THE DESIGN, SAFETY AND PROJECT DEVELOPMENT STATUS OF THE MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR IN THE UNITED STATES

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ABSTRACT

The cooperative government and industry Modular High Temperature Gas-Cooled Reactor (MHTGR) Program in the United States has advanced a 350MW(t) plant design through the conceptual development stage. The system incorporates an annular core of prismatic fuel elements within a steel pressure vessel connected, in a side-by-side arrangement, by a concentric duct to a second steel vessel containing a steam generator and helium coolant circulator. The reference plant design consists of four reactor modules installed in separate below-grade silos, providing steam to two conventional turbine generators. The nominal net plant output is 540MW(e). The small reactor system takes unique advantage of the high temperature capability of the refractory coated fuel and the large thermal inertia of the graphite moderator to provide a design capable of withstanding a complete loss of active core cooling without causing excessive core heatup and significant release of fission products from the fuel. Present program activities are concentrated on interactions with the Nuclear Regulatory Commission aimed at obtaining a Licensability Statement. A project initiative to build a prototype plant which would demonstrate the MHTGR-unique licensing process, plant performance, costs and schedule plus establish an industrial infrastructure to proceed with follow-on commercial MHTGR plants by the turn of the century, is being undertaken by the utility/vendor participants.

INTRODUCTION

The hiatus in new nuclear plant orders in the United States beginning in the mid-to-late 70's portended a need to reconsider the requirements for the acceptance of nuclear power by utilities, regulatory and financial institutions and, in general, by the public. The Three Mile Island accident in 1979 compounded the crisis. During this same period, the High Temperature Gas-Cooled Reactor (HTGR) design and development program was concentrating on large plants as the logical successors to the 40MW(e) Peach Bottom developmental plant and the 330MW(e) Fort St. Vrain demonstration plant. These large reactor designs incorporated the same well-characterized refractory-coated fuel, graphite moderator and support structures and helium coolant typical of all HTGR designs; the basic features which enable the capability to provide not only the high temperatures required for superheated steam and efficient electricity production but also the large margins in terms of time and temperature to withstand off-normal events.

A framework for revising the requirements for the acceptance of nuclear power began to evolve in the early 80's as utilities foresaw the need for smaller plants consistent with lower load growth patterns and reduced financial exposure. Moreover, the need for a greater and more obviously-apparent degree of simplicity, safety and investment protection became an often-voiced opinion requiring quantification. Driven by these requirements, the cooperative government and industry HTGR program in the United States was redirected in early 1984 to the consideration of smaller reactor designs. The requirements, as summarized in Table 1, were quantified and, through a disciplined trade-off approach, a basic design with a power output of 350MW(t) per reactor module incorporating an annular core of the basic HTGR prismatic fuel elements was selected as the design that best met the requirements. The reference MHTGR plant was selected as four reactor modules with a total thermal rating of 1400MW(t) and a net electrical output of about 540MW(e). As the design evolution progressed it became increasingly apparent that the smaller reactor was capable of more fully exploiting the inherent characteristics of the fuel and graphite to yield a unique reactor system capable of withstanding a complete loss of active core cooling without causing excessive core heatup and significant release of fission products from the fuel.

Following the design selection process late in 1985, a conceptual design phase for the reference plant was initiated, culminating in the preparation of a Preliminary Safety Information Document (PSID) which was submitted in 1986 to the U.S. Nuclear Regulatory Commission (NRC) for review. An NRC Licensability Statement is expected in early 1988 following an extensive interaction

TABLE 1

SUMMARY OF KEY REQUIREMENTS

- AN EQUIVALENT AVAILABILITY FACTOR OF 80%, WITH PLANNED DOWNTIME OF ≤10% PER YEAR.
- AT LEAST A 10% ECONOMIC ADVANTAGE OVER THE BEST COAL-FUELED ALTERNATIVE SOURCE OF ELECTRICITY.
- LOW-ENRICHED FUEL AND A ONCE-THROUGH THROW-AWAY FUEL CYCLE.
- SITING PARAMETERS:
  - 425 m EXCLUSION AREA RADIUS
  - SEISMIC DESIGN FOR A 0.3g SAFE-SHUT DOWN EARTHQUAKE (SSE) AND A 0.15g OPERATING BASIS EARTHQUAKE.
- INVESTMENT PROTECTION GOALS:
  - <10% UNSCHEDULED UNAVAILABILITY.
  - $<10^{-5}$ PER PLANT YEAR FOR A FREQUENCY OF EVENTS LEADING TO A MODULE LOSS.
- REGULATORY REQUIREMENT:
  - MEET EXISTING NUCLEAR REGULATORY COMMISSION (NRC) AND ENVIRONMENTAL PROTECTION AGENCY (EPA) DOSE AND RISK CRITERIA.
- UTILITY'S EMERGENCY PLANNING REQUIREMENT:
  - DOSES NOT TO EXCEED EPA PROTECTIVE ACTION GUIDELINES FOR PUBLIC EVACUATION OR SHELTERING DOWN TO AN ACCIDENT FREQUENCY OF 5×10^{-7} PER PLANT YEAR.

period between the NRC and the Program participants. The overall Program Plan contemplates a subsequent step-by-step approach to the ultimate goal of a standardized MHTGR plant which can receive a design certification from the NRC and be commercially replicated. A key element of the overall Program Plan is the construction of an initial plant that would provide demonstration of the MHTGR licensability, performance and economics.

The major participants in the cooperative U.S. program include the U.S. Department of Energy (DOE), Gas-Cooled Reactor Associates (GCRA), GA Technologies (GA), Combustion Engineering (CE), General Electric (GE), Bechtel National (BNI), Stone & Webster Engineering Corporation (SWEC), and Oak Ridge National Laboratory (ORNL). In addition, the U.S. MHTGR Program has cooperative development programs through the government and industry with the Federal Republic of Germany, Japan and the United Kingdom.

MHTGR PLANT DESIGN

The reference four-module MHTGR plant layout is shown in Figure 1. Each reactor module is housed in a vertical cylindrical concrete enclosure, which is fully embedded in the earth. Each reactor enclosure serves as an independent confinement structure with a vented and filtered exhaust system. The Nuclear Island (NI) portion of the reference plant is a separately-fenced and controlled-area of the plant consisting of four reactor enclosures and auxiliary structures that house common systems for fuel handling, helium processing, and other essential services. The Energy Conversion Area (ECA) is considered as completely non-safety related and is physically separated from the NI to facilitate the use of conventional, fossil-fired standards in its construction. The ECA contains all of the equipment required to provide feedwater and non-essential services to the NI and to generate electricity with the superheated steam that is returned including the rejection of the waste heat. A common control room located outside of the Nuclear Island is used to operate all four reactors and the turbine plant. A summary of the design parameters for the reference plant is given in Table 2.

Figure 2 shows the physical arrangement of a single reactor module with emphasis on the heat transport and shutdown cooling systems. The primary components are contained within three steel vessels - a reactor vessel, a steam generator/circulator vessel, and a connecting concentric crossduct vessel with hot gas flowing througn the inner duct and cooled gas returning through an annular-shaped outer duct.

The reactor vessel, about 6.9 m (22.9 ft) in diameter and 21.9 m (72 ft) in length, contains the prismatic annular core, the control rod drives and the reserve shutdown system. The

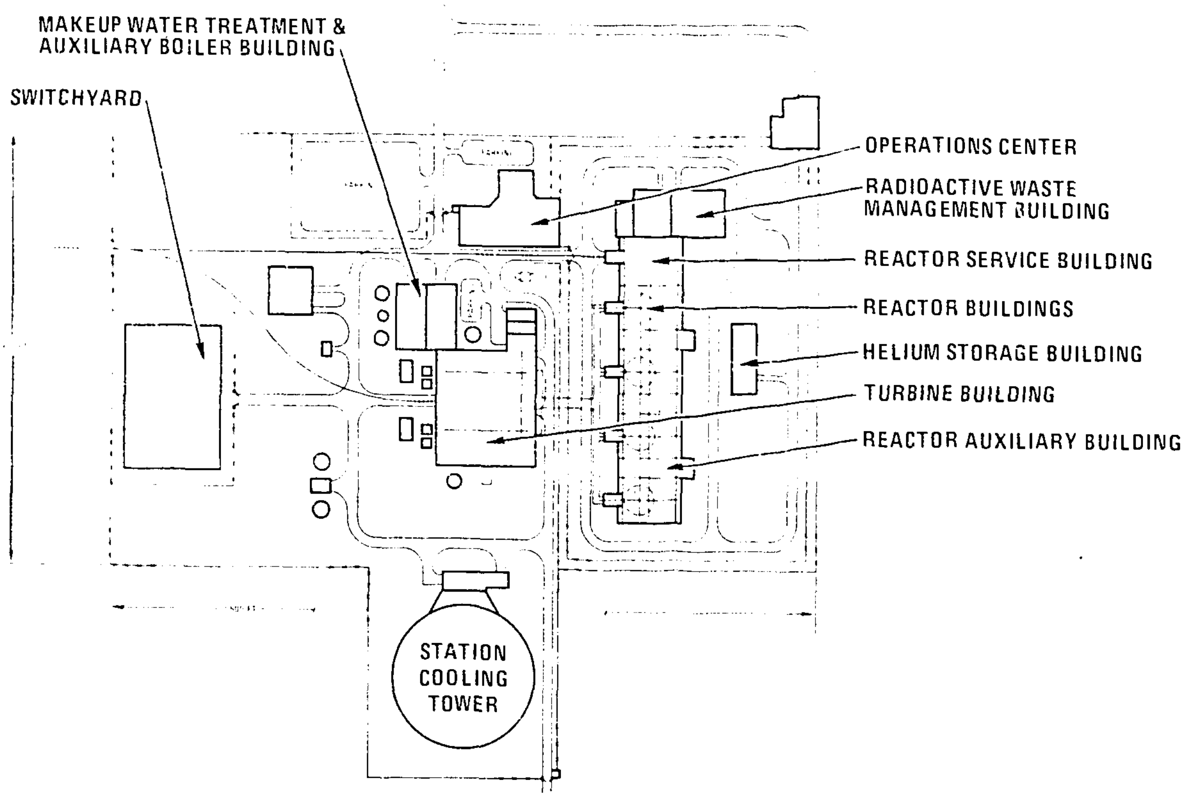

FIGURE 1. FOUR UNIT MHTGR PLANT LAYOUT

active core region consists of stacks of prismatic graphite blocks that are hexagonal in cross section. The fuel consists of coated particles of 20% enriched fissile uranium oxycarbide (UCO) and fertile thorium oxide (ThO_2). The particles are bonded together in fuel rods and are contained within sealed vertical holes in the graphite fuel blocks, as shown in Figure 3. Vertical coolant holes pass through the graphite fuel blocks. The fuel elements are stacked in columns to make up an annular-shaped core having a radial thickness of about 1.8 m (6 ft). Unfueled graphite blocks surround the active core to form replaceable inner and outer radial, and upper and lower axial reflectors. At the outer periphery of the replaceable graphite blocks are permanent reflector blocks.

The prismatic core design is based on an improved version of the reactor fuel technology developed by GA Technologies and demonstrated at the Fort St. Vrain HTGR plant. The annular core configuration (Figure 4) increases the external core surface relative to its volume. This configuration enables the decay heat under any loss-of-cooling condition to dissipate by conduction and radiation through the steel vessel to the reactor enclosure without the fuel particles exceeding their design limit temperature.

The fuel cycle is a once-thru, three-year

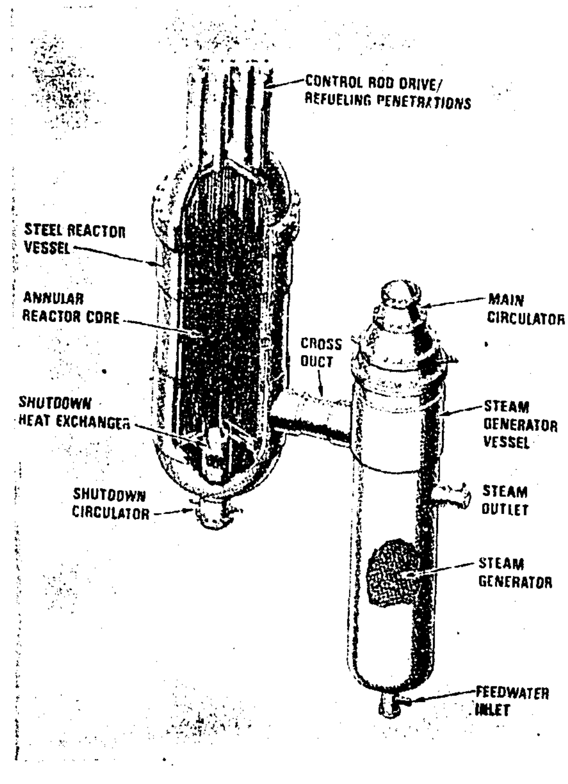

FIGURE 2. MHTGR SIDE-BY-SIDE VESSEL ARRANGEMENT

TABLE 2 SUMMARY DESIGN PARAMETERS OF THE REFERENCE FOUR-MODULE HTGR PLANT	
THERMAL POWER	1400MW(t)
ELECTRIC OUTPUT	538MW(e)
NET EFFICIENCY	38.4%
NUMBER OF FUEL COLUMNS	66
NUMBER OF FUEL ELEMENTS	660
NUMBER OF CONTROL RODS	24 IN OUTER REFLECTOR; 6 IN INNER REFLECTOR
RESERVE SHUTDOWN SYSTEM	12 IN INNER CORE ROW
FEEDWATER TEMPERATURE/PRESSURE	193°C (380°F)/20.68 MPa (3000 psia)
STEAM CONDITIONS	542°C (1005°F)/17.34 MPa (2515 psia)
CORE EXIT HELIUM TEMPERATURE	687°C (1268°F)
CORE INLET TEMPERATURE	258°C (497°F)
HELIUM PRESSURE AT CIRCULATOR DISCHARGE	6.38 MPa (925 psia)
CORE POWER DENSITY	5.9 W/cm^3
EQUILIBRIUM FUEL BURNUP	82,000 MW(d)/t

cycle with one-half of the core refueled every 1-1/2 years. Core reactivity is controlled by articulated control rods that travel in vertical channels located in the graphite reflector regions in both the center and outer periphery of the core. A reserve shutdown system, when manually activated from either the control room area or the remote shutdown area, permits boron material stored in hoppers located above the core to enter channels located in the innermost row of fuel columns.

The shutdown cooling system is contained within the bottom of the reactor vessel. Hot helium enters the water-cooled shutdown heat exchanger through a flow-actuated shutoff valve. The shutdown circulator discharges the gas into

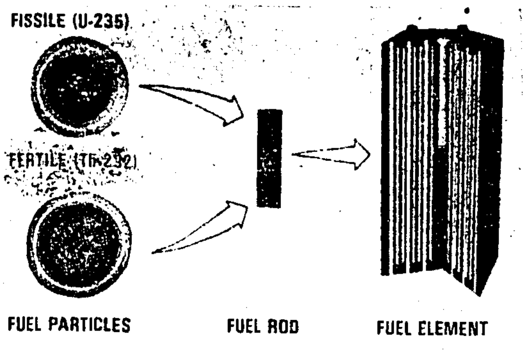

FIGURE 3. MHTGR FUEL COMPONENTS

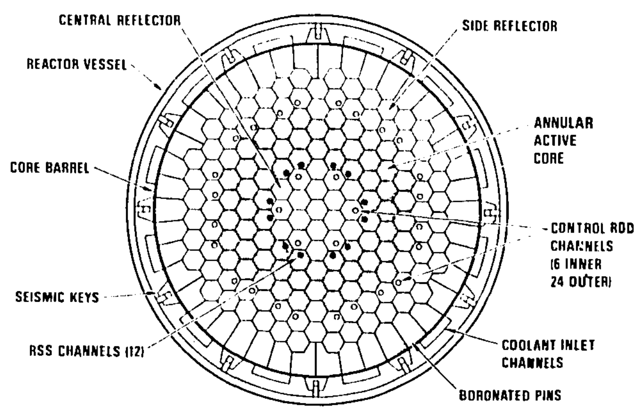

FIGURE 4. ANNULAR CORE AND CONTROL ROD LAYOUT

a plenum at the bottom of the reactor vessel, and the gas returns to the core inlet plenum by way of the outer flow passages inside the reactor vessel.

The heat transport system, located within the 4.2 m (14.0 ft) diameter, 26.5 m (87 ft) long steam generator vessel, is driven by the main circulator, located at the upper end of the vessel. The steam generator bundle, made up of helically coiled tubes, employs uphill boiling. Feedwater enters through a bottom-mounted tube-sheet, and superheated steam leaves through a side-mounted tubesheet at the upper end of the tube bundle.

If neither the heat transport system nor the shutdown cooling system is available for decay heat removal, the reactor cavity cooling system (RCCS) is used to reject decay heat from the core. Core decay heat is transferred by radiation and conduction through the prismatic graphite fuel and radial reflector blocks, the reactor vessel, and then to the RCCS. The RCCS consists of cooling panels located on the inside surface of the reactor enclosure. The flow of air by natural convection through these panels provides an entirely passive means of rejecting heat to the atmosphere.

SAFETY ASSESSMENT

Adequate nuclear safety is assured for all licensed nuclear power systems through the combination of inherent and engineered safety features. The key safety feature of the MHTGR design is the abil.cy to withstand all licensing bases events without relying on any operator actions or powered systems other than those that are battery powered. The annular core geometry, core power density, and the total module power have been chosen such that the decay heat generated within the core can be passively removed by means of conduction, radiation, and natural convection to surrounding passive media without a significant release of fission products from the coated particle fuel. This is due solely to the tolerance of the fuel and the graphite in the reactor core to withstand high temperatures and the large thermal inertia of the graphite moderator, characteristics which are unique to the HTGR. Over twenty years of fuel development and fuel testing have confirmed the high temperature tolerance of the refractory-coated fuel. As shown in Figure 5, data for HTGR fuel particles show that at temperatures up to about 2300°C (4172°F), the temperature-caused fuel failure fraction is essentially nil.

Moreover, the close physical coupling of the fuel and moderator and the use of a gaseous coolant in the MHTGR assures large negative temperature coefficients of reactivity, forcing the reactor into a shutdown condition during any accidental transient which raises the temperatures above the normal operating range.

A full spectrum of accident sequences and their consequences have been analyzed and submitted to the NRC for formal review. One of the limiting and most severe events considered was the complete loss of all active cooling systems for decay heat removal simultaneous with a depressurization or loss-of-coolant. As shown in Figure 6, the maximum core temperatures would be limited to about 1600°C which limits fission product releases to comply with the utility/user's emergency planning requirement of precluding off-site evacuation and sheltering plans for the MHTGR.

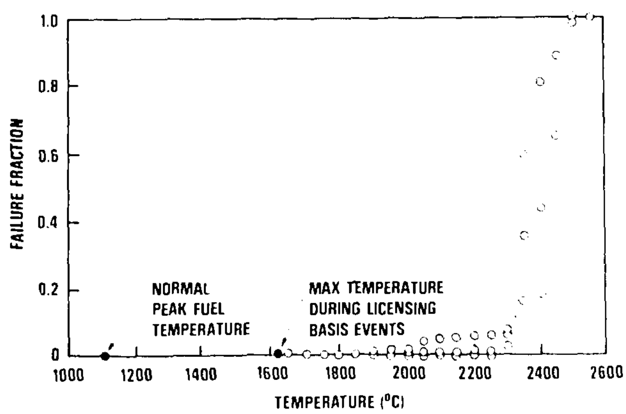

FIGURE 5. COATED FUEL PARTICLE INTEGRITY AT HIGH TEMPERATURES

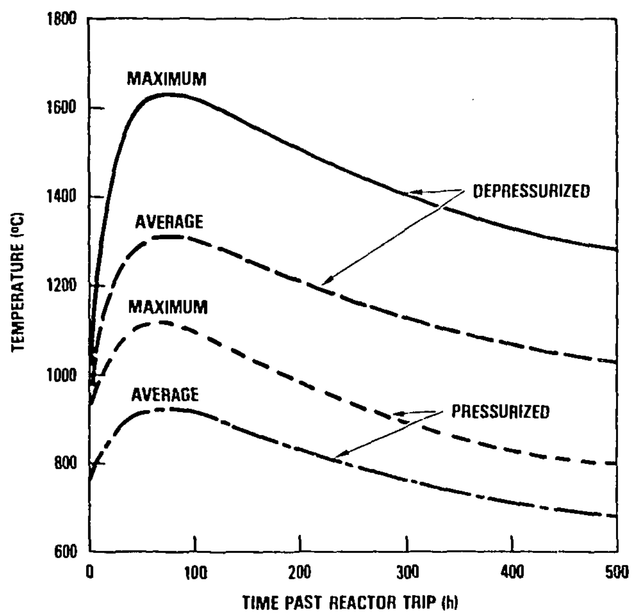

FIGURE 6. MHTGR TRANSIENTS DURING PRESSURIZED AND DEPRESSURIZED CONDUCTION COOLDOWN

PROJECT DEVELOPMENT

At present, the DOE-funded effort on the reference four module plant is concentrated on advancing the design and licensing development plus the associated technology development. In parallel and in support of this effort, the utility and vendor participants are pursuing a Project initiative for the initial prototype/lead commercial plant.

A Project Definition Study was carried out in 1986 by GCRA and the vendor and AE participants in order to establish an initial definition of the requirements and scope, the costs and schedule, and the overall benefits and limitations of an initial MHTGR Project. Through this effort, an optimum approach for an initial Project has evolved. It is based on an initial limited commitment for a one-module prototype plant that provides performance results for the subsequent commitment for expansion to a four-module lead commercial plant. Ongoing Project activities are focused on identifying and developing multiple sites and applications as options for such a prototype/lead commercial plant in parallel with the development of cost/risk sharing arrangements as the bases for an MHTGR Project initiative and proposal for Government support.

CONCLUSIONS

The MHTGR appears capable of providing key benefits that specifically address the technical, economic and policy issues that have continued to plague the nuclear power industry in the United States over the past decade. In particular, the smaller, modular passively-safe nature of the concept is expected to offer:

- Eased licensing and siting as well as enhanced public acceptance.
- Greater investment protection for owners as a result of the long time constants and the overall forgiving nature of the reactor system in responding to equipment malfunctions and operator error.
- Reduced capital and operating costs associated with the elimination of many complex, safety-related engineered components and systems.
- Reduced field labor, construction schedule and cost uncertainties due to the large content of shop fabrication.
- Easier financing and less financial risk due to the lower capital requirements associated with the flexibility of incrementally building to meet load growth requirements.
- Simpler, economic fuel cycle without the need for recycle or concern for proliferation.

These same benefits would also appear to ideally mesh with the requirements for the deployment and/or expansion of nuclear power in other countries who foresee the need to reduce their dependence on imported fossil energy sources, yet have been reluctant to consider nuclear energy due to the demands and complexities associated with the presently available technology.