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HTGR APPLICATIONS PROGRAM ADVANCED SYSTEMS

Semiannual Report for the Period
October 1, 1982 through March 31, 1983

by
Project Staff

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for the San Francisco Operations Office
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ABSTRACT

This application report is a topical summary of the program on high-temperature gas-cooled reactor (HTGR) advanced systems during the reporting period.

Work Breakdown Structure (WBS 41) activities emphasize the advanced HTGR modular reactor system (MRS) for reformer (R) and steam cycle/-cogeneration (SC/C) applications. This report describes progress in system performance for a 250-MW(t) MRS-R and a 300-MW(t) MRS-SC/C plant; it details the groundrules and parameters for the FY-83 nuclear core design and examines and compares fuel cycle economics.

This report gives results from a study on decay heat removal transients for the MRS-R and MRS-SC/C variants. It evaluates the bypass valve system and the number and location of helium circulators, and it describes the progress on circulator component design, a prestressed concrete vessel steel closure design, and plant licensing and safety.

Under the Advanced Technology Transfer Task (WBS 15), this report includes a section on a pebble bed reactor (PBR) MRS core heatup thermal model analysis.

This report also gives the results of a survey on candidate reformer tube materials from GA Technologies Inc. to identify acceptable substitute materials for Inconel 617 to alleviate possible cobalt activation and carburization problems.

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1. INTRODUCTION AND SUMMARY

The advanced systems activities in FY-83 are a continuation of earlier concept studies on the modular reactor system (MRS). Work within the MRS task focuses primarily on the direct-cycle reformer (R) variant of the high-temperature gas-cooled reactor (HTGR), with an incremental effort to update the design and cost of the steam cycle/cogeneration (SC/C) concept. This work is being performed as part of the Task Work Breakdown Structure (WBS 41) HTGR Advanced Systems Concept Evaluation Plan. Ultimately, input will be made to the final report at the end of FY-83.

Work is directed towards establishing the technical, economic, safety, and licensing feasibility of the MRS. The MRS study is an effort by many organizations under Task WBS 41. GA Technologies Inc. (GA--formerly General Atomic Company) has responsibilities in the following areas:

1. Nuclear core analysis/design.
2. Reactor internals design.
3. Systems performance/review.
4. Safety-related studies.
5. Heat removal studies.
6. Licensing support.
7. Circulator design.
8. Mechanical design of refueling approach and control rod drives.
9. Prestressed concrete reactor vessel (PCRV) concept design study.
10. Evaluation of alternate GA alloys for reformer tubes.

The GA work scope, under the advanced Technology Transfer Task (WBS 15) also involves the pebble bed reactor (PBR) for the MRS, including reviewing the monitoring work on the MRS being performed in Germany and conducting (1) a PBR fuel cycle cost estimate and (2) a PBR core heatup and fission product release analysis in FY-83.

In FY-83, the MRS work is being performed in the following three stages:

Phase I, design optimization and trade-off studies (Oct. 1 through Feb. 15).

Phase II, reference design (Feb. 15 through July 30).

Phase III, design evaluation (July 30 through Sept. 30).

The current reporting period includes documentation of the analytical and design tasks performed (in the areas of GA responsibility) in support of the optimization and trade-off studies. Work is also included in support of establishing a FY-83 reference design.

2. SYSTEMS PERFORMANCE/REVIEW

2.1. SCOPE

The work scope under this task includes the following:

1. Perform a limited parametric study to evaluate the impact of primary system parameter variations on performance for both the MRS-SC/C and MRS-R plant variants.
2. Define in-service inspection (ISI) requirements for the reactor vessel internals system.

2.2. DISCUSSION

2.1.1. Parametric Study

A parametric study was performed to provide nuclear heat system (NHS) input to an overall MRS plant performance/cost trade study. The NHS primary system parameters studied included (1) helium operating pressure, (2) core inlet temperature, and (3) in the case of the MRS-R, the reformer-to-steam generator power split. Table 2-1 gives the range of parameter variation and identifies major parameters that were held constant.

Figures 2-1 and 2-2 show the reference heat balance diagrams (design point) for the MRS-SC/C and MRS-R plants, respectively. Figures 2-3 and 2-4 show how NHS output varies with operating pressure and core inlet temperature; Figs. 2-5 and 2-6 show the impact on heat exchanger helium differential temperature due to variation in reformer-to-steam generator power split (MRS-R only); and Tables 2-2 and 2-3 provide a qualitative assessment of the parameter variation impact on NHS performance and cost.

TABLE 2-1
RANGE OF PARAMETER VARIATION

Parameter	Plant			
	MRS-SC/C		MRS-R	
	From	To	From	To
Helium operating pressure, MPa (psia)	3.45 (500)	6.9 (1000)	2.1 (300)	6.9 (1000)
Core inlet temperature, °C (°F)	238 (460)	327 (620)	404 (760)	471 (880)
Reformer-to-steam generator power split	--	--	0.44	0.62
Core power, MW(t) (constant)	300	300	250	250
Reactor outlet temperature, °C (°F) (constant)	688 (1270)	688 (1270)	950 (1742)	950 (1742)

Fig. 2-1. NSSS heat balance diagram for the 300-MW(t) MRS-SC/C plant

Fig. 2-2. NSSS heat balance diagram for the 250-MW(t) MRS-R plant

Fig. 2-3. Total heat transferred as a function of helium operating pressure and core inlet temperature for the 300-MW(t) MRS-SC/C plant

Fig. 2-4. Total heat transferred as a function of helium operating pressure and core inlet temperature for the 250-MW(t) MRS-R plant

Fig. 2-5. Temperature drop across the reformer as a function of reformer plant power split and helium operating pressure on the 250-MW(t) MRS-R plant

Fig. 2-6. Temperature drop across the steam generator as a function of reformer plant power split and helium operating pressure on the 250-MW(t) MRS-R plant

TABLE 2-2
SUMMARY OF THE IMPACT OF THE PERFORMANCE AND COST OF THE 300-MW(t)
MRS-SC/C PLANT UNDER VARYING OPERATING CONDITIONS

Operating Condition	Impact on Plant Performance	Impact on Plant Cost
1. Decrease in helium system pressure at constant core differential temperature	Constant helium flow Increase in pumping power Increase in nuclear steam supply system (NSSS) output Increase in steam generator temperature drop Reduced natural circulation flow rate, but possibly lower peak vessel temperatures during natural circulation Increase in feedwater/steam flow Decrease in plant efficiency	Increase in circulator size Increase in steam generator size Pressure vessel wall thickness reduction offset by vessel size increase Decrease in overall plant output
2. Increase in helium system pressure at constant core differential temperature	Reversed impact on plant performance from that shown in operating condition 1	Reversed impact on plant cost from that shown in operating condition 1
3. Increase in core temperature drop at constant helium system pressure	Decrease in helium flow Decrease in pumping power Decrease in NSSS output Increase in steam generator temperature drop Increased natural circulation flow rate, but possibly higher peak vessel temperatures during natural circulation Decrease in feedwater/steam flow Increase in plant efficiency	Decrease in circulator size Increase in steam generator size Increase in pressure vessel size due to increase in steam generator size Increase in overall plant output
4. Decrease in core temperature drop at constant helium system pressure	Reversed impact on plant performance from that shown in operating condition 3	Reversed impact on plant cost from that shown in operating condition 3

TABLE 2-3
SUMMARY OF THE IMPACT ON THE PERFORMANCE AND COST OF THE 250-MW(t) MRS-R PLANT
UNDER VARYING OPERATING CONDITIONS^(a)

Operating Condition	Impact on Plant Performance	Impact on Plant Cost
Increase the reformer-to-steam generator power split at constant operating pressure and constant core inlet temperature	Total nuclear steam supply system output constant Helium flow rate constant Reformer helium differential temperature larger Steam generator helium differential temperature smaller	Larger reformer size Smaller steam generator size Vessel size may increase to accommodate reformer and steam generator size changes
Decrease the reformer-to-steam generator power split at constant operating pressure and core inlet temperature	Total NSSS output constant Helium flow constant Reformer helium differential temperature smaller Steam generator helium differential temperature larger	Smaller reformer size Larger steam generator size Vessel assessment same as above

(a) Lowering the helium operating pressure will generally increase the sizes of the heat exchangers due to an increase in total NSSS power brought about by increased pumping power requirements. Lowering the core inlet temperature will also generally increase the heat exchanger sizes due to increased helium differential temperatures and thus lower log mean temperature differences.

Conclusions are inappropriate at this time, since the study results are input to an overall plant trade study.

2.2.2. In-Service Inspection Requirements

The MRS reactor vessel internals must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Division 2, "Rules for Inspection and Testing of Components of Gas-Cooled Plants." Since the MRS reactor internals are different than what this code was written for (e.g., metallic versus graphite core support structure), new portions of the code will probably need to be developed to meet the unique MRS requirements.

The ISI requirements for the steam generators, reformers, permanent reflector, core support, core barrel, and bypass valves were briefly outlined. The ISI requirements established at this time for the MRS are only preliminary, since ISI code interpretation is not straightforward and will ultimately be subject to state regulatory review.

The access required to perform these inspections will have to be demonstrated as the design proceeds, and material surveillance sample locations will also need to be identified.

3. NUCLEAR CORE ANALYSIS DESIGN

3.1. SCOPE

This task is intended to define the MRS fuel cycle and nuclear design parameters. Fuel cycle cost evaluations for the MRS cycle design options are also prepared under this task heading.

3.2. DISCUSSION

The major technical evaluations during this reporting period assessed the impact of cobalt activation in the MRS reformer and the development of the MRS fuel cycle and nuclear design updated groundrules.

3.2.1. Reformer Activation

The MRS reformer tubes are Inconel 617, which contains 12.5 wt % Co-59. Neutron captures during reactor operation lead to the formation of Co-60, which emits gammas upon decay. The gamma dose rate due to the Co-60 activation source was determined for a number of locations at the outer surface of the reactor vessel.

The maximum calculated dose was found to be directly below the reactor vessel flange. At the outer wall of the vessel, the dose was calculated to be 1410 MR/h after five years of operation and 2920 MR/h after 30 years of operation. This dose is about one thousand times the allowable limit for worker exposures.

As a result of this evaluation and because of concerns related to neutron damage doses to the side wall of the reactor vessel, boronated regions have been included in both end reflectors and in the radial permanent reflector. The preliminary design requirements are to attenuate

the core thermal neutron leakage by a factor of 500 to 100 in the upper reflector and by a factor of 100 or more from the radial and bottom reflectors. A boronated reflector design similar to that used in the large monolithic HTGR is expected to be adequate. Such a boronated reflector design does not change the reactor envelope dimensions.

In addition, alternate reformer materials have been evaluated, which do not contain cobalt, and additional analysis will be carried out on the most promising of these alloys.

3.2.2. Groundrules Development

Table 3-1 summarizes the groundrules for core nuclear design and fuel cycle economic evaluations for FY-83. The primary core and fuel cycle design has been changed from the previous reference batch reloading scheme; in this new graded reloading mode, half of the core is reloaded every two years (biennial) for the low-enriched uranium/thorium (LEU/Th) design. The reference high-enriched uranium/thorium (HEU/Th) design option utilizes half core reloading every three years (triennial).

These graded refueling modes reduce the fuel cycle cost 20% to 25% with LEU/Th fuel and 10% with HEU fuel from the previous reference batch refueling.

Figure 3-1 shows a typical fuel column distribution for the two ages of fuel for the graded refueling designs. Table 3-2 summarizes the fuel loading and reactor design parameters for the new reference designs. These are compared to the same design parameters for the previous reference batch-loaded 250-MW(t) design and to the current reference 2240-MW(t) MRS-SC/C design.

The much lower uranium loadings (C/U = 910 biennial batch versus C/U = 603 versus 250-MW(t) MRS with LEU fuel) improve the fuel cycle costs, increase control rod worths by ~25%, and greatly reduce the reactivity consequences of hydrogen or steam ingress.

TABLE 3-1
MRS CORE DESIGN GROUNDRULES UPDATE (FY-83)

Item	FY-83 Groundrule	Comment
Fuel cycle and economics	Four-year biennial, low-enriched uranium/thorium (half every two years) Six-year triennial, high-enriched uranium/thorium (HEU/Th) (half every three years)	20% reduction from four-year batch cycle 9% reduced from five-year batch cycle lower excess reactivity to control by boron compared to batch cycles
Cycle length	Four-year, LEU/Th Six-year, HEU/Th	Economic and fuel performance choice; no longer meshes with reformer schedule
Orifices	Fixed orifices (top reflector) changed at each refueling	Necessary for PH version
Core layout	85 column; eight blocks high	Graded cycle; radial zoning has increased core leakage by about 0.015 ΔK
Age distribution	To be determined. Three or four options under investigation	Maintain power shape stability and minimize fuel temperature
Radial power zoning	Radial zoning for reflector rod control over cycle	Lower peak fuel temperature during steady-state operation Higher radial neutron leakage to reflector control rods
Control rods	19 in-core, 12 reflector	Lighter uranium loading; radial zoning yields better reflector control and higher shutdown margins Evaluating impact on control philosophy (backup system)
Water flooding	Eliminate cold water flooding possibility	Graded cycle designs three times Less sensitive No work planned in FY-83

TABLE 3-1 (Continued)

Item	FY-83 Groundrule	Comment
Steam flooding	Assume batch core limit <118 kg (260 lb) in core	Graded cycle limit higher No work planned in FY-83
Reformer design	Alternate alloys to be evaluated Boronate top reflector region	Dose rate too high (>2900 MR/h)
Economic groundrules	Gas Cooled Reactor Associates FY-83 groundrules	Waste disposal, shipping, and storage costs to be updated from Fuel Technology Program Task

Fig. 3-1. Typical biennial design age distribution

TABLE 3-2
CYCLE CHARACTERISTICS COMPARISONS

MW(t)	Cycle	Power Density (W/cm ³)	Refueling Interval (yr)	Cycle Length (yr)	C/Th	C/U	APF(a,b)
2240	SC/C LEU-once-through (annual)	5.8	1.0	4.0	610	855	1.35
250	MRS-[process heat (EH)] LEU-once-through biennial ^(b)	4.1	2.0	4.0	700	910	1.15
300	MRS-SC/C LEU-once-through biennial ^(b)	4.9	2.0	4.0	875	760	1.17
250	MRS-PH LEU-once-through (batch)	4.1	4.0	4.0	600	603	1.00
2240	SC/C HEU recycle (annual)	7.1	1.0	4.0	230	2550	1.24
250	MRS-PH HEU recycle (triennial) ^(b)	4.1	3.0	5.0	225	2920	1.13
300	MRS-SC/C HEU recycle (triennial)	4.9	3.0	5.0	225	2440	1.15
250	MRS-PH HEU recycle (batch)	4.1	5.0	5.0	250	2970	1.00

(a) APF = age power peaking factor in fresh fuel.

(b) FY-83 reference designs.

A preliminary estimate of the MRS temperature coefficient of reactivity has been completed and is plotted in Fig. 3-2. This result is from a MICROX cell calculation and does not include neutron leakage effects yet. The overall coefficient is negative at all temperatures and is strongly negative at very high temperatures.

Fig. 3-2. Temperature coefficient for 250-MW(t) MRS-PH LEU four-year biennial

4. MRS DECAY HEAT REMOVAL STUDIES

4.1. SCOPE

This task is intended to determine the decay heat removal configuration and to evaluate transient performance for the MRS-SC/C and MRS-R plant concepts.

Specific topics to be addressed in this reporting period included the following:

1. MRS-R transients.
2. Elimination of the steam generator bypass valve.
3. MRS-SC/C transients.
4. Optimum circulator number and location.
5. Vessel cooling system (VCS) water storage and insulation requirements.

4.2. DISCUSSION

The decay heat removal configuration for both the MRS-R and MRS-SC/C consists of the following:

1. One main cooling system (MCS), nonsafety grade.
2. One redundant (two independent 100% loops) VCS, safety grade.

Figures 4-1 and 4-2 show the current reactor vessel designs for the MRS-R and MRS-SC/C, respectively. Figure 4-3 shows the VCS design. The MRS design configuration has one circulator with a main motor and an independently powered pony motor with 100% capacity for depressurized cooldown. Because only one circulator is used, no circulator check valve is required on the primary side. If forced circulation is not available under

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Fig. 4-1. MRS-R vessel arrangement

CONVERSION FACTOR

1 IN. = 25.4 MM
1 FT = 304.8 MM

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Fig. 4-2. MRS-SC/C vessel arrangement

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Fig. 4-3. VCS forced circulation ultimate water vaporization cooling

pressurized conditions, primary coolant natural circulation provides decay heat removal using either the MCS (steam generator) or VCS as a heat sink. A bypass valve located near the bottom of the steam generator allows the primary coolant to bypass the steam generator during VCS operation. Table 4-1 discusses the VCS design. Table 4-2 summarizes the various modes of decay heat removal in the MRS.

4.2.1. MRS-R Decay Heat Removal Transients

The MRS-R decay heat removal analysis has been updated to reflect the final 1982 design given in Ref. 4-1. Also included in the updated analysis are conservative factors, shown in Table 4-3, which are normally included to show that reactor design limits are met for licensing purposes. Table 4-4 summarizes key parameters for various decay heat transients. As can be seen, reactor trip and natural circulation using the steam generator as a heat sink are not severe transients. For these two transients, except for reformer tube temperature, the maximum value for all key parameters occurs at the design point.

The most important transient is natural circulation using the VCS since this transient is the most severe of those having a high probability of occurrence. This event determines whether the vessel temperature and pressure meet the design limits of 454°C (850°F) and 6.2 MPa (900 psia), respectively. The updated analysis, with conservative factors included, results in a maximum vessel temperature of 419°C (786°F) and a maximum vessel pressure of 5.63 MPa (816 psia).

In the current design, the reformer is isolated upon reactor trip. In the case of natural circulation using the VCS, isolation of the reformer results in a severe transient for the tubes at the reformer outlet (see Fig. 4-4). In addition, analysis performed by General Electric Company (GE) has shown an extreme temperature gradient across the reformer tube sheet for this transient. To reduce the severity of this transient, purging the reformer of process gas and catalyst has been proposed while allowing the process steam to continue flowing through the reformer to cool it. As

TABLE 4-1
SUMMARY OF VCS CIRCULATION CHARACTERISTICS

Advantages

16-day makeup for water lost by boil-off integral with primary cooling loop.
Steam collection in pressurizer by natural circulation during boil-off.

Disadvantages

2 to 8 day water storage tanks add large weight to confinement or auxiliary buildings.

Normal operation

Primary loops circulated by pumps. Heat rejected via air blast heat exchangers.

Failure of primary pumps/loss of air-blast cooling

Flow stops in primary loops and water in cavity starts to boil.
Steam rises to pressurizer and is vented through relief valves.
Check valves open and cavity water makeup supplied by natural circulation.

TABLE 4-2
HTGR-MRS MODES OF DECAY HEAT REMOVAL

MODE	CIRCULATOR				STEAM GENERATOR		HEAT REMOVAL SYSTEM
	MAIN MOTOR DRIVE		PONY MOTOR DRIVE				
	OPERATIONAL	NON-OPERATIONAL	OPERATIONAL	NON-OPERATIONAL	OPERATIONAL	NON-OPERATIONAL	
A. REACTOR SYSTEM PRESSURIZED							
1	✓				✓		FORCED CIRCULATION WITH MAIN LOOP COOLING THROUGH STEAM GENERATOR
2		✓	✓		✓		SAME AS ABOVE
3		✓		✓	✓		NATURAL CIRCULATION WITH HEAT REJECTION THROUGH STEAM GENERATOR
4		✓		✓		✓	NATURAL CIRCULATION IN REACTOR VESSEL, PASSIVE HEAT REJECTION BY RADIATION FROM REACTOR VESSEL WALL TO COOLING COILS
B. REACTOR SYSTEM DE-PRESSURIZED							
1	✓				✓		FORCED CIRCULATION WITH MAIN LOOP COOLING THROUGH STEAM GENERATOR
2		✓	✓		✓		SAME AS ABOVE
3		✓		✓		✓	RADIAL CONDUCTION FROM CORE TO LOWER VESSEL WALLS, PASSIVE HEAT REJECTION BY RADIATION FROM REACTOR VESSEL WALL TO COOLING COILS

TABLE 4-3
MRS CONSERVATIVE FACTORS FOR NATURAL CIRCULATION USING VCS

	FY-82 Nominal	Conservative ^(a)
Decay heat	Nominal	120%/110%
Differential pressure	Nominal	Nominal/1.2
T _{sink} , °C (°F)	38 (100)	93 (200)
Emissivity	0.9	0.85
Surface area	Includes support skirt area	Improved modeling of support skirt area
Vessel heat transfer coefficient	Nominal	1.1 x nominal
Vessel conductivity	Nominal	Nominal/1.1

(a) This case is conservative with respect to vessel temperature. Another set of parameters should be used to define a conservative case for helium pressure.

TABLE 4-4
MRS-R DECAY HEAT REMOVAL TRANSIENTS

Maximum Values ^(a)	Normal Operation	Reactor Trip	Natural Circulation with Steam Generator	Natural Circulation with VCS	Natural Circulation with VCS and Reformer Cooling
Helium pressure, MPa (psia)	4.63 (672)	4.63 (672)	4.63 (672)	5.63 (816)	4.8 (698)
Midwall vessel temperature, °C (°F)	382 (720)	382 (720)	382 (720)	419 (786)	382 (720)
Core inlet temperature, °C (°F)	427 (800)	427 (800)	427 (800)	540 (1004)	442 (828)
Reformer tube ^(b) temperature, °C (°F)	619 (1147)	895 (1643)	955 (1752)	982 (1800)	619 (1147)
Steam generator tube temperature, °C (°F)	534 (993)	534 (993)	534 (993)	977 (1790)	615 (1140)
Flow at 24 h, % of rated	--	15.0%	2.3% (at 10 h)	1.3%	1.5%

(a) Vessel design pressure, 6.2 MPa (900 psia); vessel design temperature, 454°C (850°F).

(b) Reformer tube temperature is given at the reformer outlet.

Fig. 4-4. Reformer outlet tube temperatures during VCS natural circulation

Fig. 4-4 shows, the reformer tubes do not reach a peak temperature for ~5 h. Therefore, even if the reformer must be initially isolated at reactor trip, ample time is available to purge the reformer and restore steam cooling.

An analysis was performed to assess the overall effect of cooling the reformer with steam during the natural circulation transient. This analysis assumes that heat is removed from the reformer at a rate sufficient to maintain the reformer tube temperatures at the design value. Figure 4-4 shows the impact of reformer cooling on the reformer tube transient. Table 4-3 summarizes the results of key parameters during the reformer cooling transient. Those initial calculations show that reformer cooling increases natural circulation flow and is beneficial in reducing reformer and steam generator tube temperatures and reactor vessel temperature and pressure. GA and GE are continuing to investigate reformer cooling. Barring unforeseen technical problems, reformer cooling should be incorporated as a reactor design feature, because it greatly reduces the severity of the natural circulation transient.

4.2.2. Bypass Valve Elimination

In the current MRS-R design, shown in Fig. 4-1, there is an annulus between the outer steam generator shroud and the reactor vessel wall. Flow passes through the steam generator during normal operation, and the annulus, which provides a parallel flow path, is closed off to flow by a bypass valve (except for a 1% leakage flow). Natural circulation using the VCS is the mode of decay heat removal if the steam generator is lost as a heat sink (regardless of whether the circulator is available). When the steam generator is lost as a heat sink, the bypass valve is opened to allow flow to bypass the steam generator, thus providing more reactor vessel surface area for transfer of decay heat from the coolant to vessel wall by natural circulation and from the vessel wall to the VCS via radiation. This additional heat transfer occurring in the top portion of the vessel also provides a greater head for natural circulation than if the valve were not opened.

Figure 4-5 shows the potential bypass valve configurations, both of which are mechanically actuated. Three ideas have been offered for eliminating the bypass valve from the MRS configuration:

1. Case 1. (Base case). Use a serpentine path (Fig. 4-5) with the bypass valve opened. With the addition of another annulus, the primary coolant flows down through the steam generator, up through the added annulus, then back down along the vessel wall. While this design continues to make full utilization of the upper reactor vessel surface area, at the expense of greatly complicating the flow path, this design has been eliminated from consideration because under certain circumstances it will lead to cold traps which could prevent or inhibit the development of primary coolant natural circulation.
2. Case 2. Eliminate the bypass valve. Maintain the current flow configuration, but completely close off the annulus between the outer steam generator shroud and the vessel wall by eliminating the bypass valve.
3. Case 3. Eliminate the bypass valve and add a flow skirt. A flow skirt provides a smaller annulus over a larger portion of the vessel between the top of the core and the bottom of the steam generator to make more effective use of this portion of the vessel for heat transfer (Fig. 4-5).

Studies were performed based on the latter two concepts to determine if the steam generator bypass valve can be eliminated. Table 4-5 gives a comparison between three cases of natural circulation decay heat removal using the VCS.

Fig. 4-5. MRS bypass valve alternatives

TABLE 4-5
NATURAL CIRCULATION DECAY HEAT REMOVAL FOR MRS-R USING VCS

	Maximums(a)		
	Vessel Pressure ^(b) [MPa (psia)]	Vessel Temperature [°C (°F)]	Core Inlet Temperature [°C (°F)]
Conservative base	5.63 (816)	419 (786)	540 (1004)
Without bypass valve	6.16 (894)	442 (828)	596 (1105)
Without bypass valve with flow skirt	5.91 (857)	449 (840)	559 (1039)

(a) Vessel design pressure, 6.2 MPa (900 psia); vessel design temperature, 454°C (850°F).

(b) These analyses are not conservative with respect to pressure.

The base case maximum vessel temperature of 419°C (786°F) and the maximum vessel pressure of 5.63 MPa (816 psia) offer large margins when compared to the vessel design temperature limit of 454°C (850°F) and the design pressure limit of 6.2 MPa (900 psia). The maximum vessel pressure of 6.14 MPa (890 psia) for Case 2 and the maximum vessel temperature of 449°C (840°F) for Case 3 come very close to the respective design limits.

In view of the marginally acceptable results obtained if the bypass valve is eliminated, GA recommends keeping the bypass valve at this conceptual stage of the MRS design. However, GA strongly favors an effort to develop a passive bypass valve design. Such a design was developed in earlier GA privately funded studies of the MRS-SC/C where the circulator head holds multiple bypass valves closed during normal operation and the valves are opened by gravity when the circulator fails (see Fig. 4-6). However, recent studies have shown that the circulator temperature transient is unacceptable unless the circulator is located at the bottom of the vessel. Therefore, the valve design shown in Fig. 4-6 is unworkable. Figure 4-5(e) shows a sketch of a similar concept of multiple gravity opening valves, which is compatible with a bottom-mounted circulator.

4.2.3. MRS-SC/C Decay Heat Removal

The MRS-SC/C decay heat removal analysis has been updated to include the same conservative factors (Table 4-2) that were included in the MRS-R analysis. Table 4-6 summarizes key parameters for various decay heat removal transients. The results are similar in nature to the MRS-R results. For reactor trip and natural circulation using the steam generator as a heat sink, the maximum value for all key parameters occurs at the design point. Again, natural circulation using the VCS is the critical transient which determines whether the vessel temperature and pressure meet the design limits. The MRS-SC/C uses light water reactor (LWR) vessel material (SA-533, Grade B, Class 1) and consequently has a lower vessel design temperature [371°C (700°F)], but higher vessel design pressure [9.3 MPa (1350 psia)] than the MRS-R which uses high-temperature materials. The maximum vessel

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Fig. 4-6. MRS-SC/C with midplane circulator location

TABLE 4-6
MRS-SC/C DECAY HEAT REMOVAL TRANSIENTS

Maximum Values ^(a)	Normal Operation	Reactor Trip	Natural Circulation with Steam Generator	Natural Circulation with VCS
Helium pressure, MPa (psia)	5.0 (725)	5.0 (725)	5.0 (725)	7.45 (1081)
Midwall vessel temperature, °C (°F)	285 (545)	285 (545)	285 (545)	3566 (674)
Core inlet temperature, °C (°F)	284 (543)	284 (54)	284 (543)	610 (1131)
Steam generator tube temperature, °C (°F)	558 (1037)	558 (1037)	558 (1037)	850 (1562)
Flow at 24 h, % of rated	--	15.0%	2.9% (at 10 h)	1.4%

(a) Vessel design pressure, 9.13 MPa (1350 psia); vessel design temperature, 371°C (700°F).

pressure and temperature during the natural circulation transient are 357°C (674°F) and 7.45 MPa (1081 psia), respectively.

4.2.4. Number of Circulators

If the single circulator in the MRS fails under pressurized conditions, natural circulation provides core cooling. However, under depressurized conditions, natural circulation is not available. This gives rise to concerns about safety, licensing, investment risk, and availability if the circulator fails under depressurized conditions. If the circulator fails under depressurized conditions but the VCS is available, vessel temperatures will remain acceptable; however, the core will heat up to 1982°C (3600°F) causing fuel particle failure and a fission product release on the order of 1% in the MRS-R. From the safety standpoint, the probability and consequences of either a depressurization accident scenario or a circulator failure during a planned reactor depressurization (such as refueling) leading to a core heatup appear acceptably small. This probability is improved further if the reactor is designed so that it can be repressurized to provide natural circulation if the circulator fails during refueling.

The addition of either a main circulator or a small auxiliary circulator, with reduced power and flow requirements, was considered as a design alternative. However, this adds additional complexity and unreliability to the system, because it requires each parallel circulator to have an isolation valve to prevent flow from going through the circulator and bypassing the core when the circulator is not operating. From a safety perspective, additional circulators offer no appreciable advantage in reducing the core heatup frequency and are therefore not recommended (Ref. 4-2).

The major problem associated with having a single circulator is availability (i.e., what happens if the circulator fails and must be removed for repair?). How many days must the system remain pressurized before the circulator can be safely removed? The proposed circulator maintenance procedures are as follows:

1. System remains pressurized (buttoned-up) for 12.5 days. Cool down with natural circulation using either the steam generator or VCS as a heat sink.
2. Depressurize the reactor. Based on Ref. 4-3 analysis, the maximum fuel temperature will rise to the normal design point value of 1204°C (2200°F) four to five days after depressurization. The initial vessel temperature adjacent to the circulator is 177°C (350°F) (less if the steam generator is operating) and is not expected to rise much following depressurization.
3. Remove circulator.
4. Install a new circulator.

A single circulator is the best design achievable in the MRS. This design is the simplest, most cost effective, and arguably meets licensing requirements. At this conceptual stage of the MRS design, GA recommends maintaining a single circulator as the reference design. As a backup, an additional small auxiliary circulator can be added to the design if required by licensing or other considerations. Adopting a single circulator design dictates additional design requirements for the MRS: (1) the reactor must be designed so that it can be repressurized if the circulator fails under refueling conditions and (2) the circulator must be designed to be removed under the depressurized scenario described above. Based on discussions with component designers, both of these requirements are believed to be feasible.

4.2.5. Circulator Location

Two circulator positions were evaluated: (1) the current design, with the circulator positioned horizontally in the lower region of the reactor vessel (see Figs. 4-1 and 4-2) and (2) an alternative, with the circulator

positioned horizontally at the reactor vessel midplane (see Fig. 4-6). Neither position appears to have a major advantage or impact on refueling or circulator maintenance. If the circulator is located at the vessel midplane, the lower plenum length can be shortened, resulting in a reduced control rod length and a slight reduction in overall vessel length. With a midplane location, the mechanically actuated steam generator bypass valve (Fig. 4-5) can be replaced by a passive valve, as shown in Fig. 4-6.

The passive valve is held in place by the circulator flow pressure forces. When the circulator fails, it automatically opens, allowing flow adjacent to upper reactor vessel; this occurs in the current design when the mechanically actuated valve is opened. In the current design, if the steam generator bypass valve is inadvertently opened at full power, a large amount of hot gas will flow past the upper reactor vessel, and unacceptably high reactor vessel temperatures will develop in a short time. The main advantage of the midplane location is that the potential for adverse consequences from this accident can be eliminated; if the passive valve inadvertently opens at full power, cold (rather than hot) gas is swept along the upper reactor vessel. However, during natural circulation decay heat removal, the midplane circulator is subjected to hot gas; this hot gas exits the annulus adjacent to the upper reactor vessel, and even hotter gas will flow through a nonoperating steam generator until the entire steam generator reaches essentially core outlet temperature and stagnates. When the circulator is positioned at the lower region of the vessel, it sees cold core inlet temperature during both power operation and natural circulation decay heat removal. The maximum allowable circulator temperature is on the order of 537° to 593°C (1000° to 1100°F). Therefore, the high gas temperatures [704°C (~1300°F) in the MRS-SC/C and 927°C (~1700°F) in the MRS-R] that a circulator located at the midplane would see during a natural circulation decay heat removal transient are unacceptable.

The circulator should be maintained in its current position at the bottom of the reactor vessel, as shown in Figs. 4-1 and 4-2.

4.2.6. VCS Water Storage Requirements

The notes on Fig. 4-3 describe the current VCS design. The amount of water required to be stored in the VCS tanks is determined by the desire to minimize the potential for core or pressure boundary (i.e., reactor vessel) damage due to the loss of VCS cooling capability. If a shutdown MRS-R is cooled for 16 days and the VCS is then lost, neither the core nor the reactor vessel will incur damage (Ref. 4-3). The VCS is passive and highly reliable; its failure rate is dominated by common-mode failures. Therefore, the required 16-day (280 MW-h) water supply can be adequately met by an eight-day water supply in each loop.

4.2.7. VCS Insulation

The maximum allowable concrete temperature for the biological shield is 74°C (165°F), unless insulation is placed between the VCS and the biological shield. With insulation, the VCS outlet temperature can be increased; this is beneficial, because (1) it reduces the amount of water flow required to reject a given heat load, (2) it decreases the required surface area for the air cooled heat exchanger in the VCS loop (see Fig. 4-3), and (3) it enhances both the natural circulation and heat storage potential of the VCS water. Indeed, the current VCS design, where stored water is boiled off as a last resort, would result in concrete damage unless the biological shield is insulated.

The scoping study was also performed to determine if a higher average VCS water temperature would be detrimental to VCS decay heat removal capability. The results, which are summarized in Table 4-7, show that the maximum vessel temperature during natural circulation is not so sensitive to average water temperature that it restricts the VCS water outlet temperature to 74°C (165°F). GA has recommended to Bechtel that insulation be incorporated in the design and that the maximum VCS water temperature be restricted to 149°C (300°F).

TABLE 4-7
VCS NATURAL CIRCULATION(a)

Average VCS Water Temperature [°C (°F)]	Maximum Vessel Temperature [°C (°F)]	Vessel Pressure [MPa (psia)]	VCS Heat Load (MW)
38 (100)	388 (730)	5.82 (844)	3.17
93 (200)	398 (748)	5.84 (847)	2.98
131 (300)	410 (771)	5.86 (850)	2.74
204 (400)	426 (799)	5.9 (856)	2.43

(a) These analyses use old data and are presented to illustrate sensitivity only.

4.3. REFERENCES

- 4-1. "Modular Reactor System (MRS-PH) Plant Concept Design Status Report," Gas-Cooled Reactor Associates Report GCRA-CE-108-101, v. 1, September 1982.
- 4-2. Everline, C., "Assessment of the MRS-R with Circulator Reliability Enhanced," GA unpublished data, January 28, 1983.
- 4-3. Parme, L., "Mini-PRA for the 250 MW(t) MRS-PH," GA unpublished data, July 12, 1982.

5. SAFETY-RELATED STUDIES

5.1. SCOPE

This task is intended to perform safety and investment risk related studies for the MRS. For this reporting period, the work scope included core heatup frequency and consequence assessments for the MRS-R plant and focused upon the following:

1. Update the FY-82 scoping Safety Risk Study.
2. Compare the core heatup accident frequencies for single module and eight module sites.
3. Evaluate the impact of helium circulator reliability enhancement on the core heatup accident frequency.
4. Analyze the impact of VCS design options on the core heatup accident frequency.
5. Perform a parametric study of depressurized core heatup consequences for various power and power density.

5.2. DISCUSSION

Although a single module site was analyzed in FY-82, a revised assessment was considered warranted to reflect subsequent design changes and apply more detailed probabilistic models. The result of this evaluation is that the median core heatup frequency of 5×10^{-5} /site-year obtained for a single site in the FY-82 study rose to 9×10^{-5} /site-year. This is rather

good agreement between a scoping study and a detailed evaluation, which indicates that the generic conclusions from the FY-82 analysis remain applicable.

Table 5-1 gives a core heatup frequency summary for single and eight-module sites. These assessments used a base plant design, which is predicated upon the reference conceptual design at the end of FY-82, except that the VCS has two instead of three replicate loops (see Fig. 5-1). The values cited in Table 5-1 are frequencies per site year. For a single module site, this is the frequency of core heatups in the module. At an eight-module site, however, this is the frequency at which one or more modules experience core heatup per year. A survey of Table 5-1 discloses that core heatup frequencies on an eight-module site are a factor of ~ 8 times higher than at a single module site. The salient reason for this difference is that eight modules require more core cooling equipment than a single module, which increases the overall failure probability. An important conclusion from Table 5-1 is that an eight-module site exceeds the 10^{-4} core heatup frequency goal, and that, although a single module site complies with this goal, it provides little margin to account for the design uncertainties inherent in a conceptual design probability risk assessment.

Table 5-2 summarizes the impact of circulator reliability enhancement. The assessments supporting Table 5-2 were performed in a bounding fashion by hypothesizing that the probability of losing forced helium circulation in any module was negligibly small. Even so, Table 5-2 demonstrates that the bounding hypothesis has an imperceptible impact on a single module site and only reduces the core heatup frequency for an eight-module site by a factor of ~ 2 . The primary reason for the greater impact on an eight-module site is that, with more modules, the VCS mission time is longer when the accident initiating event is a circulator failure. An important conclusion from Table 5-2 is that enhancing the circulator reliability will neither improve the single module margin relative to the 10^{-4} /site-year frequency goal nor bring an eight-module site into compliance with this goal. Therefore, future design activity oriented toward enhancing the circulator reliability is not recommended.

TABLE 5-1
BASE PLANT CORE HEATUP SUMMARY

Modules	Median Frequency (Per Site Year)			Total
	Loss of Main Loop Cooling	Steam Generator Leak	Depressurization	
1	7×10^{-5}	2×10^{-5}	3×10^{-7}	9×10^{-5}
8	5×10^{-4}	2×10^{-4}	3×10^{-6}	7×10^{-4}

Fig. 5-1. MRS-R cooling system (1/2)

TABLE 5-2
IMPACT OF CIRCULATOR RELIABILITY ENHANCEMENT

Median Frequency (Per Site Year)				
Modules	Loss of Main Loop Cooling	Steam Generator Leak	Depressurization	Total
1	7×10^{-5}	2×10^{-5}	1×10^{-7}	9×10^{-5}
8	1×10^{-4}	2×10^{-4}	1×10^{-6}	3×10^{-4}

Analysis of the impact of VCS design options on the core heatup frequency was conducted in two stages. The first stage focused on the incremental decrease furnished by installing a third, replicate loop in each VCS (Fig. 5-2). This, of course, is the FY-82 reference VCS design and was found to furnish negligible improvement over the base plant assessment (Table 5-1). This is because VCS common-mode failures are a primary contributor to core heatup, and these failures are dominated by blackout conditions that deprive the active VCS components of electrical power. Installing additional active components does little to assuage this problem. Qualitatively, this implies that any VCS design depending entirely upon active vessel cooling cannot significantly reduce the frequency of core heatup. For this reason, the internally diverse VCS illustrated in Fig. 5-3 was evaluated.

During normal plant operation, the internally diverse VCS relies on active vessel cooling components. However, this VCS concept is capable of both active and passive decay heat removal. In the passive decay heat removal mode, the cooling panels remain filled with water from elevated storage tanks. Heat radiated to the panels from the vessel causes boiling, and cooling is achieved by venting steam to the atmosphere through relief valves. Although both loops had a 30-day water inventory in the initial concept, only a total of 16 days is needed to prevent excessive core and reactor vessel temperatures.

Table 5-3 shows the impact of the internally diverse VCS on core heatup frequency. Comparison with Table 5-1 discloses a decrease of approximately two orders of magnitude, which results in a substantial compliance margin relative to the 10^{-4} core heatup frequency goal. Thus, this concept is strongly recommended to be incorporated into the MRS design. Further, protection should be provided for the cooling panels against missiles generated by steam or feedwater line ruptures inside the vessel cavity, since lack of missile protection can greatly jeopardize the safety advantages offered by the internally diverse VCS.

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Fig. 5-2. MRS-R cooling system (1/3)

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Fig. 5-3. 2 x 100% internally diverse loop vessel cooling system

TABLE 5-3
IMPACT OF THE 2 x 100% INTERNALLY DIVERSE LOOP VESSEL
COOLING SYSTEM

Median Frequency (Per Site Year)				
Modules	Loss of Main Loop Cooling	Steam Generator Leak	Depressurization	Total
1	7×10^{-7}	2×10^{-7}	3×10^{-7}	1×10^{-6}
8	5×10^{-6}	1×10^{-6}	3×10^{-6}	9×10^{-6}

One of the aspects that is considered by the participants in the MRS task is to reduce the core temperatures such that insignificant fuel failure would occur during a core heatup accident. Therefore, this task has directed efforts toward obtaining the sensitivity of the fuel temperature to both the core thermal rating and the power density for an accident scenario that assumes a core heatup accident with no prior cooldown, a depressurized vessel, and successful vessel cooling.

For selected core thermal ratings, the various power densities were obtained by varying the core radius and core height. The MRS version of CORCON, a two-dimensional heat transfer code, was modified to allow a variable core dimension. In this new CORCON model, the core and vessel dimension of the 250-MW(t) MRS-SC were employed as basic dimensions (see Fig. 5-4). When the core size was varied, material thicknesses outside the active core were maintained the same.

Core and component temperatures were calculated using the modified version of CORCON for various core thermal ratings and power densities. Figures 5-5 and 5-6 present the maximum core temperature versus power density for four power levels and two core heights. Each curve contains three data points which represent the specified power density and a particular core size. Figures 5-5 and 5-6 show that the maximum core temperature is reduced with similar sensitivity when either the power level or the power density is reduced. A fuel temperature of 1600°C (2912°F) is currently assumed to be a transition temperature above which the accident condition fuel failure model is applicable, and below 1600°C (2912°F), the normal operation fuel failure models are assumed to become more dominant. This transition temperature of 1600°C (2912°F) is shown in Figs. 5-5 and 5-6.

Figure 5-7 shows the maximum fission product release from the fuel particles as a function of core thermal rating for three different power densities. These curves were obtained using the existing data and an assumption of negligible release at 1600°C (2912°F). Figure 5-7 does not show the release data below the limiting release of normal operation.

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Fig. 5-4. CORCON nodal blocks for 250-MW(t) MRS-SC/C vessel and internals

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Fig. 5-5. Maximum core temperature of MRS during core heatup for core thermal rating of 50 and 100 MW(t)

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Fig. 5-6. Maximum core temperature of MRS during core heatup for core thermal rating of 100, 200, and 300 MW(t)

Fig. 5-7. Release of Kr-35 from fuel particles during MRS core heatup

These results indicate that lower power and power densities than the current levels for the MRS design are required to lower the core maximum temperature to the range [$<1600^{\circ}\text{C}$ (2912°F)] of normal operating fuel failure mechanisms.

6. PEBBLE BED CORE HEATUP ANALYSIS

6.1. SCOPE

A core heatup thermal model for the PBR core has been developed using the existing core heatup thermal and fission product release models for the MRS with a reactor core of prismatic fuel elements. Initial analyses have been made to check out the thermal model in preparation for the analysis and comparison of core heatup consequences for the modular pebble reactor (MPBR).

6.2., DISCUSSION

A core heatup thermal model was prepared for the MPBR based on the configuration of Ref. 6-1 as shown in Fig. 6-1. The effective thermal conductivity of the pebble bed core was treated according to Ref. 6-2. The effective thermal conductivity represents the combined effect of conduction through solid, contact resistance, and the stagnant gas and radiation through the void spaces. The equation of the thermal conductivity of the solid fuel ball was also obtained from Ref. 6-2. The effective heat capability was modified from the solid graphite heat capability by the bed void fraction. The decay heat generation was treated the same way as the 250-MW(t) modular reactor core with the prismatic elements using the existing HTGR decay heat curve (Ref. 6-3). The decay heat curves for the MPBR will be eventually used as and when the data are available. The fission product transport code developed for the prismatic design was modified to obtain only the fuel particle performance in terms of fuel failure and release of fission products and fuel particles during the core heatup accidents. The initial radial and axial power distributions were obtained from Ref. 6-1.

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Fig. 6-1. CORCON nodal blocks for the 250-MW(t) MPBR vessel and internals

Two test cases simulating a depressurized core heatup of the MPBR with vessel cooling available were studied. The first case treated the 250-MW(t) MPBR based on the same initial conditions as the GE study (Ref. 6-1), while the second test case attempted to duplicate the German study (Ref. 6-4) of a 170-MW(t) MPBR core heatup.

Figure 6-2 compares the core and vessel temperatures for the first test case with the GE study. The temperatures are generally in good agreement for the two studies. Figure 6-3 compares temperatures for the second test case with the German study. The maximum core temperatures are in good agreement for the two studies. The thermal model appears to give a reasonably good estimate of the MPBR core heatup consequence. However, this model needs to be, and will be, further improved by considering pebble-bed-specific data.

6.3. REFERENCES

- 6-1. Lutz, D. E., et al., "Modular Pebble-Bed Reactor Reforming Plant Design for Process Heat," General Electric Company Report GEFR-00630, September 1980.
- 6-2. Shamasundar, B. I., and M. E. Fehrenbach, "KUGEL - A Thermal, Hydraulic, Fuel Performance, and Gaseous Fission Product Release Code for Pebble Bed Reactor Core Analysis," DOE Report GA-A16222, May 1981, Section 2.
- 6-3. Sund, R. E., "Afterheat Calculations for the HTGR," GA Report Gulf-GA-A12499-A, July 1974.
- 6-4. "High Temperature Reactor Modular System for the Generation of Process Heat," GHT Report ITB78.2634.1, Germany, October 1981.

Fig. 6-2. Core and vessel temperatures for 250-MW(t) MPBR core heatup

Fig. 6-3. Comparison of maximum core temperatures during core heatup

7. LICENSING SUPPORT

7.1. SCOPE

The work scope of this task consists of providing reviews and critiques of licensing issues, MRS design features, and other information relevant to licensability of the HTGR-MRS.

7.2. DISCUSSION

Review comments were developed on the GE report entitled "Licensing Comparison of the Monolithic and Modular SC/C Concepts," October 5, 1982. In general, a comparison of the two systems does not appear to be very helpful.

Review comments were developed on the Bechtel Group, Inc. (BGI) report entitled "MRS BOP Safety Classification Review," February 14, 1983. The GA review generally agreed with the classifications assigned by BGI, but further study is needed in a few areas. Also, GA believes the classification system used by BGI is somewhat outmoded and that a more precise and usable system is in use at GA; the American Nuclear Society has also adopted a classification system for LWR's that is an improvement over that based on Nuclear Regulatory Commission Regulatory Guide 1.26.

Review comments were also developed on the GE report entitled "MRS; Design Approach (Ground Rules) Document Update (PCD Milestone 41.3)," March 1, 1983. A number of places were identified where the document would be improved by further explanation.

8. CIRCULATOR DESIGN

8.1 SCOPE

The scope of this task includes design effort to define the circulator number and configuration, interfaces, and the maintenance/removal requirements.

8.2. DISCUSSION

8.2.1. Circulator Number and Location Evaluation

A study on circulator number and location has identified two potential solutions: (1) a single circulator case not requiring loop isolation valves and (2) two circulators each with a primary coolant shutoff valve. Concerns and advantages of both alternatives have been evaluated in terms of after-heat removal, refueling, repressurization, availability, and maintainability.

A single circulator not requiring helium shutoff valve was concluded to provide sufficient service in terms of afterheat removal, refueling, and repressurization.

Critical interface requirements have been identified for the case of loss of forced circulation. Tentative figures indicate that circulator impeller may heat up to about 600°C (1100°F) while not rotating.

8.2.2. Circulator Interfaces

The design study of the interfaces between circulator and adjacent structures and systems was completed and a maintenance/removal method established.

As shown in Fig. 8-1, a cooling system was devised to protect the electric motor against the reactor vessel heat flux. Man access to the motor compartment was provided for electric motor maintenance and to remove and replace the high-pressure sliding water seal cartridge without disturbing the motor-circulator alignment. Figure 8-2 shows how remote removal and handling of the circulator and primary closure assembly has been accomplished by utilizing a horizontal ram combined with a cask and a set of isolation valves.

Previous studies to establish the number and location of circulators concluded that additional circulators offer no advantage in reducing the core heatup frequency. Therefore, a single circulator not requiring the loop shutoff valve has been selected as the viable concept for the MRS-process heat (PH) and MRS-SC/C reactors. Circulator maintenance and removal procedure and its impact on plant availability and licensing remained as some of the main concerns. In particular, removal of the circulator at elevated temperatures was a concern that this study addressed and conceptually resolved.

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Fig. 8-1. Circulator drive motor cavity cooling system

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Fig. 8-2. Circulator remote removal and handling concept

9. ALTERNATE REFORMER ALLOY SURVEY

9.1. SCOPE

This task is intended to survey and recommend GA advanced alloys to alleviate the projected cobalt activation problem in current Inconel 617 reformer tubes for the MRS-R application.

9.2. DISCUSSION

The current candidate reformer tube material for the advanced modular process heat reactor system is Inconel 617, a nickel-base alloy containing 12.5 wt% cobalt. Activation of the cobalt in a nuclear environment can make repair or replacement of the component hazardous, and corrosion products containing activated cobalt could increase the activity level of the primary coolant to unacceptable levels. This survey was intended to review the properties of a series of experimental alloys under development at GA to determine if one or more could be an acceptable substitute for Inconel 617.

Ten experimental alloys were produced which were designed to resist corrosion forming an Al_2O_3 scale and which were strengthened by gamma prime precipitation and the addition of molybdenum and/or tungsten (Table 9-1). Carburization is the primary concern in the reformer environment, and all the alloys exceeded the carburization resistance of Inconel 617 and Hastelloy X and were equivalent to or better than IN 100 (Fig. 9-1). Experimental alloy GA 909 had the best carburization of any alloy tested.

Three of the experimental alloys (GA 904, GA 908, and GA 910) had elevated temperature tensile properties that exceed the properties of Inconel 617 (Figs. 9-2 and 9-3). The creep-rupture behavior of the three strongest experimental alloys was compared to Inconel 617 at 900° and 1000°C (1652° and 1832°F). At 900°C (1652°F), the creep-rupture strength of all

TABLE 9-1
NOMINAL AND MEASURED CHEMICAL COMPOSITIONS OF EXPERIMENTAL, ADVANCED HTGR ALLOYS

Alloy Designation	Chemical Composition, Weight % Nominal [Actual (a)]							
	Ni.	Mo	W	Al	Ti	Zr	Cr	C
901	bal.	10 (10.0)	--	1 (1.04)	1 (1.05)	0.03 (0.025)	8 (7.8)	0.04 (0.019)
902	bal.	10 (10.1)	--	1 (1.07)	2 (2.08)	0.03 (0.030)	8 (7.9)	0.04 (0.035)
903	bal.	10 (10.0)	--	2 (2.15)	1 (0.98)	0.03 (0.025)	8 (8.0)	0.04 (0.019)
904	bal.	10 (9.9)	--	2 (2.02)	2 (2.06)	0.03 (0.035)	8 (7.8)	0.04 (0.033)
905	bal.	--	10 (9.9)	1 (1.00)	1 (1.06)	0.03 (0.034)	8 (8.0)	0.04 (0.029)
906	bal.	--	10 (10.0)	1 (1.12)	2 (1.96)	0.03 (0.040)	8 (8.0)	0.04 (0.023)
907	bal.	--	10 (10.0)	2 (2.12)	1 (0.98)	0.03 (0.040)	8 (7.8)	0.04 (0.027)
908	bal.	--	10 (9.8)	2 (2.10)	2 (2.17)	0.03 (0.035)	8 (7.8)	0.04 (0.024)
909	bal.	4 (3.85)	8 (7.9)	2 (1.92)	2 (1.99)	0.03 (0.047)	--	0.04 (0.02)
910	bal.	4 (4.0)	8 (7.8)	2 (1.83)	2 (1.88)	0.03 (0.047)	4 (3.7)	0.04 (0.02)

(a) Determined by x-ray fluorescence analysis (Special Metals Company).

Fig. 9-1. Effect of exposure temperature on the change in bulk carbon content of experimental, advanced HTGR alloys and selected commercial alloys after exposure in controlled-impurity helium

Fig. 9-2. Tensile yield strength (0.2%) versus temperature for experimental, advanced HTGR alloys and selected commercial alloys

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Fig. 9-3. Tensile ultimate strength versus temperature for experimental, advanced HTGR alloys and selected commercial alloys

three alloys exceeded that of Inconel 617, and at 1000°C (1832°F), Inconel 617 had the longest creep-rupture life (Figs. 9-4 and 9-5). The tests at both temperatures are within the scatter band normally associated with creep-rupture testing.

A conclusion drawn from the survey was that a wrought, cobalt-free, corrosion-resistant alloy can be made with sufficient strength for use in the reformer. Experimental alloy GA 908 should be evaluated further as a possible replacement for Inconel 617.

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Fig. 9-4. Comparison of the 900°C (1652°F) creep behavior of experimental, advanced HTGR alloys GA 904, GA 908, and GA 910 and the commercial alloy Inconel 617 at 51.7 MPa (7.5 ksi)

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Fig. 9-5. Comparison of the 1000°C (1832°F) creep behavior of experimental, advanced HTGR alloys with Inconel 617 at 22.8 MPa (3.3 ksi).

10. MRS-R PCRV CONCEPT DESIGN STUDY

10.1. SCOPE

This task is intended to study the PCRV concept design and to evaluate construction schedule, cost, and other factors that may impact selection of the vessel type. A single-cavity PCRV concept (Fig. 10-1) recommended from an earlier study will be compared with the steel-vessel variant.

10.2. DISCUSSION

Data are being collected that bear on the PCRV construction and schedule. Preliminary design and analysis of a steel dome closure and the accompanying liner forgings in the PCRV top head have been completed (Figs. 10-2 and 10-3). The design provides for a central superheat penetration exiting through the closure. A concrete closure, although feasible in this case, was not chosen because the vessel prestressing tendons that hold the closure would need to be relaxed whenever the closure removal is anticipated.

Fig. 10-1. Single-cavity PCRV with one steam generator

Fig. 10-2. Steel dome closure

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Fig. 10-3. Forging details

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