	$= 0.327$

It is estimated, based on GCFR experience, that significant air ingress effects on the cooling will be introduced approximately 10,000 seconds following a depressurization accident. The thermal power at this time is approximately 1% of the full thermal power. The thermal and transport properties of the helium-air mixture were determined from the data given in Fig. 5.8-2.

It should be noted that if a rupture occurs ahead of the circulators/heat exchangers, secondary containment air will be pulled into the system and it is possible for a significant air ingress to take place in the final equilibrium condition. If the rupture were after the heat exchanger/circulator, natural convection phenomenon in the secondary containment would govern the air ingress rate. In any case, adequate cooling will be provided by the ACS.

5.8.4 Sequential and Emergency Use of ACS

The ACS, if required for the cooling operating following a depressurization accident, could either be used in conjunction with the cooling by main loops after a deferred time interval, or within a few seconds after the accident. The modes of its possible operations are summarized as follows:

The ACS' basic function has two objectives: one, to serve as a diverse back-up with its service reserved for assistance in case of unavailability of the main loops to cool the reactor following an accident, and second, to

Fig. 5.8-2. Helium-air mixture property data

supplement the long term blanket decay-heat-removal at low thermal power ($\leq 1\%$ full power).

Subsequent to an accident or a trip of the reactor, steam from the main steam generators will be stored in separate pressure vessels and the fusion plasma power will be instantaneously terminated resulting in blanket power being dropped to 5% of its full thermal power. The stored steam will then be used to drive the main circulator turbines for helium circulation and blanket cooling will be performed through the primary cooling loops. It is estimated that the stored steam energy will be sufficient to drive the circulator turbines for about 30 minutes following the accident accounting for the decrease in the helium flow rate with the transient decrease in the afterheat.

At the signal of a reactor accident or a trip, the auxiliary boilers will be operated to capacity and auxiliary drive steam is expected to be available within a few minutes. When signs of the insufficiency of the stored main steam energy to drive the main circulator turbines are indicated, the auxiliary steam is supplied to the circulator turbines for continuing the cooling through the primary loop. The auxiliary boiler system is capable of providing the steam for cooling for shutdown and also for long term decay heat removal on the main circulators.

Thus, the need to the ACS does not arise for this mode of afterheat-removal. The ACS may be used for a long term decay-heat removal at a low thermal power (such as $\leq 1\%$ full power), if it is economically favorable over the auxiliary boiler operation at such conditions. This use in conjunction with the main loop cooling will be the logical use of the ACS following a normal or accidental shutdown.

However, if for some reason the main loops have totally failed to provide cooling and the auxiliary boilers could not come on line in time, then the ACS will be used to provide the emergency cooling. Within a few seconds, the electrically driven ACS circulators will be brought to speed and the system is ready to cool the reactor.

Figure 5.8-3 shows the sequential and emergency points of operation of the ACS and the standard fission-product decay heat curve as a function of time.

Preliminary analysis, using the conservative assumption of adiabatic heating, shows that in case of total loss of the cooling following an accident, it takes about 120 seconds before the fuel could begin to melt. The ACS system is designed to operate within a few seconds following any type of accident, and provides the cooling. In the event of loss of offsite power, starting of the emergency diesel generators and start of the ACS system will be within 85 seconds providing adequate protection of the fuel.

5.9 CONCEPTUAL PRIMARY/AUXILIARY SYSTEM LAYOUT

5.9.1 General

Based on the schematic arrangement shown in Fig. 5.2-3, conceptual layouts of the primary and auxiliary heat removal systems for the fission-fusion reactor were made as shown in Figs. 5.9-1 through 5.9-3. The studies performed indicate that a relatively compact system arrangement is achieved when the total fission blanket assembly (lying between the "Yin-Yang" magnets) is systematically split into quadrants and is interconnected by coolant piping as shown in Fig. 5.9-1. Such an arrangement eases the layout geometry and also has some systems operational advantages. For instance, if a rupture occurs in any one of the eight primary coolant ducts within

Fig. 5.8-3. Auxiliary cooling system operating ranges

Fig. 5.9-1. Hybrid reactor blanket cooling system geometry (one quadrant shown)

PLAN VIEW

SCALE: m 0 5 10

Fig. 5.9-2. Hybrid reactor cooling system

Fig. 5.9-3. Hybrid reactor blanket segment coolant interface (60 atmospheres pressure)

a quadrant, only the affected quadrant is depressurized. The potential use of emergency cooling under depressurized conditions will then be limited to the affected quadrant. The rest of the reactor will remain more efficiently cooled under pressurized conditions.

Significant features of the piping layout scheme shown are as follows:

1. The massive support and clamping structure holding the magnets along their axes will not interfere with the piping arrangement proposed.
2. The tangential entries of the pipes into the blanket modules as shown in Fig. 5.9-3 permit the magnets to be located as close as possible to the modules. This feature acts to limit plant capital cost and preserves plant performance.
3. During normal operation the quadrant interconnecting ducting acts to balance coolant flow, while under emergencies it connects the ACS to all four blanket modules.
4. Heat removal system geometry is compatible with a circular containment building.

5.9.2 Description

Each primary loop consists of four steam generator units and two turbo-circulators with outlet check valves. During normal operation all the primary loop components (indicated by solid lines) are active. Each primary loop consists of two subloops and in the event of a steam tube leak or a circulator failure, the interconnecting pipes and second circulator provide "first line of defense" cooling through the unaffected subloop with a backup of the ACS.

The principal parameters required for sizing the primary loop pipes are given in Table 5.2-1. In order to keep the parasitic cooling system pressure losses to a practical value, a system dynamic head of 3450 Pa (0.5 psi) was selected. Based on this criteria, pipe sizes were determined and found to be feasible at 60 atmospheres pressure. The hot helium delivery pipes are 1.22 m (48 in.) in diameter, while cold return ducts are 1.11 m (44 in.) in diameter. The hot helium pipes must be fabricated from one of the ASME code case 1592 materials, while carbon steel can be used for the cold return lines.

Figures 5.9-2 and 5.9-3 show the conceptual layout of the primary loop. Adequate flexibility will be provided in the ducts to accommodate thermal expansions. All pipes will be provided with suitable insulation. The primary loop also receives a fractional amount of coolant flow from the direct converter. Figures 5.9-1 and 5.9-2 show the connections for this flow.

Since the present work is conceptual in nature, a detailed study was not performed to determine a preferred structural support for the ducts. It is conceivable that a PCRV-type structure could be provided to totally enclose the system or part of it. Alternatively, structural elements, such as channel or I-beam sections could enclose the piping axially and circumferentially to provide reinforcement and rupture area control. In this case structural columns and anchors could conceptually provide ultimate structural support. In future work these features should be studied and evaluated in depth. In any case, it is expected that analysis will show that an offset duct rupture cannot be allowed to occur in the helium systems.

The auxiliary loop (shown by dotted lines in Fig. 5.2-3) provides backup cooling. This system is designed and sized to operate at 5% of

full thermal power. Table 5.9-1 shows the design data used for sizing the auxiliary loop piping. Each quadrant of the reactor has its own auxiliary system which is interconnected through the primary loop for compactness. Figures 5.9-1 and 5.9-2 also show the layout of the auxiliary loop. The ducts provided as part of the auxiliary system also form interconnecting primary loop ducts. These ducts are smaller than the primary ducts, being sized for the auxiliary system flow or the minimum cross loop flow in the event of a main loop failure.

5.9.3 Flexibilities in the Conceptual Layout

The connections of module cooling ducts to the back panel of the core modules as shown in Fig. 5.9-3 could be staggered or located one above the other in order to achieve a better structural interface. Further, the pipe diameters could be reduced at the junction of the back panel to provide a better spacing between the ducts. The resulting pressure drop increase is anticipated to be minimal.

These geometrical flexibilities will be helpful when locating the injector penetrations through the modules. From the point of the design of module piping connections to the back panel, it will be preferable to have the injector openings between two blanket modules (half opening on either side). The optimum locations of injectors with respect to the blanket should be studied in future work.

Several different schemes are being considered for connection of the helium ducts to the blanket modules. The initial concept proposed by Lawrence Livermore Laboratory used radial entry at the equator of the module. Due to module design difficulties, the concept was switched to

TABLE 5.9-1
DESIGN DATA FOR AUXILIARY LOOP SIZING

Power	5% full power
Design pressure	6.08×10^6 Pa (882 psi)
Operating pressure	1.52×10^5 Pa (22.07 psi)
Hot helium temperature	
Design	530°C (986°F)
Operating	500°C (932°F)
Cold helium temperature	
Design	282°C (540°F)
Operating	100°C (212°F)

the tangential entry concept shown in Fig. 4.2-3. As the sizes required for the duct diameter, the module thickness and the inside diameter of the magnet shield have evolved it has become clear that tangential entry, too, has some problems. If module cooling ducts were to be arranged such that one enters the back panel from the top end and the other from the bottom end, such an arrangement will have an increase in the magnet's radius. Figure 5.9-4 shows the implication of such a scheme. The inner radius of the magnet shield will be increased from the present design value 7.56 m to approximately 11 m.

Due to the large duct diameters, however, use of parallel tangential entry ducts as shown in Fig. 5.9-3 may also exhibit problems. Several connection schemes appear to be feasible. The radial entry and parallel tangential entry concepts, particularly, appear to be viable. The final choice will involve tradeoffs between the module design, the ducting design and the magnet design. These tradeoff studies will be done in the future.

A maintenance machine (Fig. 5.9-5) is proposed for detaching and removing the blanket module during refueling operations. This machine would surround a section of the duct to be detached from the module and contain all equipment required to remove a pipe section and cap both ends of the disconnected piping. Such a system appears to be within the state of the art, although it will be large. Adequate safety features for the remote machine operation would include buffered seals, remote weld inspections, and atmospheric purging prior to maintenance and/or module removal.

Fig. 5.9-4. Potential effect of piping geometry on magnet radius

Fig. 5.9-5. Hybrid reactor blanket maintenance machine

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2. Lee, J. D., D. J. Bender, and R. W. Moir, "Optimizing the Mirror (Fusion-Fission) Hybrid Reactor for Plutonium Production," UCRL 76986, November 1975.

6. POWER CONVERSION LOOP CONTROL SYSTEM

6.1 INTRODUCTION

A description of the MHFFR plant control systems are presented in this section. These systems ensure that the plant can be safely operated and in the event of an abnormal or accident condition can be shut down in an orderly manner. The control systems consist of: (1) the automatically and manually initiated systems for ensuring public health and safety under accident conditions; and (2) regulating systems used for normal plant operation. Two distinct systems are involved: (1) the plant protection system (PPS); and (2) the plant control system (PCS).

The PPS prevents an unacceptable release of radioactivity, which would constitute a hazard to the health and safety of the public. The PPS initiates action to (1) protect the fission product barriers, and (2) limit the release of radioactivity if failures occur in the barriers. Plant availability and equipment protection are outside the scope of the PPS, and are primarily in the scope of the PCS.

The PCS is designed to regulate the plant in all normal modes of operation. The PCS also handles conditions imposed on the system during loop trip, reactor trip, or electrical load rejection.

The MHFFR has certain characteristic features which affect the regulating system required. These features are:

1. The circulator turbines are located in the superheated steam lines. Superheated steam flows from each steam generator through

the circulator turbine and then is returned to the steam generator to be resuperheated before flowing with steam from the other loops to the main steam turbine. Steam density at the exit of the circulator turbine varies inversely as the square root of circulator power, a characteristic used in controlling the circulator throttle valve.

2. The log mean temperature difference between the steam and helium, which is in counter-crossflow, is relatively small. Thus the primary and secondary systems are closely coupled in performance.
3. To avoid large changes in the reactor outlet and steam temperatures, the feedwater and helium flow rate must be adjusted in direct proportion to the reactor power.

6.2 PLANT CONTROL SYSTEM (PCS)

The plant control system consists of the on-load plant control system, the operational residual-heat-removal (RHR) control system, and the startup control system. The on-load plant control system automatically regulates and controls plant electrical output in accordance with load demand throughout the normal range of plant operation. The operational RHR system controls the secondary coolant system during the transition from normal operation to long-term decay-heat-removal operation. The startup system performs the control necessary to bring the plant to the normal operating condition.

6.2.1 On-Load Plant Control System

The on-load plant control system provides automatic regulation and control of the electric power consumed and delivered by the plant in accordance with load demand.

The on-load plant control system operates to fulfill the following conditions:

1. Maintain the correct blanket outlet gas temperature (TBO) for the reactor power.
2. Maintain an approximately constant steam temperature upstream of the main turbine throttle valve over the entire normal load range.
3. Maintain the proper first-stage main turbine steam pressure (PMT) required by the load demand.
4. Maintain an approximately constant circulator turbine exhaust density over the entire normal load range by controlling the circulator turbine outlet pressure in each loop.
5. Maintain the proper feedwater flow rate to each steam generator as required by the load demand.

As a consequence of the fourth control condition given above, the ratio of helium-to-steam flow is approximately constant over the operating load range. Since steam flow is proportional to load demand, it follows that the helium temperature rise through the blanket is approximately constant independent of load. The helium temperature level will, however, vary slightly as necessary to establish the differences required to transfer the heat from the blanket to the helium and from the helium to the steam generators.

The on-load plant control system receives information on system behavior from instruments that measure a set of process variables. Then, by comparison with the proper set points of these quantities, which are functions of the actual load demand, the control system generates the feedback signals for regulation.

The proposed control system, illustrated in Fig. 6.2-1, consists of controllers whose functions and input signals are described in the following paragraphs.

1. Blanket Outlet Temperature Controller (C1 in Fig. 6.2-1). The neutron flux reaching the blankets is measured by the flux detectors F and compared with the flux-level set point determined by the gas outlet temperature TBO. Controller C1 then varies the neutral beam injection demand. This will control the flux level such that TBO will have the value correct for the plant load demand.
2. Feedwater Flow Controller (C2 in Fig. 6.2-1). There is a separate feedwater controller for each of the four main loops. Each of these controllers regulates feedwater flow so as to maintain steam pressure at the superheater outlet, PS, constant over the load range.
3. Main Turbine Throttle Valve Controller (C3 in Fig. 6.2-1). The first-stage turbine pressure PMT is measured and compared to the pressure required by the load demand. If it is not correct, C3 causes appropriate throttle valve action.
4. Circulator Turbine Flow Controller (C4 in Fig. 6.2-1). Each circulator is controlled by a controller whose function is to set the circulator turbine throttle valve so as to give the appropriate flow of helium through that loop. Analysis shows that helium flow is proportional to steam flow if the circulator turbine exhaust density is held constant. Therefore, C4 acts to maintain circulator exhaust density constant by controlling the turbine flow to a set point appropriate to the load demand.

Fig. 6.2-1. On-load plant control system

Since steam generator pressure PS is maintained constant by C2, the exhaust pressure of the circulator turbine would tend to increase with decreasing plant load if C4 did not act to close the circulator turbine throttle valve appropriately. The actions of C4 and of C3 with decreasing load are to close the respective throttles so as to maintain PT and PMT at the proper values for the load demand.

An essential consequence of this method of control is that helium temperature rise across the reactor remains approximately constant over the load range.

The following example shows how the control system behaves. Suppose a load-demand signal is given to the main throttle controller C3 to reduce the load. It will close the main throttle valve and compare PMT to the required part-load set-point pressure. With the main turbine throttle valve partially closed, pressure at PT will increase but the circulator turbine throttle controller C4 will sense this and close down on the turbine throttle valve, thus increasing pressure PS. The feedwater controller C2, sensing a higher PS, will close down on the feedwater control valve, tending to increase the pump head, but due to the pump characteristics the flow will decrease, which is actually what is required by the main turbine for the desired part load. With the smaller flow, the pressures PS, PMT, and PT will stabilize at the values required for the new load conditions. As the steam flow through the circulator turbine decreases and the new values of PS and PT are established, the circulator speed decreases so as to establish the new helium flow required. The helium temperature rise will remain unchanged but the blanket helium outlet temperature TBO set point will already have been decreased by a load-demand signal, thus lowering the

helium temperature level. Temperatures through the steam generator will tend to remain approximately unchanged.

6.2.2 Operational Residual-Heat-Removal Control System

6.2.2.1 Plant Shutdown Control System. The primary functional requirement of the plant shutdown control system is to automatically provide adequate blanket cooling during the transition from on-load plant control to decay heat removal control following a reactor trip. The system operates in all normal and emergency shutdown situations, including coolant-system depressurization events. The control system will be designed so that the thermal transients experienced by the plant are kept to a minimum.

For reactor shutdown generated by a reactor trip signal, the following actions are initiated:

1. The neutral beam injector power is removed, thus causing the plasma conditions to be less than necessary to sustain fusion.
2. Startup of the auxiliary steam supply system is initiated in preparation for transition to the decay-heat-removal control mode.
3. The main turbines are tripped.
4. Simultaneously with the main turbine trip, the resuperheater bypass circuits are activated and controlled by signals from the plant shutdown control system.
5. The circulator turbine large control valve (V2) of each reactor coolant loop is closed as the neutron flux decreases.
6. The circulator turbine small control valve (initially completely open) is controlled by signals from the plant shutdown control system.

7. The shutdown boiler feedpumps are activated approximately 1 min after reactor trip. This time is not critical because the water/steam inventory in each steam generator is ample to drive the circulator turbine for a much longer period.
8. Auxiliary cooling system operation is activated as a backup to cooling by the main cooling loops.
9. The emergency diesel generator startup sequence is initiated to provide a backup power supply.

During the power reduction from the normal power level to the decay-heat-removal-level, the plant shutdown control system performs actions 4 and 6.

The plant shutdown control system, which is shown in Fig. 6.2-2, consists of two control loops for each reactor main coolant loop. The first control loop adjusts the circulator turbine steam flow by varying the area of the circulator turbine small control valve (V2S). The second control loop maintains the proper ratio between the circulator turbine exhaust steam pressure (PCO) and the reactor inlet coolant pressure (PH) by adjusting the area of the resuperheater bypass circuit control valve (V4) until the ratio of pressures PCO and PH is at its set point. The latter control action adjusts the circulator power, and therefore the helium flow, in accordance with the reactor coolant pressure during a reactor coolant depressurization accident. This mode of control continues until orderly transition to the decay-heat-removal control system can be made.

For normal reactor shutdown, a programmed load reduction to a preset level is initiated, using the on-load plant control system. Startup of the auxiliary steam supply system is initiated at the onset of the programmed

Fig. 6.2-2. Shutdown plant control system

load reduction. Reactor power is held at reduced level until the auxiliary steam supply system is available. Plant load is then reduced to less than 5% and the turbine is tripped. The resuperheater bypass circuits are activated during this time to control circulator turbine exhaust steam pressure. Resuperheater bypass steam is used to drive the feedpump turbines. Secondary system steam and water flows are reduced consistent with the reduced blanket power. Once the power has been reduced to a sufficiently low level, the shutdown boiler feedpumps are activated and orderly transition to the decay-heat-removal control system commences.

6.2.2.2 Decay-Heat-Removal Control System. The primary functional requirement of the decay-heat-removal control system is to provide adequate cooling for an indefinite period of time following transition from the plant shutdown control mode.

As the reactor shutdown proceeds with time, the quantity and the quality of the nuclear-produced steam will become inadequate to drive the helium circulator turbines. The four auxiliary boilers (one for each main loop) will have been started when reactor trip was initiated and will be automatically brought to operating conditions during the initial shutdown phase. Approximately 30 min after reactor trip, the decay-heat-removal control system, shown in Fig. 6.2-3, is initiated by:

1. Activating the steam-generator alternate discharge circuit in each coolant loop by opening valve V5.
2. Providing auxiliary steam supply system steam to drive the helium circulator turbines of each coolant loop by opening valve V6 and closing V2S.

Fig. 6.2-3. Decay-heat removal control system

This mode of operation, with each steam generator supplied by a separate shutdown boiler feedpump and each blanket quadrant supplied by a separate auxiliary boiler, can continue indefinitely to maintain helium circulation and remove decay heat.

6.2.3 Startup Control System

Three startup conditions are possible. These are initial startup of the plant, restart of the main steam generators and circulators from shutdown operation with the main loops, and restart of the plant from shutdown operation with the auxiliary cooling system (ACS).

For plant initial startup, offsite power is used to cool down and energize the Yin Yang magnets, pump down the vacuum chamber and condition the plasma system for operation. The auxiliary boilers are then utilized to provide the initial steam. Water is circulated through the steam generators using the shutdown boiler feedpumps with the blanket coolant system pressurized. The feedwater cleanup phase is then initiated with hot water until steady-state temperatures are achieved and the ion concentration is acceptable. At this point the resuperheaters and circulator turbines are drained and the bypass lines around them are opened. The shutdown boiler feedpumps are restarted, thus permitting feedwater to flow through the steam generator to the desuperheater and main condenser. Steam from the auxiliary boilers is fed to the circulator turbines but bypasses the resuperheater as in the shutdown mode of operation. Steam passes through the circulator turbines, starting coolant circulation, and then initially through the desuperheater and subsequently the resuperheater to the main condenser. The blankets are thus warmed up by the primary coolant transfer of heat from the steam generators. During this period the conditioning of the fusion

process is completed and the overall plant becomes ready for powered operation. With blanket coolant flow now established, the neutral beam injector current is increased to produce reactor power and supply heat to the steam generators. The neutral beam injectors require substantial power which must be provided by offsite or auxiliary power supplies until the main turbine generator is loaded at about 20% of full reactor thermal power. By regulation of the bypass valves and the steam flows to the circulator turbines, the desired superheated steam conditions at the outlet of the steam generator are obtained.

At this point, the steam supply for the circulator turbines is gradually transferred from the auxiliary boilers to the main steam generators. When the transfer is complete, steam flow is limited to about 10% of normal power by the capacity of the shutdown boiler feedpumps.

To increase the steam flow rate further requires operation of the steam-driven main boiler feedpumps. This is accomplished by taking steam from the flash tank to drive the main boiler feedpump turbines. The main turbine generator can then be started and synchronized and a small load applied. The turbine load is then increased to a minimum level. Once the main turbine is in operation, extraction steam can be used to drive the main boiler feedpumps and the shutdown boiler feedpumps can be stopped. In addition, the ion injectors are switched from off-site or auxiliary power to the main generator. The plant is now self-sustained and can be brought to full load.

Restart from decay-heat-removal operation with the main loops is quite simple since the plasma system is still fully conditioned for operation and the auxiliary boilers and main circulators are already in operation. In

the decay-heat-removal mode of operation, the resuperheater is bypassed and the steam generator acts only as a heat sink for the system. To restart from this condition, the injectors are energized using off-site or auxiliary power to produce power in the blanket, the resuperheaters are put back into operation and the auxiliary boiler steam flow and bypass valves are regulated to give the desired superheated steam conditions at the steam generator outlets. This steam can then be directed gradually to the main circulator turbines and the auxiliary boiler steam decreased until the circulator bypass lines are closed and the plant is operating on its own steam. The plant is then brought to higher power in an identical procedure as described above for the initial startup.

Restart from shutdown operation with the auxiliary core cooling system is very much like the initial startup except that plasma system conditioning and the feedwater cleanup phases will be omitted unless specifically required due to the nature of the shutdown. Feedwater will be supplied by the shutdown boiler feedpumps to the steam generators and bypassed around the circulator turbines to the main condenser, as described earlier. Auxiliary boiler steam will be fed to the circulator turbines and the circulators will be brought up to the speed required to open the main-loop isolation valves. The auxiliary loops will then be shut down. The procedure for switching from auxiliary steam to main steam and obtaining higher power will be identical to that described above.

6.2.4 Supplementary Control Functions

In addition to handling the normal control functions the PCS handles conditions imposed on the system during loop trip, and electrical load rejection. Since there are two generators and only one is required to keep

the plant itself in operation, the PCS monitors the generator output voltage and will keep the plant in operation as long as one generator is producing power.

6.3 PLANT PROTECTION SYSTEM (PPS)

The protection system automatically initiates appropriate action to protect the health and safety of the public during and after plant transients that might cause a release of radioactivity from the blankets. Various critical plant parameters are monitored throughout their operating range by the protection system. Should any of these go beyond the safety limit specified by the technical specifications for the plant, the protection system automatically actuates reactor trip. Certain of these critical parameters are also used to initiate the plant's engineered safety features.

The reactor plant protection system includes all electrical and mechanical devices and circuitry that (1) actuate reactor trip, and (2) in the event of a serious reactor accident, actuate the engineered safety features.

6.3.1 Reactor Trip System

If the protection system receives a signal that a monitored parameter has exceeded its safety limit, the system actuates a reactor trip.

The basic reactor operating philosophy is to define an allowable range of power and to maintain the coolant flow proportional to the reactor power throughout the operating region. The allowable power range and power-to-flow ratio define the primary tripping functions. Additional tripping functions are provided as backup for specified abnormal conditions.

Protective Actions

Rapid reactor shutdown is provided by shutting down the neutral beam injectors. Interrupting the power to the injectors promptly causes the plasma conditions to be less than necessary to sustain fusion. Thus the fusion process rapidly ceases and after the delayed neutron precursors in the blanket decay the heat produced in the blanket is solely due to the decay-heat. The shutdown of the plant following a reactor trip is discussed in Section 6.2.2.

Plant parameters that cause reactor trip are listed in Table 6.3-1 and are described below.

1. Manual Reactor Trip. Manual reactor trip is provided to give the operator a means to rapidly shut down the reactor.
2. High Neutron Flux. High neutron flux indicates excessive heat generation requiring a reactor trip to prevent excessive blanket temperatures.
3. High Power-to-Flow Ratio. A high ratio of neutron flux to helium flow rate through the blankets indicates an excessive reactor power level for the amount of cooling provided and requires a reactor trip to prevent excessive blanket temperatures.
4. High Reactor-Coolant Moisture. A signal of high blanket-coolant moisture, due to the leakage of a significant amount of water into the blanket coolant, causes the reactor to trip. Blanket coolant moisture monitors also provide signals for loop isolation and steam-generator dump.
5. High Delayed-Neutron Activity. A signal of high delayed-neutron activity in the primary coolant is an indication of significant fuel damage and causes the reactor to trip.

TABLE 6.3-1
REACTOR-TRIP PARAMETERS

Sensed Variable	Indication	Type of Input
1. Manual	--	Hand switches
2. Neutron flux level	High	Linear power nuclear channels
3. Power-to-flow ratio	High	Linear power nuclear channels and flow measurement
4. Blanket coolant moisture	High	Moisture monitor
5. Blanket coolant delayed neutron activity	High	Delayed neutron monitors
6. Blanket coolant pressure	High	Pressure transmitters
7. Blanket coolant pressure	Low	Pressure transmitters
8. Containment pressure	High	Pressure transmitters
9. Main-steam-line pressure	Low	Pressure transmitters
10. Main feedwater pressure	Low	Pressure transmitters
11. Plant electrical system	Power loss	Undervoltage relays
12. Main loop trouble	--	Shutdown logic

6. High Blanket-Coolant Pressure. High reactor-coolant pressure is an indication of significant water inleakage into the blanket coolant system. The high blanket-coolant pressure protection constitutes a diverse backup for the high-moisture trip.
7. Low Blanket-Coolant Pressure. Low blanket-coolant pressure is the primary indication of significant helium leakage from the system. A reactor trip is required when the rate of heat generation in the blankets cannot be accommodated by the reduced cooling capacity resulting from lower pressure. The safety limit may be programmed with plant load to reduce the response time when the plant is at high power.
8. High Containment Pressure. High containment pressure indicates reactor-coolant leakage or steam-line rupture within the containment building. In either situation, the appropriate action is reactor trip. The high-containment-pressure reactor trip acts as a diverse backup to the low-blanket-coolant-pressure trip.
9. Low Main Steam Pressure. Low main-line steam pressure is an indication of main steam-line rupture (outside the containment). Depending on the system design, shutdown of the reactor could be appropriate in such a situation. The safety limit is below the main steam-line pressure level during normal operation and anticipated transients.
10. Low Main-Feedwater Pressure. Low main-feedwater pressure indicates loss of both main feedpumps or a feedwater pipe rupture. In either situation, the appropriate action is reactor trip.

11. Plant Electrical-System Power Loss. Plant electrical-system power loss requires that the reactor be shut down due to the energy imbalance between the blanket and the generator.
12. Main-Loop Trouble. A reactor trip is required to rapidly reduce power upon receipt of a main-loop trouble signal. Blanket cooling with the remaining main loop will continue.

6.3.2 Engineered Safeguards Actuation

The engineered safeguards systems are actuated by the engineered safeguards actuation channels. Actuation of the engineered safety function for emergency cooling with the auxiliary loops is provided only when concurrent signals of loop trouble and low reactor-coolant pressure are received by the protection system. Fast transfer to auxiliary loops is then performed automatically.

Automatic containment isolation is provided and the containment cleanup system initiated when either of the following signals is received by the protection system: (1) high containment pressure, or (2) high containment radioactivity.

7. SAFETY

7.1 BLANKET SAFETY EVALUATION

Two aspects of the blanket safety during accident conditions were evaluated. They are the fuel heat-up rate in the event of loss of coolant flow and the forces on the fuel generated by rapid depressurization of the blanket.

7.1.1 Blanket Rod Heat-up Rate

Time required to cause fuel melting with loss of cooling following reactor shutdown is an important parameter for safety analysis. In the MHFFR blanket, the melting point of the fuel (U_3Si) is $930^{\circ}C$. This is much smaller than melting point of the clad. Heat-up rate of the fuel for complete loss of cooling (adiabatic heat-up rate) can be obtained by the following equation:

$$\frac{d\theta}{dt} = \dot{q}'''(t) / \rho C_p \quad . \quad (1)$$

where, θ = temperature of fuel, $^{\circ}C$

t - time, secs

$\dot{q}'''(t)$ = decay heat, watts/cc

ρC_p = volumetric specific heat of the fuel, watts-sec/cc- $^{\circ}C$

For the MHFFR blanket analysis, it will be assumed that the standard fission-product decay curve (Ref. 7-1) represents the power after shutdown. Figure 7.1-1 shows this relation. Figure 7.1-2 represents the heat-up rate

Fig. 7.1-1. Standard fission-product decay heat curve

Fig. 7.1-2. Adiabatic heatup rate of MHFFR blanket fuel after shutdown

as a function of time U_3Si fuel at the peak power portion of the blanket rod. Increase in temperature of the fuel after time t , is given by:

$$\Delta\theta = \int_0^t \frac{d\theta}{d\tau} d\tau \quad (2)$$

This relation is plotted in Fig. 7.1-3, for $\dot{q}'''$ during normal operation of = 200 watts/cc, $C_p = 0.05$ watt-hr/Kg-°C and $\rho = 15.5$ gr/cc.

The nominal temperature of the MHFRR blanket fuel will be 600°C during normal operation. Hence, the time available before fuel reaches the melting temperature following loss of cooling with reactor shutdown is about 120 seconds.

7.1.2 Forces During Depressurization

A depressurization accident caused by first-wall rupture can result in large pressure differences across the blanket rod bundle. In order to reduce the resulting force, venturies have been provided in the exit manifold of the blanket. These venturies will limit the velocity of flow through the blanket and the resulting forces on the blanket during depressurization caused by first wall rupture.

7.2 PLANT SAFETY ANALYSIS

7.2.1 Introduction

7.2.1.1 Objective and Scope of Safety Analysis. The objective of this section is to analyze the preliminary design of the 4380 MWt Mirror Hybrid Fusion-Fission Reactor so that the adequacy of the proposed design to control expected transients and prevent or mitigate accidents can be evaluated. Safety analyses of those systems which do not directly involve the blanket and power conversion system are not within the scope of this study.

Fig. 7.1-3. Adiabatic temperature rise in MHFFR blanket fuel after shutdown

This safety analysis is focused on the more severe expected plant transients and on the design-basis accident. An important objective of the safety analysis is the choice of a postulated accident for use in establishing the design bases for the engineered safety features — emergency cooling and containment atmosphere cleanup — and for the containment system. After examination of potential accidents, the design-basis accident chosen is the postulated gross failure of the helium ducting which causes rapid depressurization of the blanket coolant into the containment building. The design-basis accident is similar to that being used for the Gas-Cooled Fast Breeder Reactor (GCFR) because the blanket coolant systems and the fast-fission fuel, which have a predominant influence on safety, are essentially similar (Refs. 7-2, 7-3).

In the analysis, potential equipment failures and combinations of failures are considered that have a significant probability of occurrence. Systems and components with adequate redundancy are assumed to maintain their functional capabilities. As an important example, adequate reliability of the plant protection system and the neutral beam injector systems to perform reactor shutdown when required is assumed throughout this section.

7.2.1.2 Outline of Safety Analysis. In line with the objectives stated above, the following subjects are treated in this safety analysis section.

Plant shutdown cooling using forced convection of helium in normal and emergency conditions other than the design-basis accident is treated in Section 7.2.2.

Potential reactivity accidents are considered and seismically induced reactivity insertions are discussed in Section 7.2.3.

The development of local flow blockage accidents and their potential propagation are treated in Section 7.2.4.

Secondary coolant leakage into the reactor coolant system, due to steam generator failures, or into the containment building due to a pipe rupture, is analyzed in Section 7.2.5.

Preliminary safety criteria have been proposed for fuel handling and preliminary design work should be performed in future work. Fuel handling accidents are not of primary importance in this evaluation of plant safety and, therefore, are only briefly discussed in Section 7.2.6.

Blanket coolant leakage is discussed in Section 7.2.7, and the potential failure that would result in a most rapid primary-system depressurization is discussed.

In Section 7.2.8, the design-basis depressurization accident (DBA) is defined and the primary-system response, containment building response, and radiological effects are evaluated.

Finally, in Section 7.2.9, conclusions and recommendations for future efforts are given.

7.2.2 Plant Shutdown Cooling

7.2.2.1 General. Assured shutdown cooling using forced circulation of helium in all normal and emergency situations is a fundamental safety requirement of the plant. Therefore, much attention has been given and will continue to be given to means for providing this function with sufficient redundancy and reliability. The shutdown-cooling capabilities of the design presented in this report are discussed here.

Shutdown cooling in the MHFFR consists of two phases: an initial period of power reduction, either in a programmed manner or rapidly through a reactor trip, followed by a long-term decay-heat-removal phase.

Normal shutdown of the MHFFR plant is described in Chapter 6; reactor trip by the plant protection system is described below. Reactor trip results in rapid reactor power reduction to the blanket decay-heat level through the trip of the neutral beam injectors together with a trip of the turbine generator. The associated temperature transients are potentially the most severe that a plant would be expected to experience during normal operations.

During the decay-heat-removal phase, the shutdown cooling of the blanket can be accomplished with either the main cooling loops or the auxiliary cooling loops. The parts of both cooling-loop systems necessary to perform emergency cooling in the MHFFR plant are designed to provide this function during transients and postulated accidents, including design earthquakes, the loss of offsite power, and single equipment failures. It should be noted that in the MHFFR, if one quadrant must be shut down, then the whole plant must be shut down. Neutron flux to the blankets from the plasma cannot be preferentially prevented from reaching a particular quadrant.

Section 7.2.2.2 describes the plant performance in the initial period following reactor trip. Sections 7.2.2.3 and 7.2.2.4 describe plant performance during decay-heat removal using the main and auxiliary cooling loops respectively.

7.2.2.2 Plant Operations and Performance after Reactor Trip from Full Load. The various signals that can initiate reactor trip are given in Table 7.2-1. Upon receipt of a trip signal, the plant protection system

TABLE 7.2-1
REACTOR-TRIP PARAMETERS

Sensed Variable	Indication	Type of Input
1. Manual	--	Hand switches
2. Neutron flux level	High	Linear power nuclear channels
3. Power-to-flow ratio	High	Linear power nuclear channels and flow measurement*
4. Reactor-coolant moisture	High	Moisture monitor
5. Reactor-coolant delayed neutron activity	High	Delayed neutron monitors
6. Reactor-coolant pressure	High	Pressure transmitters
7. Reactor-coolant pressure	Low	Pressure transmitters
8. Containment pressure	High	Pressure transmitters
9. Main-steam-line pressure	Low	Pressure transmitters
10. Main feedwater pressure	Low	Pressure transmitters
11. Plant electrical system	Power loss	Undervoltage relays
12. Main loop trouble	--	Shutdown logic

* Type of flow sensor needs to be established.

initiates reactor trip. Power to all the neutral beam injectors is interrupted, causing the plasma to rapidly fall below fusion-producing conditions. Concurrently, a trip signal is sent to the turbine.

Closure of the turbine stop valve causes the control system to change from the normal turbine control mode to the shutdown control mode. Turbine trip causes the opening of the bypass valves around the four resuperheaters in each quadrant and the opening of the loop feedwater discharge lines to the desuperheater. It also causes the programmed closure of the circulator turbine throttle valves. Loss of extraction steam will result in tripping of the main boiler feedpumps shortly after turbine trip unless a separate desuperheater path from the steam generators to the feedpumps is provided.

The reactor trip signal also initiates the startup sequence of the auxiliary boilers and associated equipment, the auxiliary circulators and associated equipment, and the emergency diesel generators.

The plant design includes many features that contribute to the reliability of cooling after reactor trip. In particular, the two main loops in each quadrant are made as independent from one another as possible. Each main circulator continues to use steam directly from its associated steam generators. Also steam from any steam generator in any quadrant can be routed to the circulators in any other quadrant to insure forced convection cooling of the blankets. Separate electrically driven shutdown boiler feedpumps are started to supply feedwater to each steam generator. Each shutdown boiler feedpump is supplied from an independent shutdown feedwater header. The normal shutdown feedwater supply from the deaerator is backed up by an emergency water supply. Pump-NPSH-tanks connected in each shutdown feedwater header ensure the NPSH (net positive suction head) of the boiler

feedpumps during a changeover from normal to emergency water supply. The shutdown boiler feedpumps are assumed ready to supply feedwater shortly after reactor trip. However, analyses have shown that the water inventory in the steam generators is ample to cover a considerable delay in pump startup or an interruption, such as that required to change from normal to emergency power supply to the pumps.

To ensure an independent, uninterrupted supply to the main circulator bearings, the bearing fluid pump for each circulator may be driven by a steam turbine using steam produced in its associated steam generators. The drive steam would be extracted at the outlet of the steam generators before the circulator turbine throttle valve so that it would undergo essentially no transient during a reactor trip.

The system is designed to operate without a significant transient in case of loss of nonseismic equipment, e.g., due to an earthquake. In case of loss of all nonseismic equipment, the steam from the circulator turbine would exhaust to the atmosphere outside the containment building rather than to the condenser, and feedwater would be supplied from the emergency water supply.

The control actions required in each loop upon reactor trip have a backup wherever possible and will be made as independent of the other loops as possible, so that the most severe single equipment malfunction or failure could result in loss of only one loop.

During shutdown transients the feedwater flow to each of the four steam generators and the steam flow through each of the two circulator turbines in each quadrant are controlled. This may be done as follows:

1. The feedwater flow is reduced to zero as the extraction steam supply for the main boiler feedpumps decreases. It remains at

zero until 60 sec and then is set constant at ~2% of full load flow for the remainder of the initial period, i.e., for more than 30 min. The feedwater flow should not be critical because of the large steam generator inventory; the flow could remain at zero for considerably longer than 1 min and the set value could depart substantially from 2% without important consequences.

2. The circulator-turbine steam flow rate is governed by two valves in parallel located upstream of the turbine in each of the two main loops of each quadrant. The large control valves are used for plant control during normal operation, whereas the smaller valves, which operate wide open during normal operation, are used to control the steam flow to the circulator turbine after the reactor has been shut down. When the reactor is tripped, the large valves are closed rapidly (when a reactor trip signal and a confirmatory indication from the neutron-flux detectors that the reactor power has been reduced are received) and then the shutdown-control system manipulates the small valves to achieve the desired plant transient behavior.

Initially, after a reactor trip, the helium flow falls rapidly as a consequence of the rapid reduction in power to the circulator turbines with, of course, a rapid drop in circulator speed.

The blanket outlet helium temperature will probably swing down and up during the first 2 min over a small range but should not significantly exceed the initial full load value. The maximum cladding temperature will closely follow the blanket outlet temperature and will also probably remain well within acceptable limits. Gas inlet temperature to the steam generators

is expected to vary less than blanket outlet temperature because of the gas volume in the ducting and the heat capacities of walls and other parts.

Thus, these estimates which are based on the GCFR system design indicate that this mode of shutdown cooling can maintain metal temperatures and their rates of change within acceptable limits. There is ample margin in steam conditions and supply to provide time for transfer to the decay-heat-removal cooling mode.

7.2.2.3 Decay-heat Removal with Main Cooling Loops. Approximately one-half hour after reactor trip, the decay-heat-removal phase will be initiated. If the main cooling loops are used, steam from the auxiliary boilers will be introduced into each main circulator supply line ahead of the circulator turbine throttle. Then the resuperheater outlet isolation valves will be closed and the steam from the steam generators will be routed through a bypass line to a desuperheater and then to the main condenser.

Procedures for initiation of decay-heat removal with the main cooling loops are the same following any reactor trip. The procedures for an unplanned shutdown are described in Chapter 6. However, if the plant operator knows of the plant shutdown in advance and chooses to use the main loops in the decay-heat-removal phase, he could bring the auxiliary boilers up to the desired steam condition prior to plant shutdown.

Independent water supplies to the four auxiliary boilers and to the main steam generators (which in this mode are the heat-removal exchangers) are provided. These supplies come normally from the condensate system. In case all nonseismic equipment is assumed to be inoperative, operation continues without feedwater supply and circulator steam is exhausted to the atmosphere outside the containment until cooling is transferred to the

auxiliary loops. If further analysis shows it to be necessary, a seismic Class 1 feedwater supply system could be provided so that open-ended main-loop system operation could continue for an extended period.

The three auxiliary boilers and associated piping and equipment are independent so that the loss of a piece of equipment will affect only one auxiliary boiler system. Each auxiliary boiler system is expected to be capable of providing for 100% decay-heat-removal. (The capacity of the auxiliary boilers is actually based on plant startup requirements.)

7.2.2.4 Decay-heat Removal with Auxiliary Loops. Decay-heat removal can be provided by the auxiliary loops instead of the main loops. Normally, both loops in each quadrant will be used. However, the operator will not normally initiate the automatic switchover sequence that will make the transfer before 15 min after shutdown, at which time the decay heat being generated in the core can be removed using only one auxiliary loop. After the auxiliary blanket-cooling system equipment startup sequence is completed, cooling is initiated with the auxiliary loops. The auxiliary circulators overpressure the main loops to force open the auxiliary-loop isolation valves and to close the main-loop isolation valves. These procedures are essentially the same, though the rate of transfer is different, as those when these loops are used for emergency blanket cooling.

7.2.2.5 Use of Three Main/Auxiliary Loops per Quadrant. In the preceding discussions it is assumed each quadrant of blankets is cooled by two loops, each having one circulator and two steam generators. Further, it is assumed there is no credible event which could simultaneously prevent both loops from cooling the blankets. Nonetheless, fission reactor industry

practice requires that a single active failure be assumed to occur following any postulated event. In the present case, the initial postulated event could be the failure of one circulator, the subsequent "single" failure could then be the failure of the remaining circulator in the quadrant. The blankets would then be without forced convection cooling until the auxiliary cooling system restores blanket coolant flow.

Since the plant protection system trips the reactor on any main loop trouble, the blanket power in this event is only ~5% of full power. As was shown in Section 7.1, the fuel can accommodate 120 seconds of adiabatic heating before fuel melting occurs (clad melting would be further delayed). Since the auxiliary cooling system can restore cooling within 85 seconds after the loss of the second main loop, the present design is considered acceptable even in light of "single failure" criteria.

Economic optimization or future safety analysis could show that having three main loops per quadrant is highly desirable. With three main loops, the failure of any single loop allows safe blanket cooling to continue on the remaining two loops. In the present case a single loop failure most probably requires an immediate plant shutdown. Also, even if a "single failure" is assumed to make two loops unavailable for blanket cooling, the third remaining loop can readily keep the blankets in a quadrant cool. This would greatly reduce the requirements of the auxiliary cooling system and could limit the potential for a blanket-meltdown accident.

7.2.3 Reactivity Insertion Accidents

Although the blankets in the MHFFR are designed to always be sub-critical, there may be events which could cause increases in reactivity. These increases must be considered: overheating in the blankets with

associated fuel melting and the possibility of major changes in fuel geometry during a partial melt, earthquake or depressurization accident.

Fuel melting is the most critical of the reactivity insertion mechanisms. Any accident which leads to fuel melting - earthquakes, depressurization accidents, loss of forced cooling, flow blockage, etc. - should be carefully analyzed. In the GCFR analysis, the Nuclear Regulatory Commission analyzed fuel melt accidents which resulted in reactivity insertion of $> \$100.00/\text{sec}$ (Ref. 7-4). If such events are possible in the MHFRR, then the assumption that the blanket is always subcritical becomes open to question. A rapid increase in local multiplication could lead to potential safety problems even if the system remained subcritical.

Due to the low burnup and resulting low plutonium content of the blanket, it is expected that analysis will show criticality to be impossible under any circumstances. If this is the case, it is expected that reactivity insertion accidents will not pose any significant hazard to the MHFRR.

7.2.3.1 Review of Reactivity Insertion Mechanisms. Reactivity insertion accidents are characterized by reactor power increases caused by incidents that result in inadvertent augmentations of the blanket multiplication factor. Potential reactivity effects include:

- Externally induced blanket element movements,
- Changes of coolant density,
- Introduction of steam into the core,
- Thermal bowing effects,
- Forced displacement of fuel elements,
- Fuel melting.

Blanket movements induced by external forces represent a potential mechanism for changing reactivity, particularly in a fast reactor. In general, such forces will excite vibratory fuel element motions. The effect of this motion could not be analyzed here.

Coolant-flow-induced fuel element vibrations are, at worst, expected to cause a small reactivity noise that is not a safety problem. Coolant pressure forces on the fuel support plate also will cause a small reactivity effect. Earthquakes are the most severe external forces considered in the blanket design, and seismically induced blanket vibratory motions could potentially cause a reactivity effect. Parameters that affect this phenomenon include the quality of the soil as well as the stiffness and the damping factor of the construction.

The only mechanism that can cause a sizable reduction of the coolant density is a depressurization of the reactor coolant system. Depressurization from normal working pressure to atmospheric pressure could increase the core reactivity. This insertion is controlled by the rate of depressurization.

The introduction of steam into the core, which may result from a failure within a steam generator, influences the neutron energy spectrum of the core. The reactivity effect depends on the steam concentration and the state of the blanket. Preliminary estimates indicate that at all times in the blanket life the effect of steam addition is negative and small and does not represent a safety problem. Other safety aspects of steam inleakage are discussed in Section 7.2.5.

Thermal bowing of blanket elements due to temperature gradients across them will have a reactivity effect that is expected to be very small. However, this is strongly dependent on the mechanical design of the blanket. Thus, that analysis is beyond the scope of the present effort.

7.2.4 Local Flow-Blockage Accidents

7.2.4.1 Characteristics of Local Flow-blockage Accidents. Local flow-blockage accidents initially involve only a small portion of a blanket. Potential causes for such accidents are localized blockage within a fuel element or flow reduction in an entire element, submodule or module. Depending on the failure mode and time sequence and the response of the protection system, serious damage either may or may not be prevented and the failure may or may not propagate.

"Whole-blanket" accidents, initiated by overall loop flow reduction or reactivity insertions, are indicated by unambiguous signals that are readily detectable and can be used for initiating protective actions; local flow-blockage accidents, in which only a restricted area of a module is involved, are more difficult to detect with sufficient assurance to make a decision for reactor shutdown. However, when the signal indicating damage is weak, a fast protective action is presumably not required, whereas a large signal from sizable damage means that fast protective action is desirable.

The objective in detecting a local blockage incident is to limit propagation of the damage. The sensitivity specifications for the protection

systems can be defined with regard to an acceptable amount of damage. This maximum tolerable blanket damage may involve more than one fuel element, but it must not be so great as to involve extensive loss of blanket-cooling geometry.

The occurrence of major coolant flow blockages caused by foreign matter is felt to be highly unlikely. The numerous parallel and rather tortuous coolant flow paths leading to each fuel element would prevent large pieces of foreign matter from completely blocking a whole module, submodule, or single element. Additionally, it would be essentially impossible to totally block the annular flow entry area into each fuel element by any single piece of debris.

Analysis of local flow-blockage accidents is hindered by the intrinsic difficulty that an actual sequence of events is unknown. There is a large range of potential local-flow-blockage-accident sequences. For example, consider the consequences of a local flow blockage within a fuel element. Although the deposition of material of unknown origin on a rod spacer grid would, most probably, not completely block the coolant flow around adjacent rods, the assumption of only a partial flow reduction is difficult to justify. Therefore, from a safety standpoint, a complete local loss of coolant to the fuel rods involved may have to be postulated. Similar degrees of conservatism may have to be introduced when analyzing the propagation of a local fuel-rod failure to neighboring rods and across the element wall into adjacent elements.

The effects of flow blockage within a fuel element are treated in Section 7.2.4.2. Detection of local flow-blockage accidents is discussed in Section 7.2.4.3.

7.2.4.2 Flow-blockage Accidents within an Element

7.2.4.2.1 Partial Blockage Accidents. Partial fuel element flow blockage could potentially be caused by material being deposited on the fuel-rod spacers or support grid, or by fuel-rod swelling and thermal distortion. The consequent local undercooling could result in cladding failure and eventually in fuel-rod damage and relocation of fuel and cladding debris.

Preliminary analysis shows a relatively large amount of coolant flow reduction can be tolerated before the cladding fails. As an example, a flow reduction in excess of 70% is required in the GCFR plant in order to raise the maximum cladding temperature to the melting point.

Mechanisms that could potentially lead to the propagation of failure from fuel rod to fuel rod include:

1. Relocation of fuel-rod debris in adjacent cooling channels and on spacers.
2. Flow reduction through the entire fuel element as a result of the locally increased flow resistance.
3. Flow reduction through parts of an element as a result of coolant bypassing through holes in the element wall as a result of a melt-through failure of the wall.
4. Reactivity changes of the entire system caused by blanket material relocation, which may result in a power increase and therefore in an acceleration of the failure development.

Because of the complexity of the problem, a simplified model has been used to obtain lower-limit estimates of the periods of time involved for the

propagation of failures from rod to rod. This model is based on the following assumption:

After the coolant channels around a particular fuel rod have been blocked, the heat transfer between the rod and its surroundings is interrupted (no convective or radiation heat transfer). Therefore, the heatup of the rod is adiabatic.

The adiabatic heatup of the fuel rod to the relocation temperature can be represented by:

$$\frac{\Delta T}{\Delta t} = \frac{Q}{c_p} \quad , \quad (3)$$

where ΔT = temperature increase to reach the relocation temperature.

Δt = time to reach melting (sec),

c_p = mean volumetric specific heat of the fuel rod,

Q = average full-power volumetric heat generation in the fuel.

From Eq. (3) the time to reach melting at full power amounts to

$$\Delta t = \frac{\Delta T c_p}{Q} = 6 \text{ sec.}$$

Propagation of the failure to other channels is difficult to assess at this time. However, for reference, GCFR propagation times (element to element) are on the order of 10-70 seconds. Propagation times are expected to be larger in the MHFRR blanket due to the large amount of structural material between elements.

It should be noted that because of the loss of structural material, there is less neutron scattering in the melted region. Thus the neutron spectrum is hardened and depending on the change in neutron leakage there may be a local reactivity insertion. This means the multiplication factor is larger and heat generation rates increase. This in turn promotes more

melting. Prompt detection of melting and reactor trip can limit the extent of fuel melting. During a localized fuel-melt transient it is expected that the blanket will remain far subcritical.

7.2.4.3 Detection of Local Flow-blockage Accidents. Rapid detection of local failures is desirable in order to initiate protective actions that will limit the damage and reduce the contamination of the primary circuit.

According to the present design, the following signals will indicate the occurrence of failures:

Blanket module coolant outlet temperature

Blanket-coolant-activity

Power level

Whereas the coolant outlet temperature is a signal for an alarm, the coolant activity and the power level are signals that will cause a reactor trip after having exceeded preset operation limits. In the subsequent paragraphs, the response times of these signals are discussed with respect to the propagation time of local failures. A value of 15 sec is used as a representative number for the minimum propagation time of a local failure to an adjacent element.

- a. The coolant outlet temperature of each module is monitored. This temperature would increase from the beginning of the accident and, assuming a thermocouple response time of 1 to 2 sec, unambiguous signals might be expected after 2 to 3 sec in the case of a major flow blockage accident, and thus before propagation of damage to adjacent elements.

In any circumstance, the thermocouples are intended as operational system devices and will not be included as part of the safety related plant protection system.

- b. In cases of cladding failures the fission gases would be transmitted by convective flow into the helium coolant. Instrumentation lines monitoring the activity will pick up increased activity or delayed neutrons. Estimates indicate that the system response time is less than 5 seconds for detection of clad melting. This would allow shutdown of the reactor before the damage could propagate to adjacent fuel elements.

To detect the potential for fuel overheating, the power level (neutron flux level) in the blanket can be monitored. Due to the geometry of the blankets an accurate determination of local flux levels may be difficult. However, if this problem can be surmounted, the flux monitors should be able to detect an overpower problem in less than 20 seconds.

7.2.5 Secondary-Coolant-System Leakage

Leakage of water or steam from the secondary-coolant system can occur within the helium loop, outside of the helium loop but inside of the containment building, or outside of the containment building. Because of the positive pressure differential from steam to helium under all normal and accident conditions (except for the resuperheater - see Section 7.2.7), it is impossible for significant amounts of fission products to leak from the

reactor coolant system into the secondary system. As such, failure of a water or steam line outside of the secondary containment is not considered to be a serious safety problem.

Discussions of secondary-coolant leakage are divided into steam and water leakage into the blanket coolant system, treated in Section 7.2.5.1, and steam-line breaks within the containment building, treated in Section 7.2.5.2.

7.2.5.1 Leakage into the Reactor Coolant System. Steam in the reactor coolant system of a gas-cooled fast reactor with metal-clad fuel does not result in significant damage to the plant. No exothermic chemical reactions with reactor internals will occur. Any significant metal-water reaction requires temperatures much higher than that of the cladding. The circulators, steam generators, and auxiliary heat exchangers will be designed to operate satisfactorily with the higher coolant densities associated with steam-leak accidents. With continuing steam or water addition to the blanket coolant system beyond what the helium purification system can remove, the system pressure will increase. The moisture detection system will detect the moisture in the leaking loop and the plant protection system will actuate reactor trip, isolate the loop and dump the steam-generator inventory to the atmosphere.

If there is no operator or protective action to control the leak, the blanket-coolant-system pressure would continue to increase until the safety valves would open and a steam-helium mixture would be blown into the containment building.

7.2.5.1.1 Potential Sources of Water Leakage. Possible leakage sources of water or steam into the reactor coolant system are the steam generators, the auxiliary heat exchangers and, if water bearings are used, the circulator bearing-water supply. The auxiliary heat exchanger has a lower pressure and smaller tube sizes than does the steam generator, and it has a much smaller coolant inventory. Consequently, the water or steam flows due to possible ruptures in the auxiliary heat exchanger are less severe than those of the main steam generators. Potential leakage of circulator bearing water could be prevented by a water seal and two helium seals between the water bearing and the reactor cavity (Ref. 7-4). The seal leakage would be collected in the helium-water drain and discharged into a high pressure separator tank external to the circulator. Even if both buffer-helium supplies are unavailable and the jet pump of the helium-water drain system does not function properly, it is estimated that the maximum leakage of bearing water into the blanket coolant system could be limited to 2 lb/sec. Since this leakage rate is considerably less than the maximum rate caused by a steam-generator failure, the leakage of circulator bearing water was not treated as a separate accident.

Steam or water leakage from the steam generator into the helium can be caused by a steam-generator-tube rupture in either the economizer, evaporator, or superheater sections, or a break in a lead-in tube or a lead-out tube.

7.2.5.1.2 Leak Rates. The leak rates from a steam generator failure can be readily limited by orificing. It is a commonly used practice in both the HTGR and GCFR. The leak rate limits should be based on insuring that

the helium coolant pressure relief valves do not open during a water ingress transient. Preliminary estimates indicate that ingress flows in excess of 100 lb/sec could be tolerated.

7.2.5.1.3 Sequence of Protective Actions Following a Secondary-coolant System Leak. It is estimated that the largest leak of secondary coolant into the blanket coolant system is caused by a rupture of a superheater leadout tube. This is considered to be a credible occurrence and, although a double-ended offset failure is very unlikely, it is taken into account in the design of the plant. The first protective action is initiated by the plant protection system upon detection of moisture. At a certain moisture level, the plant is tripped, and the loop identified to have the leak shut down. The circulator turbine is tripped, which in turn causes the helium valve to close, the leaking steam generator valves are closed, and the inventory is released to the atmosphere. This procedure need not be very rapid.

Loop moisture detection will be made highly reliable through the use of redundant sensors to minimize the possibility of faulty detection that could possibly lead to isolating and dumping the wrong loop. The automatic loop shutdown logic will prevent dumping more than one loop. If the loop dump fails to work or if the wrong loop is dumped, the reactor would still be shut down. As a backup, high blanket-coolant-system pressure will cause a programmed load reduction at an intermediate high pressure and the reactor will be tripped at a higher system pressure. Upon reactor trip, the feed-water flow will decrease rapidly and subsequently the leak flow into the

helium system would be reduced. Another signal that will cause reactor trip because of a secondary system leakage is the blanket-coolant high moisture signal.

The consequences of failure of the loop steam dump system to operate properly are not severe. As a consequence of the loop shutdown, the circulator of the leaking loop stops and the steam generator loses its heat load since the isolation valve on the helium side closes. From then on, the steam leaking into the blanket coolant is produced by stored heat in the steam generator tubes and by flash boiling of saturated water. Analyses indicate that the amount of steam leaking into the reactor coolant system could be >10,000 lb, which is sufficient to cause the relief valve to open to relieve the pressure. About 10% of the helium inventory could be released into the containment building. The consequences of wrong loop moisture detection are similar. The wrong loop would be shut down and dumped by the plant protection system, but a high-moisture signal to the protection system would cause a reactor trip that would stop the normal flow of feedwater to the leaking loop, thus limiting the amount of water leaking into the blanket coolant system. Enough water would leak to cause the helium loop relief valve to open but only a small fraction of the reactor-coolant inventory would be released into the containment building.

7.2.5.1.4 Conclusions. The analysis shows that a steam or water leak into the blanket-coolant system does not endanger the plant or any equipment. It is estimated that adequate time is available for the response of the plant protection system, with respect to the initiating event and the following transient.

In the calculations of the superheater lead-out tube leakage, a rupture directly at the tubesheet was assumed such that the break creates an ideal nozzle without any friction. The redundancy of the moisture detection system assures a proper response of the operational protection system. The time necessary for the dump (25 sec) and the linear decrease of the leak rate during dump used in the analysis are also considered conservative assumptions. Even in the case of the largest possible steam leak, a dump of the steam generator inventory within about 50 sec would prevent the discharge of reactor coolant into the secondary containment. The actual delay time of the moisture detection system is expected to be between 10 and 12 sec.

7.2.5.2 Secondary-coolant Pipe Break within the Containment Building. A rupture of a secondary-coolant pipe in the containment will cause the loss of one loop and a pressure rise in the containment atmosphere. The plant protection system will initiate an automatic loop shutdown and a programmed load reduction but the leak will not damage any plant equipment and consequently poses no radiological danger to the environment.

Separate feedwater pipes penetrate the containment and supply each steam generator with feedwater. The steam from the resuperheater outlet flows to the main turbine through the steam lines that penetrate the containment individually and are joined together in the turbine building.

Any major steam leak within the containment could be detected by acoustic detectors, by loop resuperheater steam pressure, or by circulator speed signals in the affected loop. Low-main-feedwater-pressure and high-containment-pressure signals would actuate a reactor trip.

7.2.5.2.1 Feedwater-line Rupture. A conservative assumption made in this study is that the full feedwater flow of the plant is discharged into the containment until the loop is isolated or until the boiler feedpumps are shut down. Actually, the feedwater flow would drop rapidly due to flash boiling of the feedwater (at the saturation pressure) in the line and in the high-pressure heater. The control system reduces the feedwater flow to zero after reactor trip. Reactor trip is actuated by low-feedwater-pressure and by high-containment-pressure signals.

For the GCFR plant an analysis was based on an initial leak flow which was almost twice the total normal feedwater flow for the plant and about six times the normal flow through the ruptured pipe, which accounts also for the initial transient after the pipe rupture (Ref. 7-3). The leak rate was decreased linearly to zero within 20 sec, which is a very conservative assumption. A total of 25,000 lb of feedwater with an enthalpy of 391 Btu/lb is discharged into the containment, causing a pressure rise of about 1 psi and a temperature rise of 10°F. Similar results could be expected for the MHFFR.

7.2.5.2.2 Failure of a Steam Line between Superheater and Circulator Turbine. A failure of a high-pressure steam pipe upstream of the circulator turbine will cause the largest possible steam leak into the containment. In this event, the steam-generator inventory is blown off rapidly and subsequently a large fraction of the plant's feedwater flow will be discharged through the steam generator and the ruptured pipe into the containment. Since the circulator will lose its power supply, it will slow down very rapidly, the helium loop isolation valve will close, and the steam generator

will lose its heat load. Consequently, the feedwater will not evaporate in the steam generator but will leak into the containment at its inlet enthalpy.

7.2.5.2.3 Steam-line Break Downstream of the Circulator Turbine. At full power, the circulator turbine steam flow in the turbine nozzles would be almost choked out. This limits the largest leak of a single steam-line rupture downstream of the circulator turbine in the containment. If a steam-line rupture occurs after the circulator turbine, the amount of steam dumped into the containment is smaller than if a rupture occurs upstream of the turbine, yet the enthalpy of the steam may be higher.

In the event of a steam-line break, a signal from loop acoustic sensors, from a low loop-resuperheater-steam pressure, or from circulator overspeed could rapidly actuate the plant protection system to isolate the affected loop and limit the steam leakage into the containment. High containment pressure provides an additional backup, causing reactor trip and containment isolation.

7.2.5.2.4 Conclusions on Steam-line Breaks in the Containment Building. Detailed analysis of the pressure and temperature histories of the containment after different steam-line ruptures could not be fully assessed here. However, preliminary analysis indicates that the pressure and temperature rises caused by any secondary-coolant leak are smaller than the design containment pressure and temperature, based on a depressurization accident.

7.2.6 Fuel-Handling Accidents

The design of the fuel-handling equipment and fuel storage system will make the occurrence of a serious fuel-handling accident an unlikely event.

Continuous and uninterrupted cooling of irradiated blankets during fuel-handling operations is essential. For this reason, each fuel-handling machine or set of machines that handles an irradiated blanket will have two independent cooling systems operating while handling that blanket. (These systems can be in the form of direct air, helium or water cooling.) When the blanket is transferred from one machine or set of machines to another, both independent cooling systems of the machine or set of machines receiving the blanket will be in operation and providing cooling before the operation of the cooling systems of the machine or set of machines delivering the blanket is terminated. At no time during the blanket handling cycle will cooling be impaired to the point that the handling motions could not be terminated and the blanket remain safely cooled.

Due to the conceptual stage of the design for the MHFFR, adequate assessment of fuel handling accidents could not be made. The potential for dropping blankets in the containment or during handling should be assessed in future studies. Also the potential for and consequences of cooling system failure or interruption should be determined.

7.2.7 Reactor-Coolant Leakage

The primary hazard associated with reactor-coolant leakage is the radiological exposure from the coolant-borne tritium and fission products that leak out with the coolant. The radiological aspects of this are discussed in Section 7.2.8.4.

The MHFFR blanket coolant system is similar to that of the GCFR in that it uses high pressure helium. The only system extensions outside the main ducts are small-diameter sample, instrument, and purge lines, resuperheater tubes. In addition to having a small diameter, these lines are

provided with shutoff and backflow check valves wherever necessary to prevent significant coolant leakage in the event of a break. Consequently, the only significant areas of potential accidental blanket-coolant leakage are in the main ducts and the blanket assembly, these being closed by welded seals and held in place by redundant fastenings tied into the magnet support structure. The welded seals are backed up by "rip-stops" that act as flow-restricting devices. Cracks in the ducts could allow leakage, but such cracks are unlikely because of the design of the ducts and the inservice inspection program. Further, the leakage would be much smaller and lower than that through a major duct rupture.

7.2.7.1 Leakage from External Lines. As indicated above, most blanket coolant leaks are slow. They would arise primarily from failures of small piping. The sizes of the small lines have not been fixed; however, they are expected to be on the same order as those used in the 330 MW(e) HTGR being built for Public Service Company of Colorado at Fort St. Vrain. The study conducted for the Fort St. Vrain HTGR indicated that the largest credible leak would be due to a failure involving the helium-purification-system lines. In the GCFR and in the HTGR, the potential leak rates are small, i.e., 3.4 lb/sec (Refs. 7-5, 7-6, 7-7). In the MHFFR the helium purification system is quite large to accommodate tritium recovery; therefore, larger flow rates could be expected.

The flow rates can be reduced by orificing at the tap-off points in the main ducts, by going to smaller diameter piping or by incorporating fast valving in the system to limit the leak after it occurs. In other words, there are several available methods to reduce the potential leak rates at any location to acceptable levels.

It should also be noted that all the radioactivity bearing portions of the helium purification system can be contained within the containment building. This implies there can be no direct release to the public in case of a failure and, therefore, that some latitude in design approach is available.

7.2.7.2 Resuperheater Failure. The operating steam pressure of the resuperheater is less than the operating pressure of the secondary loop helium. Therefore any resuperheater failure results in fission products from the helium coolant entering the steam system. Once in the steam system, the fission products can be released to the public via the turbine gland seals or air ejectors.

This accident has a precedence in the HTGR reheater (Ref. 7-7). Analysis has shown that in the HTGR, a reheater failure does not cause undue risk to the public. The same should be true of a resuperheater failure in the MHFRR.

A resuperheater failure can easily be detected by radioactivity monitors at the outlet lines of the resuperheater. Isolation valves could then stop the release of helium to the secondary system before significant radioactive contamination occurs.

7.2.7.3 Failure of Main Ducting, Heat Exchanger Vessels and Blanket Assemblies. The most rapid depressurization accidents would be caused by main helium duct, heat exchanger vessel or blanket assembly failures. In the Light Water Reactor industry it has been demonstrated that pressure vessels can be designed and built such that the probability of catastrophic failure is so small that a failure does not need to be considered. In both

the blanket assemblies and helium ducts, large ruptures must be considered and efforts must be made to limit the consequences of such ruptures.

A massive failure of a blanket assembly or helium duct is probably unacceptable in the MHFFR. The shock waves generated by a massive failure could potentially cause severe structural damage to all cooling systems and possibly put the fuel into a geometry which would prevent cooling. The assessment of these forces and their effects during a depressurization transient is extremely complex. Therefore, in general, it is prudent to insure that any failure will produce such moderate pressure transients that the shock waves and larger inertial forces will not be limiting. On the other hand, it is equally important to choose a large enough "limiting rupture size" such that overall design and economics of the plant are not jeopardized.

Due to the large helium inventory in each quadrant of blankets and the long duct runs inherent to the power conversion loop design, the limiting hole size for the blankets (where unacceptable damage is prevented) could be large compared to the 0.065 m^2 (100 in.^2) limiting hole size for an HTGR or the 0.032 m^2 (50 in.^2) limiting hole size of the GCFR (Refs. 7-3, 7-8). The assessment of the appropriate limiting hole size for the MHFFR is beyond the scope of this study. However, for the remainder of this section it is assumed that the hole size caused by a rupture has been limited such that forces imposed on the fuel and cooling system remain within acceptable limits throughout a depressurization transient.

Methods of limiting the size of a rupture might include the use of "rip stops," internal flow restrictions, wire winding, added structural supports, use of prestressed concrete reactor vessel technology, and doubly enclosing all potential failure locations. Due to the difficulty of designing and

supporting the system of ducts presently considered, it is reasonable to assume that doubly enclosing all failure locations would present technical problems whose cost far outweighs the benefits. Therefore, double enclosures are not considered a feasible method of limiting a depressurization in this report. "Rip-stops" can take the form of rings or cross bars which are welded onto the ducts or blankets at potential failure locations. In this way it is expected that the maximum credible rupture could only produce a hole whose area is equal to the area between the "rip stops." Added structural supports would probably be needed to prevent excessive movement of the ducts or blanket during a rupture event. Excessive movement could cause forces beyond the capability of any "rip stops" and the classical double-ended offset rupture would then be considered credible. It is fairly certain that an offset shear or equivalent rupture could not be tolerated.

With wire winding, the effects of "rip-stops" and added structural supports could all be combined into one system. Multiple rows of wire winding could limit a rupture size plus insure that the assembly or duct stays in position and does not cause added failures. Nonetheless the feasibility of wire winding could not be fully addressed here. A concept that may have promise is to use a PCRV to enclose part or all of the reactor system. Unfortunately neither these nor other nonconventional methods could be considered within the available time frame of this study.

7.2.8 Design-Basis Accident

7.2.8.1 Introduction. A preliminary consideration of the design-basis accident (DBA) for the MHFFR is addressed here. The DBA as defined here is initiated by a hypothetical sudden, complete failure of the largest

helium duct so that the blanket coolant system is depressurized into the secondary containment.

The DBA is postulated for analysis of the engineered safeguards capability; however, the DBA is considered highly unlikely because of the special design features incorporated in the plant's design as noted earlier.

7.2.8.1.1 Sequence of Events in Design-basis Accident. The DBA is postulated to start with the nonmechanistic gross failure of a helium duct in one quadrant of blankets. It is assumed that the reactor coolant depressurizes into the secondary containment through an opening of the size limited by the flow restrictor or rip stops discussed in the preceding section.

The plant protection system receives a signal of low blanket-coolant pressure and initiates reactor trip.

Rising pressure in the containment building caused by the leaking reactor coolant causes containment isolation and starts the containment-atmosphere cleanup system.

In the early part of the accident (first few seconds), the control system and the plant respond in a manner similar to that in a pressurized trip. The feedwater flow from the main boiler feed pumps is assumed to stop shortly after the turbine stop valve closes. Steam to drive the main circulators and circulator bearing-water pumps continues to be provided by the inventory in the steam generators of each loop. Low-capacity shutdown boiler feed pumps, each supplied with power from an essential bus, are started and 60 sec. after reactor trip feedwater flow to each steam generator is resumed. As the reactor-coolant pressure decays, the control system provides the required increase in circulator power to maintain adequate helium circulation.

Several seconds after the start of the accident, the reactor coolant system will have reached pressure equilibrium with the pressure in the containment building. After several more minutes, the reactor temperatures reach a quasi-steady-state decay-heat-removal condition. During this period, temperatures rise to maxima and then slowly decrease.

After the plant shutdown period, which lasts for approximately one-half hour, the decay-heat-removal phase is initiated in a manner similar to that following any reactor trip. The procedures are discussed in Section 7.2.2.3.

During depressurization of the reactor coolant system, the containment atmosphere increases in temperature and pressure. Because of the energy content of the reactor coolant, the containment pressure will rise rapidly from atmospheric to a peak and then fall to the equilibrium backpressure (estimated to be within approximately 20 min). Concurrently, the containment temperatures increase and then decay to their initial values.

The source of activity available for release to the containment building is the blanket-coolant fission-product inventory, which is primarily due to fission products released from blanket elements that may be defective at the time of the DBA. The containment-atmosphere cleanup system will remove activity from the containment building and rapidly reduce the amount available for leakage to the environment.

7.2.8.1.2 Plant Performance Following Depressurization. The plant design includes many features that contribute to the overall reliability and the assurance of effective blanket cooling following a DBA. Either of the two independent main loop reactor cooling systems is capable of providing this shutdown cooling. Each reactor cooling system is designed to

retain functional cooling capability during and after operation basis and safe shutdown earthquakes, the loss of offsite power, and single equipment failures.

Analysis shows that the MHFFR plant has the capability to safely cool the blanket after a blanket-coolant depressurization accident where no blanket cooling is performed with the loop in which the failure occurs. After this accident, the reactor can be safely cooled even after a failure that terminates the cooling capability of the remaining main loop (worst single failure). Upon receipt of a loop trouble signal caused by the termination of cooling with the second loop (a low-reactor-coolant-pressure signal had been previously received), the transfer of cooling to the auxiliary cooling loops is started.

The design of the main cooling loops is such that the loss of electrical power does not have a major effect on the use of these loops for shutdown cooling. The loss of offsite power might cause a delay in restoration of feedwater flow to the steam generators by the shutdown boiler feed pumps or an interruption following startup. However, analyses have shown that the water inventory in the steam generators is ample to cover a considerable delay in the startup or interruption in operation of the shutdown boiler feed pumps.

The auxiliary cooling system is driven by electrical power, but it is designed to provide adequate performance with the loss of off-site power.

The emergency generators and the emergency power system will be designed so that the failure of off-site power and the transfer to the emergency diesel generators will not cause critical temperatures to exceed specified limits.

Both the main and the auxiliary cooling systems can provide adequate shutdown cooling after a Safe Shutdown Earthquake. All equipment in the auxiliary cooling system is Class I and, therefore, the entire system remains available for operation after a DBA with a concurrent Safe Shutdown Earthquake.

The parts of the main cooling system that are not Seismic Class I or do not have Class I capability are in the secondary coolant system. If all nonseismic equipment is lost, the normal steam system is not available to receive the steam exhausted by the circulator turbines; this steam is therefore exhausted to the atmosphere outside the containment building. All steam piping in the containment is Class I. If the normal feedwater supply is not available because of the loss of the heat sink or because of a break in the nonseismic equipment of the feedwater system, the Class I emergency water system will supply water.

The containment-atmosphere cleanup system is Class I. The system contains multiple units to circulate and clean up the containment atmosphere after a DBA. Should any one unit fail to start or perform properly, sufficient capacity is available in the other units. Each unit is powered from a separate essential power bus. Loss of off-site power at any time when this equipment is required would delay the operation of the equipment to permit the transfer to emergency power for a short period (less than 2 min). The containment-atmosphere cleanup system is an engineered safety feature and has a high priority among equipment that receives emergency power. To show margin in the design of the containment and the containment-atmosphere cleanup system, a hypothetical accident involving the release of fission products associated with a TID-1844 accident (Ref. 7-10) needs to be

analyzed. Also the effects of hot helium jets on other systems in the containment need to be analyzed. This analyses could not be attempted during this study.

7.2.8.2 Blanket-coolant-system Response. The blanket-coolant-system response during a depressurization accident depends on the size of the leak that occurs. The response also depends on the location of the leak because for some locations the flow associated with the leak tends to help the cooling whereas for others the tendency is the opposite. Also, for a given size of leak area the rate of depressurization is affected by leak location because of the temperature of the leaking gas and the resistances of the paths between the various system plenums and the leak. After depressurization is complete, the system reaches a quasi-steady-state condition in which reactor temperatures are determined by the decay-heat level, the coolant flow rate, and the reactor coolant inlet temperature. The coolant inlet temperature is directly related to the temperature of the feedwater entering the steam generators. Furthermore, the response of the system to accidents compounded by failure of system components is also of importance. In the present context, only the worst single failure-loss of the remaining main loop was considered.

7.2.8.2.1 System response During Depressurization Accidents. In the depressurization accident studies, the reactor is tripped by a low-pressure signal when the helium pressure drops significantly below its normal operating value. The reactor trip signal also trips the main turbine and changes the system control from the normal operation mode to the shutdown mode. In the shutdown mode, the steam flow bypasses the resuperheater section of the

steam generator and the control is of steam flow through each of the two circulator turbines and of feedwater flow to each steam generator. This control is initially identical to that following any reactor trip, which is described in Section 7.2.2.2. The feedwater flow in each loop was ramped to zero as extraction steam decreases following turbine trip. It remains at zero until approximately 60 sec and then is maintained at 2% of full-load flow for the remainder of the transient. The circulator turbine steam flow is reduced to a level compatible with the decay heat being generated.

The circulator turbine steam flow is reduced further as depressurization becomes complete, and held constant at a minimum value until about 10 min, after which it is gradually reduced so as to maintain the reactor outlet temperature below 530°C.

Preliminary analysis shows that the steam-generator inventory stays at a satisfactory level. During the transient, each circulator speed initially increases after the turbine trip then falls rapidly to correspond to the blanket power after the reactor trip and then accelerates and levels off as the coolant pressure comes to equilibrium and air enters the system. Steam conditions for the circulator turbines is estimated to remain satisfactory throughout the initial 30 min.

For arbitrary, but reasonable, control of circulator steam and feedwater flows, preliminary estimates show that during depressurization and approach to a quasi-steady-state decay-heat-removal condition, metal temperatures are well below damage limits.

Approximately one-half hour after the depressurization is sensed, the cooling system is transferred from the shutdown phase to the decay-heat-removal phase; the auxiliary boilers will supply steam to drive the circulators and the steam leaving the steam generators will be bypassed to the

condenser. The latter phase of shutdown cooling and the decay-heat-removal phase will be very similar to that following any reactor trip, as discussed in Section 7.2.2.3. The electrically powered auxiliary cooling loops remain in standby condition as a backup cooling system.

7.2.8.3 Containment-building Response. The results of the depressurization accident studies discussed in Section 7.2.8.2 should be used to evaluate the pressure and temperature response within the containment building during a depressurization accident. The calculated helium temperature and leak rate into the containment would be used as input for a code such as CONTEMPT which performs detailed mass and energy balances within the containment and accounts for energy removed from the containment atmosphere by heat transfer to internal structures (walls, floors, ducting, and equipment) (Ref. 7-9).

The pressure and temperature within the containment building during a design-basis accident could approach 30 psia and 300°F. This implies the containment must be designed to accommodate these pressures and remain leaktight. Industrial experience indicates that in general the seismic and missile design requirement for the containment building automatically make it capable of withstanding this moderate peak pressure.

The large mass of the containment building will not be significantly affected by the transient temperatures in the containment. However, future analysis should consider the effect of these temperatures on fusion related equipment in the containment, e.g., cryogenic flasks or superconducting magnets.

7.2.8.4 Radiological Evaluation. As has been shown in the light water reactor (LWR) industry, the mechanisms for clad-fuel interactions are not

clearly understood. Because of this uncertainty in long term clad behavior, it is generally assumed for design purposes that 1% of the clad is defective. This is consistent with both LWR and GCFR practice.

Given the excellent retentive properties of U_3S_4 fuel, the fission products which leak out from the 1% defective fuel will still result in substantial "plate-out" on the internal surfaces of the helium loop. This plateout is estimated to be $\sim 10^4$ ci/quadrant. During depressurization accident, significant amounts of the plate-out will be lifted-off and blown into the containment building. Also, the fission product release from the fuel will lead to a steady state circulating radioactivity level in the helium coolant amounting to 10^3 ci/quadrant. It is expected that the coolant tritium activity will be about the same level. It would have to be assumed that this will all be released in a depressurization accident. Because tritium is much less radiologically toxic than fission products, the tritium does not contribute significantly to the hazard potential of a depressurization accident.

The problems of having the circulating fission products routinely flow to the helium purification/tritium recovery systems and building up inventories in those systems should be addressed in future studies. If necessary these problems can be reduced by continually venting the fuel to the rad-waste system in a manner similar to that used in GCFR's Pressure Equalization System (PES) (Ref. 7-5).

The circulating activity in the HTGR is 10^2 larger than that of the MHFTR. Analysis has shown that releasing these fission products to the containment building does not endanger public health. Other analysis has shown that even if all the circulating activity in the HTGR were released

directly to the atmosphere that the off-site doses would be well within 10CFR100 guidelines (Refs. 7-11, 7-12). Because the MHFFR circulating inventory is much less than the HTGR inventory, it can be assumed that the release of the circulating inventory alone is not a significant problem.

The effects of releasing the fission products in the fuel with defective clad or the release of shutdown due to fuel melting is of much greater significance. In Ref. 7-3, it was shown that the release of fission products from fuel with defective cladding during a depressurization accident resulted in a maximum off-site dose of less than 0.5 rem. A similar dose should result from an MHFFR depressurization accident. Under all circumstances this dose is acceptable under the regulations of 10CFR100.

In both the GCFR and MHFFR analysis, it is assumed the blanket cooling systems can keep blanket temperatures below the fuel melting point. Therefore, no plutonium is released to the containment. In the analysis of the GCFR, the NRC postulated that a portion of the core did melt during a depressurization accident, 100 Kg of Pu in the form of an aerosol was released to the containment. The results of site doses were calculated to exceed 10^3 rem. Consequently, in the MHFFR it is of paramount importance to insure the adequate cooling of the blanket at all times.

Preliminary assessment indicates that the continued cooling of the blanket can be provided with the same high reliability of the GCFR cooling system. This assures that the potential for Pu release is adequately minimized. Given this fact, it appears that all accidents in the MHFFR will, at most, have a consequence comparable to those experienced in an HTGR or GCFR for a similar accident. Therefore, the MHFFR can be designed in such a manner that the health and safety of the public are insured.

7.2.9 Conclusions and Recommendations

A preliminary assessment of the MHFFR blanket and power conversion system indicates that an adequately safe design is technically feasible. However, considerably more analysis than that described here will have to be performed if an optimized and readily licensable design is to be developed.

To point the direction toward a design that is optimum from a safety and economic point of view, there are several areas that should be studied:

1. The cost/benefits of 3 versus 2 main/auxiliary loops per quadrant should be analyzed. Also, other forms of auxiliary cooling should be considered, e.g., direct water cooling, conduction cooling via heat sinks in the blanket, sharing of main/auxiliary loop components and/or systems.
2. The potential for and effects of reactivity insertions, fuel melting with the release of Pu aerosols and the potential for criticality occurring during a major blanket meltdown should be evaluated. It is anticipated that studies will show there can never be a criticality accident in the MHFFR despite large scale fuel melting. This is based on the expectation that only low concentrations of Pu will be present in the blanket.
3. Depressurization accident design criteria should be developed, e.g. hole size, flow limiter design/feasibility, and induced forces on critical components.
4. The effects of fission products and plate-out in the helium purification and tritium recovery system due to clad defects should be estimated. Inservice inspection and maintenance could be significantly affected. The cost/benefits of vented fuel should be considered.

5. During refueling, large, bulky, relatively unshielded blanket assemblies must be replaced in a complex manner. The potential for and consequences of accidents during fuel handling should be properly assessed so that appropriate design changes can minimize any radiological risks.
6. During any major accident there is a possibility for the fusion-related equipment to interact with the blanket-related equipment synergistically to produce a more severe accident. This potential problem should be addressed in future efforts.
7. Failures in the blanket ducting may create flow paths which cause the blanket inlet helium to bypass the fuel. This could prevent adequate blanket cooling even after the plant shuts down. This problem is highly dependent on the blanket design. The level of detail of the blanket design presented in this report would not permit a meaningful evaluation of this potential problem at this time. If found necessary, these bypass flow paths could be adequately limited by using independent inlet and outlet ducts to each module (rather than concentric plenums) or by insuring the bypass flow generated in a failure is limited to a small fraction of the blanket cooling flow.

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8. COST EVALUATION

An important aspect of the technical assessment of a design concept is the economic feasibility. Cost estimates have been made for the blanket modules and for the nuclear steam supply system of the MHFRR. These estimates are of necessity crude due to the preliminary nature of this study. Both the fuel and NSSS, however, are generically related to the components in General Atomic's gas cooled fast reactors which provided a comparative basis for estimating the cost of the blanket and NSS components.

8.1 BLANKET COST EVALUATION

The entire blanket structure is replaced when the blanket fuel is reloaded. As a consequence, the total blanket cost is treated as a fuel cycle cost rather than as a capital expenditure.

The cost estimate is based on certain unit cost assumptions:

U ₃ Si material cost	\$30/Kg
Rod hardware cost	\$ 7/rod
Pelletizing cost	\$ 6/rod

Based on these assumptions the cost estimates shown on Table 8.1-1 were derived. The initial blanket cost is \$94 M. One-quarter of the blanket is replaced at each reload interval. The reload blanket cost, then, is \$23.5 M.

TABLE 8.1-1
COST ESTIMATES FOR THE MHFTR BLANKET
U₃Si Fuel - 16 Modules

Number of rods	1.3 x 10 ⁶
Basic fuel cost (M\$)	23
Blanket rod hardware cost (M\$)	9
Pelletizing cost (M\$)	8
Rod costs (M\$)	40
Blanket structure cost (M\$)	44
Total blanket assembly cost (M\$)	94

8.2 SYSTEM COST EVALUATION

A very preliminary conceptual cost estimate was made for the nuclear steam supply (NSS) for the 1574 MW(e) (gross) Mirror Hybrid Fusion-Fission Reactor. Included in the estimate is the primary heat transfer system, the auxiliary heat transfer system, the refueling equipment allowance and special instrumentation and controls associated with the primary coolant system. The cost estimate specifically excludes the Nuclear Plasma Island, fuel and the NSS interfacing system that are normally included in the balance of plant (BOP).

The principal results are given in Table 8.2-1 and include project engineering costs, indirect costs, contractual risks and fees in terms of present day dollars but without interest during construction, or escalation. Also, the estimate is for an "equilibrium" commercial plant; i.e., the first-of-a-kind engineering and development costs are assumed to have already been recovered by the sale of earlier plants and this is a subsequent installation (for example this might be the tenth plant).

The estimated cost is \$295,000,000 for these systems or \$187 per KWe (gross) for this portion of the plant. Due to the large power requirement of the neutral beam injectors the net MHFFR output is 889 MWe. The NSSS cost, based on net electrical output is \$332/KWe.

A more complete listing of the equipment and systems included in this NSS scope of supply is given in Table 8.2-2. The systems that are not included in the cost estimate are listed in Table 8.2-3 (Nuclear Plasma Island) and Table 8.2-4 (Balance of Plant).

TABLE 8.2-1
COST ESTIMATES FOR THE 1574 MW_e MHFRR
July 1975 Dollars

NSS Scope of supply ^(a) -Equilibrium plant (Million of \$)		
Account ^(b)	Title	Cost ^(c)
222	Primary heat transfer system	\$189
223	Auxiliary heat transfer system	33
225	Refueling equipment allowance	63
227	Instrumentation and controls	<u>10</u>
	Total	\$295 M

(a) Excluding Nuclear Plasma Island, NSS interface systems in BOP.

(b) NUS code of accounts to numbering system.

(c) In addition to these totals for the NSS Scope of Supply, additional contributions will be added to these accounts by the BOP scope of supply.

TABLE 8.2-2
LIST OF SYSTEMS IN NSS SCOPE OF SUPPLY

Account	
222	Primary Heat Transfer System Helium pressure vessels and pipes Piping insulation Steam generators Main helium circulator and valves Main circulator service system Helium purification system
223	Auxiliary Heat Transfer System Auxiliary heat exchangers Auxiliary circulators, controllers and valves Auxiliary circulator service systems
225	Refueling Equipment Refueling equipment allowance
227	Instrumentation and Controls Primary coolant instrumentation Nuclear and in-blanket instrumentation Plant protection system Operational plant control system Fuel handling control station Radiation monitoring system NSS data acquisition and processing system Fluids regulating equipment Radioactive gas waste system

TABLE 8.2-3
LIST OF SYSTEMS IN NUCLEAR PLASMA ISLAND
(Not Included In NSS Cost Estimate)

Magnet

Injectors

Shielding

Direct Energy Convertor

Vacuum System

Tritium System

Diagnostics

Plasma Startup Equipment

Taken from UCRL-51797 but deleting blanket modules and refueling equipment.

TABLE 8.2-4
BOP SYSTEMS WITH NSS REQUIREMENTS
Not Included In NSS Scope Of Supply

System	
2401	Helium Storage System
2501	Nitrogen System
2601	Feedwater and Condensate System
2701	Water Treatment System
4201	Service Water System
4601	Plant Cooling Water System
4801	Spent Module Storage Cooling System
5101	Turbine Generator and Controls
5201	Turbine Steam System
6101	Decontamination System
6201	Radioactive Liquid Waste System
6501	Radioactive Solid Waste System
7202	Reactor Containment Building Crane
7205	Reactor Containment Building
7301	Ventilation System
7400	Reactor Service Building
7401	Reactor Service Building Crane
7402	Reactor Service Building Ventilation
8201	Instrument and Service Air System
8401	Auxiliary Steam Supply System
8601	Hydraulic Power System
8701	Auxiliary Service System
9201	NSSS Class 1E Standby Power Requirements
9202	NSSS Uninterruptible Power Requirements
9204	NSSS Electrical Power Systems and Equipment Requirements
9205	NSSS Electrical Equipment AC Power Requirements
9207	NSSS Grounding, Shielding and Electrical Noise

The largest contributors to the NSS cost are: (1) the helium piping and pressure vessels (~30%), (2) the steam generators (~25%), (3) fuel handling equipment (~20%), and (4) core auxiliary cooling loops (10%). The most significant cost uncertainties are those for the helium piping and for the fuel handling equipment.

The primary coolant pressure boundary (helium piping) may have potential for cost reduction through several possible approaches: (1) reduction of piping lengths through evaluation of alternate plant arrangements; (2) concentric ducting arrangements limiting the pressure-containing outside wall to the reactor inlet temperature and eliminating the high temperature pressure boundary, and (3) evaluation of partially integrated loop arrangements using pre-stressed concrete reactor vessel (PCRVR) technology.

Fuel handling equipment costs were not estimated from a design concept but instead an arbitrary allowance was made based on comparison with fission reactor systems of the weight and size of fuel modules and a subjective judgement made of the associated degree of difficulty and is included here only for completeness. Depending upon the refueling concept chosen, the final cost could differ by a large factor.

Additional cost reduction potential may exist for the auxiliary cooling system. These loops may be reduced in cost by reducing the level of redundancy assumed through a further study of the safety requirements, philosophy, and through enhancement of the possibilities for natural convection cooling.

9. CONCLUSIONS AND RECOMMENDATIONS

9.1 CONCLUSIONS

The purpose of the work reported here was to assess the feasibility of applying gas-cooled fission reactor technology to the mirror hybrid fusion-fission reactor. The conceptual design of the blanket and power conversion system presented in this report are directly derived from General Atomic's design experience with the High Temperature Gas-Cooled Reactor and the Gas Cooled Fast Reactor. From the analysis and discussion of the conceptual design that was presented, we have concluded that the hybrid blanket and power conversion system concepts are technically feasible. Existing gas-cooled fission reactor technology is directly applicable to the mirror hybrid reactor. There are a number of aspects of the present conceptual design that require further design work, analysis and optimization. There are several aspects of the present design that may present problems. Nevertheless, in general, the design appears feasible.

There appear to be no inherent features of the hybrid concept that present fundamentally new safety considerations to the reactor design. Further work is needed, however, in the area of helium ducting design, support and containment to assure that the reactor will be adequately cooled under all circumstances. The apparent total absence of potential accidents that could cause the system to become supercritical does offer the hybrid reactor some advantages compared to fission reactor systems. If the maintenance of forced cooling can be adequately assured, the hybrid

reactor could be licensed under the current Nuclear Regulatory Commission fission reactor regulations.

More detailed discussion is presented below for the various aspects of the hybrid design.

9.1.1 System Parameters

The temperature and pressure parameters selected for use in the conceptual design are typical of those used in gas-cooled fission reactor systems. System components have already been developed to operate in this pressure and temperature environment. The hybrid reactor concept appears to make no unique demands on the system parameter range; acceptable power conversion system performance is obtained using conventional temperature and pressure choices. The system operating parameters have not been optimized. Further improvement in the overall system performance and economics may be possible when this optimization is done.

9.1.2 Fuel Design

The best fuel design choice for the hybrid appears to be Inconel 718-clad U_3Si rods arranged in a close-packed array parallel to the radial coolant flow. This fuel configuration is similar to that used in the Gas Cooled Fast Reactor. The rods are shorter and larger in diameter than those used in the GCFR core. The power density and the maximum burnup are smaller. Because of this similarity and the less demanding environment, the fuel design appears to be quite feasible.

Further analysis is required to fully assess fuel performance under irradiation. Materials behavior in a fusion neutron environment is not well understood. Radiation-induced creep in the cantilevered fuel rods

must be considered as well as swelling, loss of ductility and other radiation-induced material damage.

9.1.3 Module Design

The module concept used in the hybrid reactor was proposed by Lawrence Livermore Laboratory. The design based on this concept that is presented in this report appears to be feasible although it must be emphasised that this design is quite preliminary and that further analysis may raise further questions. Although massive, the modules can be constructed with existing techniques. Inconel has not been qualified for a welding material in nuclear systems but is expected to be qualified before a hybrid reactor would be built.

A number of practical considerations remain to be investigated. Support of the massive modules has not been studied. The manufacturing tolerances required to achieve acceptable fit of the 16 modules into the spherical blanket shell need to be considered. Distortion due to thermal expansion, flow-induced vibration, or radiation-induced dimensional changes will have to be accounted for. The interface between the module and the helium ducting needs further study to select the best concept. There are several alternative module design concepts that should be investigated. Consideration should be given to reusing the outer structural portion of the module when the fuel is reloaded.

9.1.4 Power Conversion System Design

The power conversion system used in this design study follows conventional gas cooled reactor technology and relies heavily on General Atomic's experience with the HTGR and GCFR systems. The design appears feasible

although the helium ducting design needs further study. The method of primary system support and containment needs to be studied further. The pressure and temperature characteristics of the power conversion system need to be optimized specifically for the hybrid reactor.

9.1.5 Reactor Control System

The control system used in the reference design is a conventional gas-cooled reactor design. There appear to be no controllability problems, although there are some areas of uncertainty. The minimum thermal power at which the power conversion system might be able to produce electricity is expected to be about 15 to 20% of full power. This means that the neutral beam injectors must operate on off-site or auxiliary power at up to 20% of their full input power or 200 MWe. The possibility of getting power from the direct converters at very low load could reduce this off-site or auxiliary load demand and should be studied.

The control of the neutral beam injectors requires further study. It is important that these injectors be able to operate in a continuous manner over the range from 0 to 100% power. The ability to remain at part load indefinitely must be provided. These capabilities are required to allow controlled startup of the hybrid reactor.

9.1.6 Safety

The safety and licensing considerations for the gas-cooled hybrid reactor concept are very similar to those of gas-cooled fission reactors except for the apparent lack of any problems related to criticality. This potential for elimination of any criticality hazards relieves the necessity of having two redundant safety grade active reactor shutdown systems.

A total loss of forced cooling in the MHFFR will almost surely lead to melting of the blanket fuel. It maybe possible that melting of the blanket could be tolerated without undue hazard to the public, but the licensability of such a plant would be quite uncertain. To avoid this, the maintenance of adequate cooling must be shown to be assured with a high degree of confidence. Further design and anlysis of the module, helium ducting and primary pressure boundary needs to be done to assure that adequate cooling is always maintained.

From a licensing point of view the lack of criticality hazard is a definite advantage. Detailed design and anlysis of the coolant ducting system will have to be done in the future to ensure that licensing requirements for cooling integrity can be met. Assuming this can be done, we expect that the MHFFR could be licensed under existing Nuclear Regulatory Commission rules.

9.1.7 Cost

The \$332/kW_e cost estimated for the MHFFR nuclear steam supply system is high compared to those for fission reactors. This cost, however, does not account for the production of plutonium which is the main product of the hybrid reactor. Further, cost was not an important consideration in this design. The identification of the large cost components of the hybrid system will allow cost reduction to be done in the future.

9.2 RECOMMENDATIONS

Although the Mirror Hybrid Fusion-Fission Reactor appears to be a feasible concept, several areas require further investigation.

1. The selection of the system temperature and pressure parameters needs to be studied to see if the reactor performance can be optimized.
2. The method of support and containment of the primary coolant loop of a hybrid system needs to be investigated further. Licensing considerations particularly need to be addressed.

In addition to these general hybrid reactor considerations, there are items of specific interest to the mirror hybrid reactor that should be studied further.

3. The size and shape of the blanket modules should be considered; potential alternatives need to be investigated.
4. The support and alignment of the modules in the reactor during operation needs to be studied.
5. The interface between the blanket modules and the helium ducting especially needs attention due to the space restrictions imposed by the Yin-Yang magnet coils.
6. The safety and licensing aspects of potential depressurization accidents need to be considered as do design concepts to eliminate or mitigate these accidents.

With the results of these studies in hand, an integrated "reference design" could be made that would allow the feasibility and practicality of the mirror hybrid design concept to be more fully assessed.

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