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ABSTRACT

An increase in reactor module power from 350 to 450 MW(t) would markedly improve the economics of the Modular High Temperature Gas-Cooled Reactor (MHTGR). The higher power level was recommended as the result of an in-depth cost reduction study undertaken to compete with the declining price of fossil fuel. The safety assessment confirms that the high level of safety, which relies on inherent characteristics and passive features, is maintained at the elevated power level.

Preliminary systems, nuclear, and safety performance results are discussed for the recommended 450 MW(t) design. Optimization of plant parameters and design modifications accommodated the operation of the steam generator and circulator at the higher power level. Events in which forced cooling is lost, designated as conduction cooldowns, are described in detail. For the depressurized conduction cooldown, without full helium inventory, peak fuel temperatures are significantly lowered. A more negative temperature coefficient of reactivity was achieved while maintaining an adequate fuel cycle and reactivity control. Continual improvement of the MHTGR delivers competitive performance without relinquishing the high safety margins demanded of the next generation of power plants.

MHTGR PROGRAM OVERVIEW AND STATUS

The U.S. Department of Energy (DOE) is sponsoring the development of an MHTGR that can provide safe, economic, and reliable power for the next generation of power plants. The MHTGR design team consists of General Atomics, ABB-Combustion Engineering Nuclear Power, Bechtel National, Incorporated and Stone and Webster Engineering Corporation with technology development by Oak Ridge National Laboratory and with utility input through Gas-Cooled Reactor Associates.

Currently, the MHTGR Program is supporting the safety and licensability review by the Nuclear Regulatory Commission (NRC). The draft Preliminary Safety Evaluation Report[1] in March 1989 has served as a substantial confirmation of the reference 350 MW(t) MHTGR's safety concept, although there are remaining issues. As the result of the recommendation from a cost reduction study (CRS) in October 1990, confirmatory studies are progressing on a recommended 450 MW(t) MHTGR which offers competitive economics with fossil fuel plants while maintaining the same high level of safety as the reference design.[2] Since the safety margins for the fuel are increased and essentially unchanged for the other components, it is judged that the key licensing issues of the improved design will not differ from those identified for the reference MHTGR.

A description of the MHTGR plant and comparison between the reference and recommended designs follows. A summary of the CRS discusses the MHTGR safety requirements, the key factors contributing to the recommendation to increase reactor module power and the associated economic gains. Preliminary evaluations of the 450 MW(t) design, subsequent to the CRS, are included in the systems, nuclear, and safety performance sections.

The reactor module systems and component design strategy is discussed in the systems performance section using the helium circulator as a representative example. Also, the expected steady-state performance is provided for key MHTGR components at 100% power. In the nuclear performance section, the MHTGR fuel cycle, water ingress reactivity worth, control rod reactivity worth, and temperature coefficients of reactivity are compared for the 350 and 450 MW(t) designs. Pressurized and depressurized conduction cooldown events are defined in the safety performance section. For each transient, the temperature histories are provided for key reactor components. Also, the maximum temperatures reached for the key components are compared for the reference and recommended designs. The significant design improvements identified in the performance sections are compiled in the closure.

Calculations, estimates and findings are based on best-estimate methodologies and represent conclusions characteristic of a conceptual design study. As such, many major conclusions need confirmation by verified and validated computer codes and related experimental programs, particularly in the area of fuel performance and fission product transport. The Department of Energy is reviewing the 450 MW(t) design option but no selection has been made.

MHTGR PLANT DESCRIPTION

The MHTGR design is based on generic, gas-cooled reactor experience and specific HTGR programs and projects, including the 52 carbon dioxide-cooled reactors developed in the United

Kingdom and built around the world, and the 5 helium-cooled reactors built in Western Europe and the United States. The MHTGR is being designed to meet the rigorous requirements established by the Nuclear Regulatory Commission (NRC) and the electric utility-user industry for a second-generation power source for the late 1990s. The plant is expected to be equally attractive for deployment and operation in the United States, other major industrialized nations, and the developing nations of the world.

The typical MHTGR plant includes an arrangement of four identical modular reactor units, each located in a single reactor building.[3] The plant is divided into two major areas: the nuclear island (NI), containing the four reactor modules, and an energy conversion area (ECA), containing turbine generators and other balance of plant equipment. The basic layout for a single reactor module is shown in Figure 1. Each reactor module is housed in adjacent, but separate, reinforced concrete structures located below grade and under a common roof structure. The below-grade location provides significant design benefits by reducing the seismic amplifications typical of above-grade structures.

The reactor module components are contained within three steel pressure vessels; the reactor vessel, a steam generator vessel, and connecting cross vessel. The uninsulated steel reactor pressure vessel is approximately the same size as that of a large boiling-water reactor and contains the core, reflector, and associated supports. Top-mounted penetrations house the control-rod drive mechanisms and the hoppers containing boron carbide pellets for reserve shutdown. The penetrations are also used as access for refueling and inspection. The reactor core and the surrounding graphite neutron reflectors are supported on a steel core support plate at the lower end of the reactor vessel. The reactor core is composed of hexagonal cross-section graphite blocks. Unfueled graphite blocks fill the top, bottom, center, and outer part of the core near the core barrel forming reflectors completely surrounding an annular fuel region. The annular, active core is composed of graphite fuel blocks containing fuel compact material. The fuel itself is in the form of coated particles. Two types of particles are used in the composition of the fuel blocks; fissile particles which contain low-enriched uranium, and fertile particles which can contain either thorium or natural uranium. The fuel particles are then bonded together in fuel compacts. These compacts are contained in sealed vertical holes in the graphite fuel blocks.

The heat transport system (HTS) provides heat transfer during normal operation or under normal shutdown operation using high pressure, compressor driven helium that is heated as it flows through the core from top to bottom. The coolant is collected in the plenum below the core and flows through the

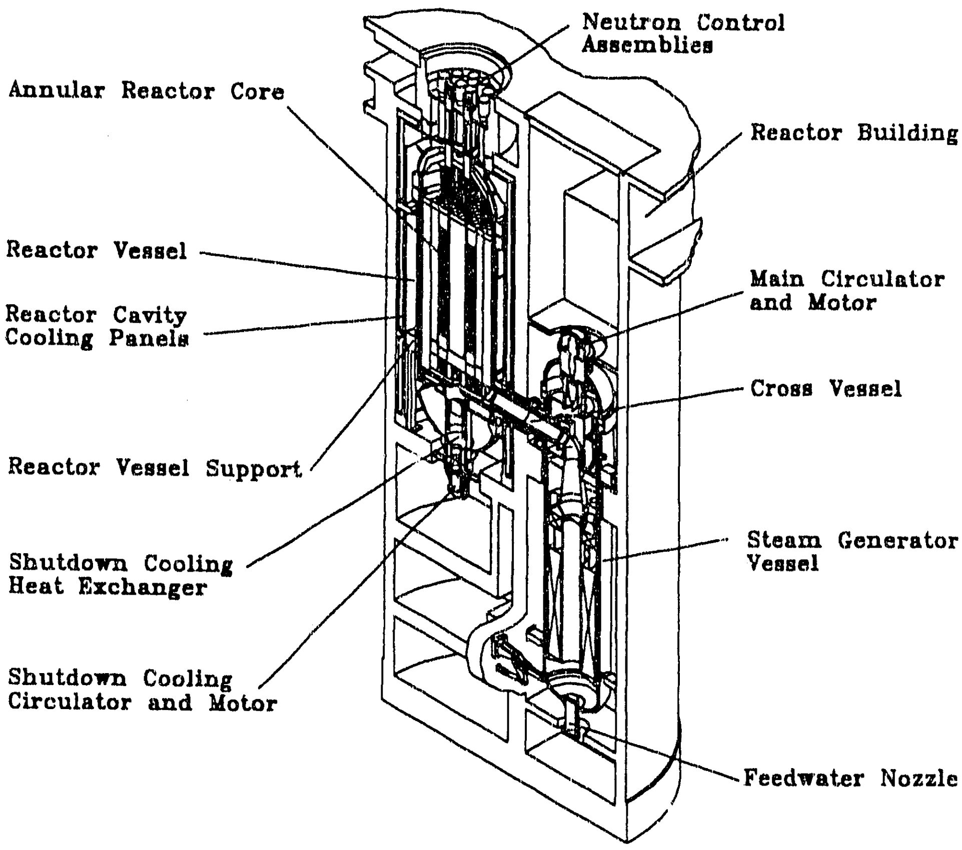

Figure 1. Isometric drawing of one of the plants four MHTGR modules

coaxial hot duct inside the cross vessel to a once-through helical bundle steam generator. After flowing downward over the steam generator tubes, the helium flows upward, in an annulus, between the steam generator vessel and a shroud leading to the main circulator inlet. The main circulator is a helium submerged, electric-motor-driven, two-stage axial compressor with active magnetic bearings. The helium is discharged from the circulator and flows through the annulus of the cross vessel and hot duct and then upward past the reactor vessel walls to the top plenum over the core.

For availability and maintenance requirements, a separate shutdown cooling system (SCS) is provided as a backup to the primary HTS. The shutdown heat exchanger and shutdown cooling circulator are mounted on the bottom of the reactor vessel. The heat removal systems allow hands-on module maintenance to begin within 24 hours after plant shutdown. The reactor cavity cooling system (RCCS) is located in the concrete structure external to the reactor vessel to provide a passive heat sink to remove residual heat from the reactor cavity if the HTS and SCS are unavailable to perform their intended functions. The RCCS consists of above-grade intake structures that naturally convect outside air down through enclosed ducts and panels that surround the below-grade core cavity before returning the warmed air through above-grade outflow structures. The core heat is transferred by conduction, convection, and radiation from the core to the RCCS. This system has no controls, valves, circulating fans, or other active components and operates continuously during normal operation and during shutdown conditions.

In the reference MHTGR design each of the reactor modules produces a 100% rated power of 350 MW(t). The reactor modules are paired, with each pair feeding one of two turbine generators, to produce 538 MW(e) net power for a four module plant. As the result of the CRS, the recommended design is configured to yield a 100% rated power of 450 MW(t) per reactor module. Each reactor module is connected independently to a single turbine generator to provide 692 MW(e) net power for a four module plant. A comparison of nominal plant parameters for these two designs is offered in Table 1.

The cross sectional core layouts for the 350 and 450 MW(t) designs are shown in Figure 2. The 84 fuel column, 450 MW(t) configuration is achieved by adding one ring of graphite reflector blocks to the inner reflector region of the 66 fuel column reference design while maintaining the width of the active core annulus at three blocks. In the 84 fuel column core, in-core control rods are selected to achieve shutdown. This change increases the reactivity shutdown margins for the larger core while accommodating vessel layout and refueling requirements.

COST REDUCTION STUDY

It was anticipated from the outset of the CRS that raising the power level of each MHTGR module would yield the greatest economic improvement. However, it was judged essential to commercial deployment to maintain the safety characteristics of the MHTGR.[4] Two requirements for the MHTGR plant are the keys to providing this product distinction:

TABLE 1
MHTGR PLANT PARAMETERS

Reactor Module Parameters	Reference Design	Recommended Design
Thermal Power, MW(t)	350	450
Fuel Columns	66	84
Fuel Cycle	LEU/Th	LEU/Natural U
Average Power Density, W/cm^3	5.91	5.99
Vessel Inside Diameter, cm (ft)	655.3(21.5)	722.4(23.7)
Primary Side Pressure, MPa (psia)	6.38(925)	7.07(1025)
Core Inlet Temperature, °C(°F)	258(496)	288(550)
Core Outlet Temperature, °C(°F)	689(1273)	704(1300)
Steam Temperature, °C(°F)	541(1005)	541(1005)
Steam Pressure, MPa (psia)	17.3(2515)	17.3(2515)
Circulator Power, MW(e)	3.1	3.6

1. The plant shall be designed to perform its safety functions without credit for sheltering or evacuation of the public beyond the plant's exclusion area boundary.
2. The plant shall be designed to perform its safety functions without reliance on control room equipment, the automated plant control system, or operator actions.

Hence, before any design alternative was selected, a critical overall system check involved assurance that the above requirements were met. In essence, this required acceptable response to three limiting design basis events.

1. A pressurized conduction cooldown (limiting temperature for the upper plenum shroud, core barrel and for the reactor vessel).
2. A depressurized conduction cooldown (limiting temperature for intact fuel particles).
3. A depressurized conduction cooldown with water ingress (limiting temperature for release due to hydrolysis of defective fuel particles).

The latter two events dominate fission product releases used in calculating doses for comparison with Protective Action Guideline (PAG) limits associated with the need to plan for evacuation and sheltering of the public.[5]

The principal change and primary contributor to cost reduction is an increase in the four-module plant electrical output of 28.6%, corresponding to an increase in reactor module power from 350 to 450 MW(t). This was achieved by increasing the

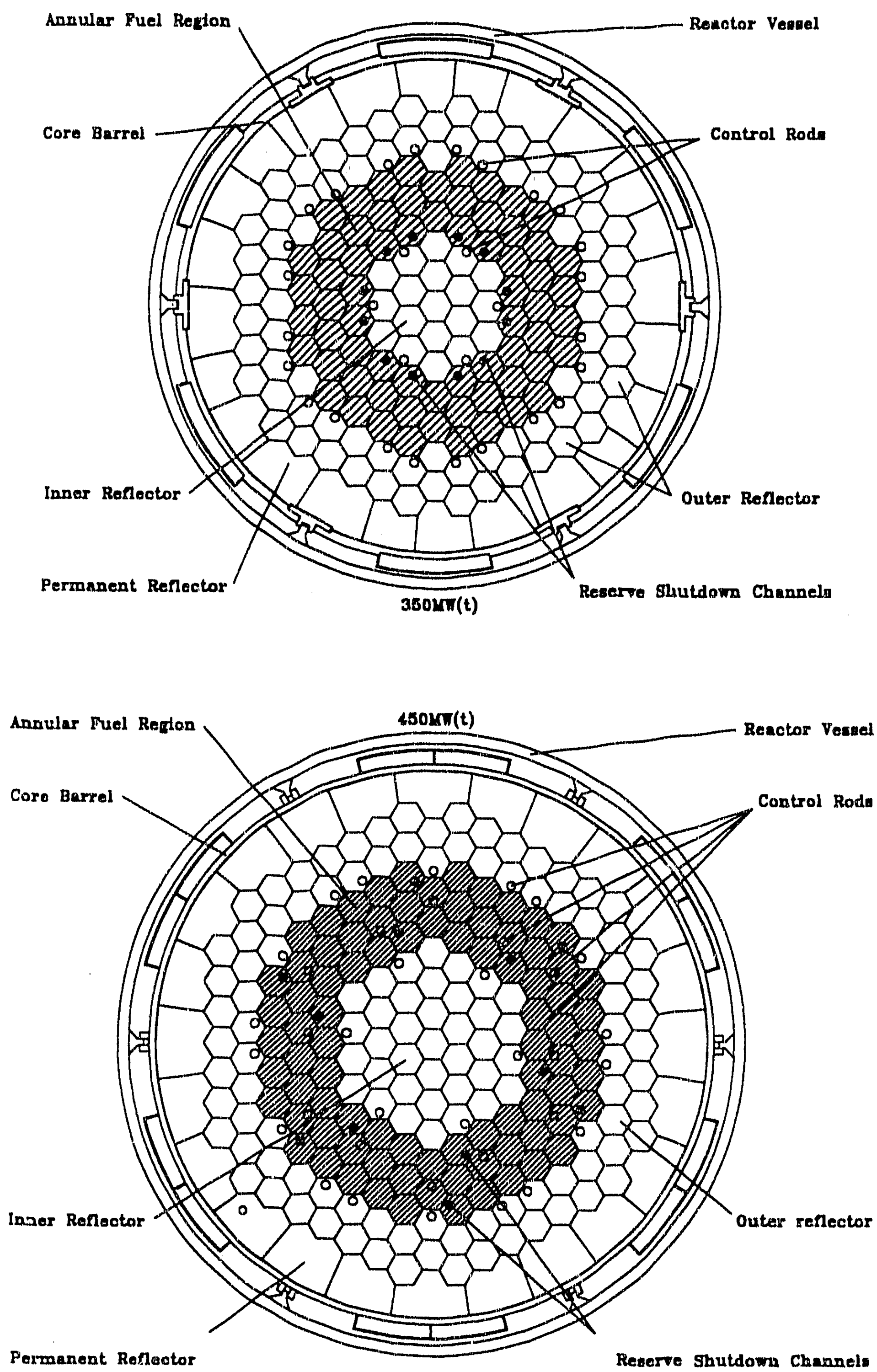

Figure 2. Plan view of reference 350 MW(t) and recommended 450 MW(t) annular core

annular core array from 66 to 84 fuel columns while maintaining the core average power density essentially unchanged. The reactor vessel increased 2.2 feet in outer diameter as a consequence of the larger core. Primary system parameters were optimized for the circulator and steam generator and their designs modified to accommodate the higher reactor power level. A number of factors contributed to the recommendation to increase the module rating to 450 MW(t). The most important are:

1. A more accurate evaluation of decay heat levels which support the use of 12% margin for uncertainty and instrument error versus a margin of 23% used for the reference design.
2. A change of fuel cycle from a low enriched uranium/thorium (LEU/Th) cycle to a low enriched uranium/natural uranium (LEU/U) cycle, resulting in still lower decay heat and attendant peak fuel temperatures during conduction cooldown events.
3. A determination that the reactor and steam generator vessel sizes and weights remain within manufacturing and transportation capability.
4. Adjustment of the primary system parameters to support an engineering and experience-based judgement that the steam generator and main helium circulator can be developed to accommodate the higher power level.
5. An overall judgement that 450 MW(t) is a prudent maximum power for the reactor module when appropriate design margins are considered.

These factors combine to allow an increase in the reactor module power from 350 to 450 MW(t) while increasing the margin on peak fuel temperatures for conduction cooldown events. Provisions for additional margin on peak fuel temperatures are considered an especially important outcome of this study since added margin further reduces the challenge to fuel particle coatings.

Based on screening economic calculations, the recommended 450 MW(t) design unit capital costs are 13% less than the 350 MW(t) design. Most of the cost reduction can be attributed to the economies of scale with design optimization accounting for the remainder. A comparison of costs for the MHTGR and a comparable coal plant is shown in Figure 3. The unit capital costs ($/kWe) for the coal plant are still less than those of the 450 MW(t) MHTGR. However, the mean busbar costs (mills/kW-hr) for the 450 MW(t) MHTGR are ~15% less than for the comparable coal plant. Overall, the MHTGR offers competitive economics with fossil fuel plants. Based on the promising safety and economic results of the CRS, further analysis was performed to confirm the systems, nuclear, and safety performance of the 450 MW(t) core configuration. The following results are considered preliminary.

450 MW(t) MHTGR SYSTEMS PERFORMANCE

Reactor module systems and components are designed to perform within the thermal performance envelopes specified for each key component. The purpose of these envelopes is to maximize plant performance reliability with optimized design. The thermal performance envelopes account for performance uncertainties in actual plant as-built parameters and plant

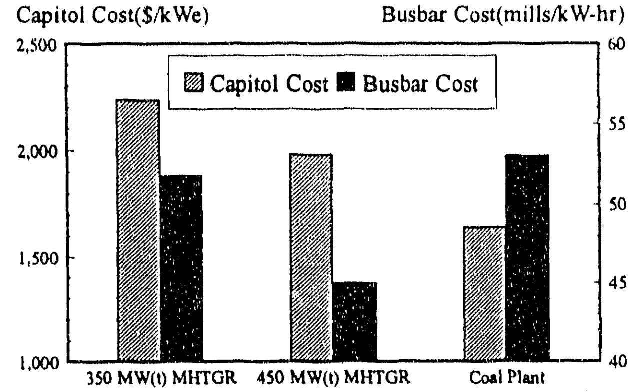

Figure 3. MHTGR cost comparison

operation, and provide a means to determine required margins in a disciplined and visible fashion. The consideration and incorporation of module performance uncertainty early in the design phase significantly reduces the risk of disrupted plant operation due to unexpected performance characteristics.

An MHTGR component which can be used as an example to explain this relation is the helium circulator. By circulating the primary coolant, this component transports the core generated heat to the steam generator. The circulator design is crucial for successful MHTGR operation. If sized for point design requirements, the integrated plant has to perform as expected or otherwise the circulator is either undersized or oversized. If undersized, the circulator is incapable of providing the required heat removal rate to achieve rated power output. As a result, the plant may have to be derated or a properly sized circulator be refitted, measures which in any case result in economic loss. If the circulator is oversized, then the module will operate successfully but the eventual cost of excess margin may also result in an economic penalty. Also, adding margin to other components as well will almost certainly result in unexpected module performance, unless these margins are integrated on a systems level.

An example of the thermal performance envelope developed for the helium circulator is depicted in Figure 4. Shown is the system helium pressure drop as a function of circulator helium

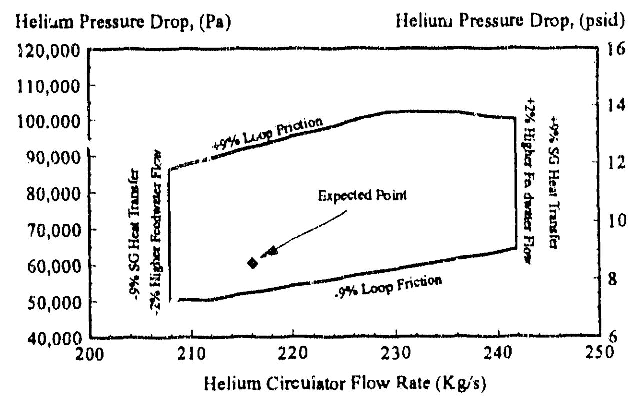

Figure 4. Main circulator thermal performance envelope at rated reactor power

flow. The solid triangle located within the envelope represents the expected operating point. The area contained within the envelope boundaries specifies the range of potential module performance when accounting for performance uncertainties in primary coolant loop pressure drop and steam generator heat transfer effectiveness, and the measurement uncertainty in feedwater flow rate.

Key MHTGR primary and secondary coolant parameters are provided in Table 2. Prominent are the steam conditions of 541°C (1005°F) at 17.3 MPa (2515 psia) which are typical steam conditions for fossil generating stations. Typical for the 450 MW(t) MHTGR are the primary coolant core inlet and outlet temperatures of 288°C (550°F) and 704°C (1300°F), respectively. The four reactor modules are coupled to individual turbine generators to produce a net electrical power output of approximately 692 MW(e) which results in a net efficiency of nearly 38.4%. Preliminary nuclear and safety assessments are based on the MHTGR plant parameters in Table 2.

TABLE 2
EXPECTED STEADY-STATE PERFORMANCE AT 100% POWER

NI Heat Balance, MW(t)	
Heat generated by core	451
Heat added by circulators	2.7
Loss to RCCS	1.1
Loss to Heating, Ventilation and Air Conditioning	0.1
Loss to Shutdown Cooling Heat Exchanger	0.4
Loss to ECA	452
Active Core	
Inlet helium flow rate, kg/s	211
Inlet helium temperature, °C	286
Inlet helium pressure, MPa	7.06
Outlet helium temperature, °C	697
Regenerative heat, MW	0.14
Steam Generator	
Inlet helium flow rate, kg/s	213
Inlet helium temperature, °C	692
Inlet helium pressure, MPa	7.02
Outlet helium temperature, °C	284
Inlet feedwater flow rate, kg/s	176
Inlet feedwater temperature, °C	193
Inlet feedwater pressure, MPa	23.1
Outlet steam temperature, °C	541
Outlet steam pressure, MPa	17.3
Steam/water pressure drop, MPa	5.8
Number of active tubes	556
Main Circulator	
Circulator helium flow rate, kg/s	215
Outlet helium temperature, °C	286
Outlet helium pressure, MPa	7.07
Helium temperature rise, °C	2.3
Helium pressure rise, MPa	0.06
Circulator speed ratio	0.84

450 MW(t) MHTGR NUCLEAR PERFORMANCE

The reactor power was increased from 350 to 450 MW(t) by increasing the core volume while keeping the core power density relatively constant, so that the basic physics parameters would not change significantly. This was accomplished by moving the active core out one row, which increased the number of fueled columns from 66 to 84, while keeping the active core height constant at ten elements per column. The increase in the reactor power therefore corresponded to a change in core average power density of only 5.91 to 5.99 W/cc (see Table 1 and Figure 2). The cycle time between reloadings was also decreased from 1.65 years for the 66 column, 350 MW(t) design (66/350), to 1.50 years for the 84 column, 450 MW(t) design (84/450). This was changed to satisfy a utility desire to have reloadings on multiples of six months, i.e. at intervals of 12, 18, or 24 months.

The fertile material was changed from thorium to natural uranium (NU) to decrease the decay heat, so that lower temperatures would be obtained during conduction cooldown events, as discussed in the following section. In addition, the average fertile loading per element for the equilibrium cycle was decreased by 35%, from 1.07 kg Th/element, to 0.69 kg NU/element. This allowed the equilibrium cycle fissile loading per element to be decreased by 6%, and to still meet the fuel cycle requirements. The fertile and fissile loadings per element were decreased in this manner to obtain a more thermal neutron energy spectrum, which is desirable to decrease the positive reactivity effect of water ingress into the core from a steam generator tube break accident. The maximum positive reactivity worth of water ingress in the 706 kg Th design at operating temperature is about 3.7%. This value is reduced to 3.1% for the 84/450 design with 583 kg of NU.

An additional benefit of moving the core out one row, is the ability to insert control rods into the core as well as the reflectors. Flexibility in control rod placement is a result of having a larger reactor vessel head with more space for the penetrations. This allows the total number of control rods to be increased from 30 for the 66 column core to 36 for the 84 column core as shown in Figure 2. Though this is actually a decrease in the number of control rods per core column, the total reactivity worth of the control rods increases due to locating twelve of the control rods in the core. For the 66/350 design with 30 control rods in the inner and outer reflector, the total control rod worth is 28.7% for hot, dry, end of equilibrium cycle (EOEC) conditions. This value increases to 42.2% for the 84/450 design with 12 control rods in the core and 24 control rods in the reflectors. The total control rod worth is more than enough to reach cold shutdown during normal operating conditions, even with full nuclide decay, as given in Table 3.

Not included in Table 3 is the effect of the reserve shutdown control (RSC). This consists of boron carbide (B_4C) pellets that can be released into twelve RSC channels in the core, located as shown in Figure 2. Up to six fixed burnable poison (FBP) rods per element are also used for reactivity control. These rods, which contain B_4C, are inserted into the elements at the time of element fabrication, and the boron gradually burns out over the cycle. The FBP is designed to reduce the maximum excess reactivity to about 3.5% at the middle of equilibrium cycle (MOEC), as in Table 3.

TABLE 3
ROD WORTH REQUIREMENTS FOR COLD SHUTDOWN
FOR THE 84/450 DESIGN

	BOEC % Reactivity Change	MOEC % Reactivity Change	EOEC % Reactivity Change
Shutdown Requirement			
Core Reactivity, Hot, Unrodded	1.0	3.5	0.0
Temperature Defect, Hot to 27°C	5.8	2.7	1.7
Nuclide Decay(3 years)	3.9	4.9	6.1
Total Required	10.7	11.1	7.8
Worth of 36 Control Rods (cold)	27.6*	29.7	36.3*
Shutdown Margin Below Critical	16.9	18.6	28.5

* Estimates based on hot calculations and MOEC cold calculations.

Density changes in the helium coolant, and dimensional changes in the reactor components, that might result from temperature or pressure changes, minimally affect the core reactivity. But a temperature increase in the fuel, which broadens resonances, and in the graphite moderator, which hardens the spectrum, has a significant impact on the core reactivity. The core temperature coefficients are shown in Figure 5 as a function of the average temperature, for the 66/350 design with 697 kg of Th and the 84/450 design with 583 kg of NU as fertile material.

The temperature coefficients for both designs are negative at all temperatures, and increasingly negative at higher temperatures due to the neutron spectrum shifting into the Pu^{240} capture resonance at 1.1 ev. The temperature coefficients for the design with natural uranium as the fertile material is more negative than the previous design with thorium as the fertile material. Compared to the reference 350 MW(t) design, the

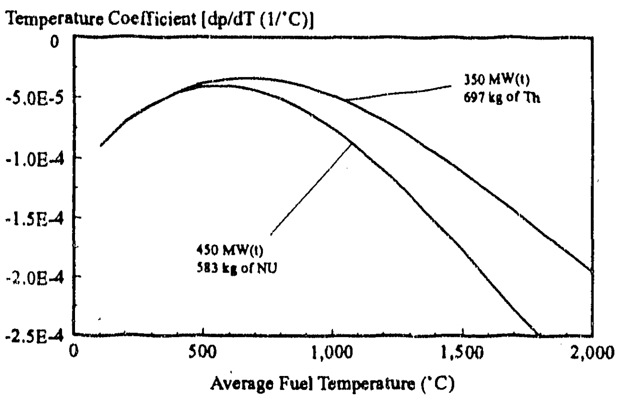

Figure 5. Core temperature coefficients at EOEC

recommended 450 MW(t) design with 583 kg of NU has a more negative temperature coefficient by a factor of ~ 1.3 at an average fuel temperature of 700°C.

The core temperature coefficients are negative for the 84/450 design with 583 kg NU throughout the equilibrium cycle as shown in Figure 6. For very slow reactor transients, where the reflector is also heating up, the net temperature coefficient for the entire system is always negative.

A change in the core configuration and fertile material from the 66/350 design with 706 kg Th to the 84/450 design with 583 kg NU improved the nuclear design in regards to water ingress events, control rod worth, and temperature coefficients of reactivity. The preliminary safety assessment which follows, assumes that the decay heat fraction after shutdown for the 84/450 design with 583 kg NU is the same as that calculated for the 66/350 design with 642 kg NU. Based on initial screening calculations, the assumption is judged to be conservative. The total decay heat for the first 100 hours after shutdown in the 450 MW(t) compared to the 350 MW(t) conduction cooldown evaluations is 6% less.

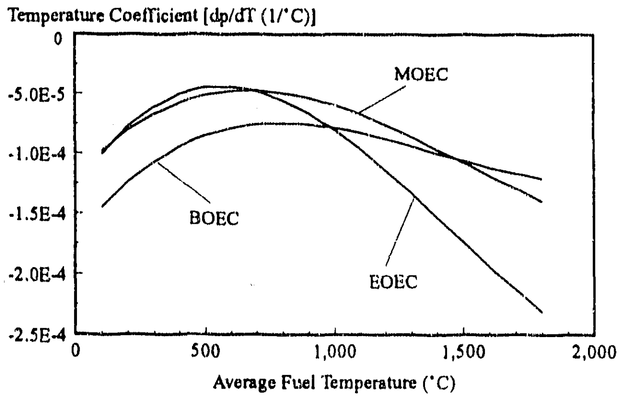

Figure 6. Core temperature coefficient for the 84/450 with 583 Kg NU design

450 MW(t) MHTGR SAFETY PERFORMANCE

Conduction cooldown accidents, both pressurized (PCC), and depressurized (DCC), are an important class of events which challenge decay heat removal and the control of radionuclides. The most important PCC and DCC events are when the HTS & SCS fail immediately and indefinitely to perform their respective cooling functions. Therefore, decay heat must be removed by thermal radiation from the reactor vessel to the air-cooled RCCS. A conservative value of decay heat, 1.12 times nominal, is used in the following assessments of PCC and DCC transients.

A pressurized loss of forced core cooling results in an elevated temperature transient in and around the core because of the mismatch between decay heat deposition and heat removal. This results from failures of the HTS and SCS. The PCC event sequence analyzed is given by the following:

1. Loss of offsite power and turbine-generator trip results in a loss of the main circulator and feedwater pumps.

2. The plant protection and instrumentation system senses the loss of flow and automatically trips the reactor and the outer reflector control rods are inserted.
3. SCS fails to start because of a failure of the backup power supply.
4. RCCS, which is always in operation, removes the core decay heat from the vessel by conduction and radiation to the air naturally circulating within the RCCS.
5. The reactor vessel remains pressurized, primary coolant boundary integrity is maintained, and no radionuclide release occurs.

The function of removing core heat is challenged when loss of offsite power, turbine-generator trip, failure to start the two backup generator sets, and failure of the SCS to start results in a loss of forced core cooling. This leads to a slow heatup of the core. Natural circulation within the core redistributes heat from the hottest portions of the core to the cooler regions, enhancing the conduction and radiation heat transfer from the core by distributing the heat over a larger surface area. The function of afterheat removal is then accomplished by convection and thermal radiation to the RCCS. As forced circulation flow through the core is lost, natural convection within the core develops, driven by the difference in hydrostatic pressure between the hot and cold regions of the core. The warming of the substantial volume of gas in the core inlet plenum, caused by this recirculation, brings about a rise in primary coolant pressure. However, over a very short period of time, the core inlet plenum comes into equilibrium with the upper reflector temperatures. Further temperature change is limited to the slow rate at which the reflector temperatures change and the pressure rise is terminated. Simultaneously, the heat loss from the primary coolant at other locations acts first to limit the pressure rise and finally, beyond 10 hours, to reverse the transient while system pressure is still below the pressure relief valve setpoint.

A comparison of the peak fuel, average fuel, and peak vessel temperature transients, using conservative decay heat, is shown in Figure 7. The peak fuel curve remains below the 1600°C threshold for thermally induced fuel failures. Fuel failure can occur at 1600°C only if the fuel stays at this temperature for many hundreds of hours. These conditions are not approached during the PCC event. The peak fuel temperature reaches a maximum of 1092°C(1997°F), and the average fuel temperature reaches a maximum of 879°C(1615°F). Both of these fuel temperatures peak at approximately 65-70 hours after shutdown. The vessel reaches its maximum temperature of 402°C(755°F) approximately 85 hours after shutdown. The total time which the peak vessel temperature remains over 371°C(700°F) is approximately 205 hours. The thermal transient in the core is a function of the decay heat, core heat capacity, and the ability of the primary coolant to transport heat out of the core via natural circulation to the reflectors, core barrel, and the vessel. Ultimately, heat from the vessel is rejected to the RCCS by natural convection and thermal radiation.

Loss of forced cooling and the helium inventory results in a depressurized conduction cooldown. A failure in one of the instrument and service system lines which penetrate the pressure vessels could cause a primary coolant leak ranging in size from [$2x10^{-4}$ cm^2, $3x10^{-5}$ in^2] below which the helium purification system makes up the loss in inventory, to an offset rupture of the

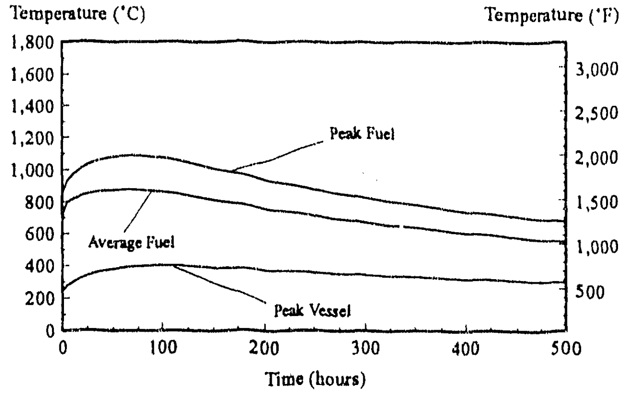

Figure 7. Conservative PCC peak fuel, average fuel, and peak vessel temperature histories

line [6.5 cm^2, 1.0 in^2]. The DCC event sequence analyzed, based on a risk dominant leak size, is given by the following:

1. Primary coolant escapes through a 0.32 cm^2 (0.05 in^2) area leak near the top of the reactor vessel.
2. Reactor trips automatically on low reactor pressure and all the outer reflector control rods are inserted.
3. HTS fails immediately after initiating event.
4. SCS fails to start on demand.
5. Primary coolant depressurizes and is released into the below grade silo and then into the environment.
6. RCCS, which is always in operation, removes the core decay heat from the vessel by conduction and radiation to the air naturally circulating within the RCCS.
7. Reactor building functions properly with the dampers opening to relieve pressure buildup due to the instrument line leak.
8. Reactor building leaks at ground level a mixture of air, primary coolant helium and radionuclides to the environment.

For a 0.32 cm^2 (0.05 in^2) leak, the primary system undergoes a relatively slow depressurization which challenges the function of removing core heat. The penetration leak, located at the top of the reactor vessel, will depressurize the reactor system over a period of approximately 24 hours. Following the blowdown, gas continues to escape from the vessel. This is due to thermal expansion resulting from the core heatup and will last until after the average core temperatures have peaked, (approximately 70 hours). The vessel system pressure following blowdown is nearly atmospheric. Because of the low coolant pressure and the large core coolant channel length to diameter ratio, L/D, (i.e., approx. 700), natural convection cooling is ineffective.

A comparison of the peak fuel, average fuel, and peak vessel temperature transients, using conservative decay heat, is shown in Figure 8. The peak fuel temperature during the DCC event remains below the 1600°C threshold for measurable, thermally induced fuel failure. The peak fuel temperature reaches a maximum of 1531°C(2787°F), and the average fuel temperature reaches a maximum of 1211°C(2212°F). Both of these fuel temperatures peak at approximately 68 hours after shutdown.

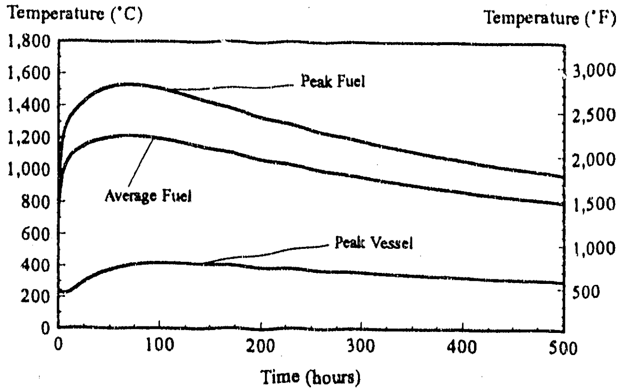

Figure 8. Conservative DCC peak fuel, average fuel, and peak vessel temperature histories

The vessel reaches its maximum temperature of 416°C(782°F) approximately 106 hours after shutdown. The total time which the peak vessel temperature remains over 371°C(700°F) is approximately 210 hours. The thermal transient in the core is a function of the decay heat, core heat capacity, and the ability of the annular core to conduct and radiate heat to the reflectors and the vessel. Ultimately, heat from the vessel is rejected to the RCCS by natural convection and thermal radiation.

A comparison of key reactor component temperatures from the 450 MW(t) design with the 350 MW(t) design can help to illustrate safety margin improvements. For both PCC and DCC events, the peak fuel temperatures are lower for the 450 MW(t) reactor design providing a greater temperature margin over the 350 MW(t) design as given in Table 4. The table also shows that for the DCC event, which is the most challenging to the peak vessel temperature, the 450 MW(t) design is lower than the 350 MW(t) plant. A summary of the key safety improvements of the 450 MW(t) compared to the 350 MW(t) design is included in the closure.

CLOSURE

Based on the promising results of the CRS, more detailed evaluations were made on the 450 MW(t), 84-column core option with an all uranium fuel cycle. Optimization of plant parameters and design modifications accommodated the operation of the steam generator and circulator at the higher power level. To maximize plant performance reliability, reactor module systems and components are designed to perform within the thermal performance envelopes specified for each component.

The preliminary, key improvements identified in the previous sections of the recommended 450 MW(t) MHTGR design compared with the reference 350 MW(t) design are as follows:

1. Unit capital costs are 13% less.
2. Total decay heat during the first 100 hours after reactor shutdown is 6% less.
3. Maximum positive reactivity worth of water ingress at operating temperatures was decreased from 3.7% to 3.1%.
4. Total reactivity worth of control rods was increased from 28.7% to 42.2% for hot EOEC conditions.

TABLE 4
COMPARISON OF 350 AND 450 MW(t) KEY COMPONENT TEMPERATURES

Conservative Results	350 MW °C	450 MW °C	Limits °C
PCC			
Max Fuel	1286	1092	1600
Max Vessel	391	402	427
Max Core Barrel	698	702	760
Max Control Rod	872	801	1175
DCC			
Max Fuel	1621	1531	1600
Max Vessel	456	416	482
Max Core Barrel	613	697	760
Max Control Rod	1133	1118	1175

5. The core temperature coefficient is more negative at 700°C by a factor of 1.3.
6. Maximum fuel temperatures during DCC are 90°C lower.
7. Maximum fuel temperatures during PCC are 194°C lower.
8. Maximum vessel temperatures during DCC are 40°C lower.

Continuous improvement of the MHTGR delivers competitive performance while preserving high safety margins.

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