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NUCLEAR ISLAND ENGINEERING MHTGR PRELIMINARY & FINAL DESIGNS

Technical Progress Report
for the Period Dec. 12, 1988 through Sept. 30, 1989

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AUTHORS/CONTRACTORS

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FOR THE DEPARTMENT OF ENERGY
CONTRACT DE-AC03-89SF17885

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**NUCLEAR ISLAND ENGINEERING
MHTGR PRELIMINARY & FINAL DESIGNS**
Technical Progress Report
for the Period Dec. 12, 1988 through Sept. 30, 1989

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1. INTRODUCTION AND SUMMARY

This report summarizes the Department of Energy (DOE)-funded work performed by General Atomics (GA) under the Nuclear Island Engineering (NIE)-Modular High-Temperature Gas-cooled Reactor (MHTGR) Preliminary and Final Designs Contract DE-AC03-89SF17885 for the period December 12, 1988 through September 30, 1989. This reporting period is the first (partial) fiscal year of the 5-year contract performance period.

The objective of DOE's MHTGR program is to advance the design from the conceptual design phase into preliminary design and then on to final design in support of the Nuclear Regulatory Commission's (NRC's) design review and approval of the MHTGR concept. GA's scope of work, as a member of the MHTGR Design Team, is focused on the Nuclear Island portion of the technology and design, primarily in the areas of the reactor and internals, fuel characteristics and fuel fabrication, helium services systems, reactor protection, shutdown cooling, circulator design, and refueling system. Maintenance and implementation of the functional methodology, plant-level analysis, support for probabilistic risk assessment, quality assurance, operations, and reliability/availability assessments are included in GA's scope of work.

In FY-89, the authorized funding was approximately 20% of the contract cost plan, consequently a significant fraction of the preliminary design was deferred into subsequent years. The preliminary design effort, performed under the constrained funding, was directed at supporting the NRC review of the MHTGR concept. The objective of obtaining a Safety Evaluation Report (SER) was achieved with the issue of a draft SER by the NRC in February 1989. The preliminary design of the reference MHTGR plant continued with emphasis on addressing the key technical issues identified during the conceptual design.

Support of base technology activities continued in the areas of graphite and metals characterization, fuel process development, fuel materials qualification, and fission product behavior.

The specific scope of work performed by GA under this contract was as prescribed in the Summary Level HTGR Program Plan (SLPP) for FY-89 (DOE-HTGR-88225, Revision 3).

The following sections of this report summarize the technical accomplishments for the reporting period, organized by Work Breakdown Structure (WBS) consistent with the SLPP: Section 2 - HTGR Technology (WBS 1000); Section 3 - MHTGR Design (WBS 5000); and Section 4 - Program Support (WBS 9000). Section 5 provides a listing of documents representing SLPP milestone deliverables for FY-89.

2. HTGR TECHNOLOGY (WBS 1000)

The objective during this contract reporting period was to continue support for ongoing HTGR technology activities, including HTGR base technology program support, experimental design support, and technology transfer from external programs.

2.1. TECHNOLOGY TRANSFER (WBS 1400)

The primary objective of this task is to maximize the HTGR technology exchange with foreign organizations having parallel programs and operating reactors. Technology transfer and data exchange were carried out under the auspices of the United States/Federal Republic of Germany (US/FRG) Umbrella Agreement for collaboration on gas-cooled reactor development, the USDOE/Japan Atomic Energy Research Institute (JAERI) Technology Exchange, and the USDOE/United Kingdom Atomic Energy Authority (UKAEA) Graphite Technology Exchange.

US and Foreign Countries Cooperative Program (WBS 1400.1)

A GA representative attended the US/FRG subprogram managers' meeting [May 22-26, 1989 at Kernforschungsanlage Julich GmbH (KFA)] held to review status of the Arbeitsgemeinschaft Versuchsreaktor (AVR) test program.

The US/FRG subprogram managers' meeting [June 6-7, 1989 at Oak Ridge National Laboratory (ORNL)] was held to review cooperative programs on fuel, fission products, and graphite pertinent to MHTGR development. GA participated in an experts' meeting on fuel performance modeling held in conjunction with the subprogram managers' meeting. Three presentations were given by GA on requirements for fuel materials

performance, update of the fuel performance model, and results of a joint comparison of US and FRG predictions of radionuclide release during accident conditions performed under the US/FRG safety subprogram.

GA, KFA, and ORNL prepared a joint paper entitled, "Modeling of Fission Product Release from HTR Fuel for Risk Analyses," which was presented at the Post-SMIRT conference seminar on small- and medium-sized nuclear reactors, held on August 21-23, 1989 in San Diego. The paper compares US and FRG fuel failure and release models.

The US/UK Graphite Technology group meeting (June 20-22, 1989 at ORNL) was held to exchange graphite creep and strength data. The US and UK theoretical approaches to creep behavior are very similar with only minor differences that can be resolved through the technology exchange.

A meeting was held in March 1989 at GA with a representative of JAERI to discuss testing needs for the Toshiba fissions chamber for use in in-core flux mapping in the HTGR. It was determined that the JAERI requirements for the detectors can possibly be met by testing in the TRIGA.

2.2. HTGR BASE TECHNOLOGY (WBS 1600)

The primary objective of the HTGR Base Technology is to develop information for design on fuel/fission products, graphite, and metals. The major focus is to confirm assumptions and technology behavior used in the design of the MHTGR.

Fuel/Fission Products Base Technology (WBS 1601)

Fuel Materials Development (WBS 1601.1). The joint US/FRG fuel performance model was revised and the report issued. The revised model improves the agreement between the model prediction and the data base. The empirical data showed that silicon carbide (SiC) decomposition,

rather than SiC corrosion by fission products, was the dominant failure mechanism under long-term high temperature accident conditions. The model for that effect was made part of the revision. In addition, 6×10^{-6} fraction was established as the minimum predicted failure during an accident event to recognize that statistically significant data base only supports predictions greater than 6×10^{-6} .

In support of ORNL's planned tests to obtain fission product release data from fuel under representative normal operating conditions, the test specification for the irradiation of capsule HRB-21 in High-Flux Isotope Reactor (HFIR) was issued. The capsule HRB-21 preirradiation report was revised. The revised report provides the rationale for the statistical relevance of the test and accounts for differences between the particles as specified and current MHTGR specifications revised after the fuel was fabricated.

Fission Product Performance (WBS 1601.2). The final fuel specification for the compacts which contain "designed-to-fail" particles to be used in the COMEDIE BD-1 test was issued. The COMEDIE BD-1 test specification was updated to resolve Commissariat a l'Energie Atomique (CEA) comments and the revised test specification issued. The export license to ship fuel to CEA for use in the COMEDIE tests was obtained from NRC. The shipment consists of coated particle samples for flux mapping tests and the BD-1 fuel compacts.

GA continued to support the capsule HFR-B1 irradiation tests being performed at the Petten Joint Research Center for the purpose of obtaining experimental data to quantify the effects of moisture on gaseous fission product release from, and on metallic fission product transport through, the fuel compact matrix and graphite.

The postirradiation examination (PIE) test specification for capsule HFR-B1 was issued. The document describes the measurements and documentation required for the PIE to be conducted in the Petten and KFA

hot cells. It also includes test requirements for postirradiation heating work to be conducted on irradiated samples at ORNL.

The pretest analysis report for the planned AVR depressurization tests was revised to reflect the predicted results for the AVR HTA-6 K1 and H1 tests using updated test parameters and sequences. HTA-6 K1 is a "cold" test in which the reactor is initially shutdown and the circulator is tripped at the same time depressurization begins; HTA-6 H1 is a "hot" test in which the reactor is at approximately 65% power and the depressurization is initiated 10 seconds, after circulator trip. The report provides the pretest predictions which indicate that maximum shear ratios in the primary circuit will not exceed unity in either the K1 or the H1 depressurization tests. Performance of the AVR HTA-6 tests is subject to receipt by KFA of the licensing approval from FRG regulatory authorities which had not been received as of September 30, 1989.

Fuel/Fission Products Engineering (WBS 1601.4). A draft test specification for plateout and lift-off tests in the out-of-pile loop ("DABLE") to be constructed at Massachusetts Institute of Technology (MIT) under subcontract to ORNL was completed. MIT will construct a high-quality test facility and will perform a limited number of well characterized plateout, lift-off, wash-off, and dust effects tests.

Graphite Base Technology (WBS 1602)

Graphite Engineering (WBS 1602.3). The Graphite Technology Development Plan was updated in those sections for which GA had prime responsibility; revised sections were transmitted to ORNL and DOE by letter report. The draft test specification for reactor core graphite was completed. The test specification includes the technology development examinations and tests necessary to define the corrosion of graphite used in the reactor system.

Metals Base Technology (WBS 1603)

Materials Engineering (WBS 1603.1). The Metals Technology Development Plan was updated in those sections for which GA had prime responsibility; revised sections were transmitted to ORNL and DOE by letter report. Changes include the deletion of certain Alloy 800H tests described in existing design data needs (DDNs) because either the design has changed or the required data are available from other programs.

2.3. SUPPORTING TECHNOLOGY (WBS 1700)

Work under this technology task includes the development of design data, validation and verification of design/analysis methods and criteria, and component testing to support the design definition, evaluation, and licensing of the MHTGR.

Reactor System Design Support (WBS 1711)

Reactor Core Design Support (WBS 1711.3). The core fluctuation test specification and test procedure were completed. Design, fabrication, and assembly of the core fluctuation test rig were initiated. The needed refurbishment of existing parts of the 1/4-scale test apparatus used for the large high-temperature gas-cooled reactor (LHTGR) core fluctuation model test was defined and additional components were designed. Design of the test apparatus was completed with the issue of drawings and the design report. Purchase orders were placed for the fabrication of the prototype model fuel element; checkout, repair, and calibration of the data acquisition system; and supply test rig components. Initial assembly of the test apparatus was underway at the end of FY-89; a status report on the core fluctuation test was issued.

Fuel Process Development (WBS 1711.5). A draft procedure for SiC defect detection using the high pressure mercury intrusion technique was issued. The technique is applicable to detection of microscopic SiC

flaws which release metallic fission products but are too small for detection by burn-leach methods. The procedure for detection of missing buffer layers using the automatic image analyzer (AIA) was issued. The procedure includes the preparation of radiographs and the operation of the AIA to identify defective particles. The equipment was qualified by 20 separate tests where radiographs of 10,000 particles were analyzed by a human operator and then by AIA. The same particles were identified as defective by both techniques each time and no incorrect indication of defective particles were made. These results met the requirement for 95% confidence that no more than 5% of particles with missing buffers would go undetected.

The coating process procedure for using the B_4C poisoned draft tube was issued after approval by nuclear safety and quality control. The procedure applies to coating of up to 5 kg of 20% enriched uranium in fuel particles. This capacity meets the technology program goal.

Assembly of the ThO_2 kernel process demonstration facility was initiated. Purchase orders were placed for a steam denigrator, blow back system, three sol preparation tanks, enclosures, and shielding cabinets. The enclosure and lead shielding for the denigration line were installed. The hoods, columns, and denitration vessel were assembled.

Heat Transport System Design Support (WBS 1713.1). The air-flow test specification, which defines the tests required to predict the performance of the hot portion of the primary coolant circuit during plant operations with the heat transport system or the shutdown cooling system, was issued.

3. MHTGR DESIGN (WBS 5000)

The objective during this contract reporting period was to continue development of the MHTGR preliminary design as a basis for technology program definition and support for licensing activities. MHTGR design-related tasks include substantiating analyses, technical inputs to design and licensing documentation, identification of DDNs, and review of corresponding technology development plans, design management and cost estimating activities.

3.1. MHTGR PLANT-LEVEL DESIGN AND ANALYSIS (WBS 5100)

Development and documentation of plant requirements through functional analysis and allocation of requirements to system to support the plant design were performed under this task. Plant-level and multisystem analyses as required to support development of design requirements, interfaces, and selection of design alternatives, and plant-level assessments of the reliability, availability, safety, and investment protection aspects of the MHTGR design were performed.

Plant-Level Design and Integration (WBS 5101)

Plant-Level Design and Integration (WBS 5101.1). The MHTGR reactor module general arrangement drawing, which integrates the individual reactor component subassemblies within the vessel system and verifies that the various system and component interfaces match dimensionally and that primary coolant flow requirements are addressed properly, was updated to reflect current reactor component designs. Revisions include reactor vessel bottom head changes due to shutdown cooling system component design changes and reactor core changes (addition of boron shielding at top and side of core).

A partial redirection of task effort was implemented in accordance with a DOE request to perform a containment system trade study. Selected for initial study were five alternative containment designs: (1) vented, moderate leakage building; (2) vented, low leakage building; (3) vented, low leakage, filtered building; (4) low pressure, low leakage building; and (5) moderate pressure, low leakage building. The definition of the five containment concepts was subsequently expanded to consider a broader range of alternatives to include moderate and high pressure buildings with air-cooled and water-cooled reactor cavity cooling systems.

Each alternative containment concept was reviewed to determine the design basis event for the building. The likelihood of major steam and feedwater leaks within the MHTGR reference reactor building was evaluated, and it was concluded that major feedwater leaks can be expected to occur greater than 1×10^{-4} per year. Major steam line leaks are below the design basis provided that no steam isolation valves are located within the building. It was recommended that a 13 in.² opening for primary coolant depressurization be the design basis for all alternative containment buildings. This event was selected since it results in the highest building pressures above a frequency of 1×10^{-4} per year. Helium, steam, and feedwater blowdown rates were calculated for use in assessment of building pressure transients for each alternative concept.

The safety and cost-to-benefit impact of the selected containment alternatives were evaluated. The safety evaluation showed that the reference MHTGR (Alternative 1) and all the other containment alternatives evaluated in this study meet all the safety requirements with large margin. The risk of latent cancer fatality for the reference MHTGR, conservatively calculated (i.e., without credit for containment) for an individual standing at the site boundary for the duration of the accident, is several orders of magnitude lower than not only the cancer

risk from natural background radiation exposure, but also the safety risk goal, which is one thousandth of the cancer risk from all causes including background exposure. The other alternatives meet safety requirements with larger margins. However, any additional reduction in risk obtained with respect to Alternative 1 is not measurable and has no meaningful safety benefit.

The cost-to-benefit assessment showed that relative to the reference design, the cost-to-benefit ratio for all the other alternatives is very large (>one billion dollars per man-rem) as the potential gain in dose reduction from Alternative 1 is very small. In accordance with the study constraints, this assessment considered only capital cost. In practice there would also be significant impact of operating and maintenance (O&M) costs due to the imposition of a containment.

These results were documented in the MHTGR Containment Study Report issued to DOE in June 1989.

Plant-Level Analysis (WBS 5102)

Plant-Level Analysis (WBS 5102.1). The primary coolant chemistry control requirements document was issued. The primary coolant chemistry control requirements were developed using a top-down approach. In this approach, the plant design was assessed for potential sources of primary coolant contaminants. Once the sources were established, requirements were derived for ingress detection and termination, for maximum allowed concentrations of contaminants, and for primary coolant cleanup. Requirements were established based on compliance with plant-level requirements and the ability of graphite and metallic components to withstand the level of contaminants within the primary coolant environment. The contaminants covered by the requirements were oxygen, water, carbon dioxide, carbon monoxide, hydrogen, methane, hydrocarbons, nitrogen, chlorine gases, sulfur gases, and oil vapor. Carbon dust and rust

particles were also included. Requirements were developed for both operating and shutdown conditions.

A design change proposal on the thermal performance envelope requirements at 25% feedwater flow, applicable to plant configurations, including 2 (2 x 1), 4 x 1, and 4 (1 x 1), was prepared and submitted to Plant Design Control Office (PDCO) for approval. These revised requirements allow a single standard reactor module design to accommodate the performance characteristics of each of the reactor/turbine-generator configuration of the overall plant design specification (OPDS). (Note: PDCO has withheld approval pending further review and evaluation of the design change proposal.)

Plant Dynamic Analysis (WBS 5102.2). A design change proposal on the revised plant design duty cycle report (DOE-HTGR-86029, Revision 3) was issued for program participant review/concurrence. The plant design duty cycle events and their design number of occurrences provide requirements for the design of plant structures, systems, and components to ensure that the plant-level goals are met. The duty cycle events encompass a range from normal operation to off-normal/accident conditions. Event categories were derived by identifying the frequency ranges and allocating the OPDS top-level requirements. Specific design duty cycle events were allocated into the event categories based on plant-level requirements and plant assessments which relate to plant performance, scheduled outage, forced outage, investment risk, and safety risk. Design number of occurrences were specified to assure that the expected number of occurrences of all events will be accommodated in the plant design.

Plant Seismic Analysis (WBS 5102.3). In support of the task to perform an updated plant seismic analysis for the MHTGR by Bechtel National Inc. (BNI), the core seismic model was updated to incorporate latest design changes associated with the core and reactor internals.

The revised seismic model was incorporated in the nuclear steam supply system (NSSS) seismic model.

Decay Heat Removal Analysis (WBS 5102.6). The description of the two-dimensional fluid flow/heat transfer SINDA-FLUINT model of the reactor vessel and internals, cross vessel and steam generator vessel, developed to model both pressurized and depressurized conduction cool-downs in the MHTGR, was documented. The heat transfer model incorporates conduction, thermal radiation, and both forced and natural circulation heat transfer where appropriate. The core is modeled using three average channels for power distribution, heat transfer, and fluid flow.

Availability/Reliability Assessments (WBS 5103)

Availability/Reliability Assessments (WBS 5103.1). Allocation of plant-level availability and investment protection requirements was completed. A report was prepared which provides a structure for deriving lower level requirements of plant performance in terms of functional success, failure, and recovery criteria in a top-down fashion. These lower level requirements, when met by the design, ensure compliance with the top-level requirements of Goal 2. The report describes the allocation methodology and contains a general format for the documentation of requirements.

Development of the statistical uncertainty model for depressurized conduction cooldown was completed. The mechanistic model is a finite difference one-dimensional approximation of the reactor core, internals, and vessel.

Nuclear Island (NI) Scheduled Outage (WBS 5103.2). Input to the NI Scheduled Outage Assessment Report was completed. Input was prepared in the form of annotated data sheets. Each data sheet included a brief description of the specific scheduled outage activity, component

approach and access, personnel and calendar time requirements, and impacts and constraints.

Operations, Maintenance, and In-Service Inspection (WBS 5104)

Operations Assessments (WBS 5104.1). Development of shutdown/startup sequence following heat transport system loop trip-off (Goal 2) in one module while operating at full load was completed. The loop trip-off can be triggered by several causes, such as, failure of electronic panels and devices external to the main circulator, failure of circulator motor water cooling pump, fittings and valves, and water ingress due to steam generator tube failure. In each event, the reactor is tripped-off and the shutdown cooling system is automatically actuated, and the restart depends upon the repair/replacement effort and module status. Plant level control action and manual interaction by operators as appropriate were identified for the restart.

Maintenance Assessments (WBS 5104.2). Development of the maintenance sequence for removal/replacement of the main and shutdown circulators was completed. The main circulator removal/replacement process is estimated to take 50 calendar hours with a crew of three and a total dose exposure of 0.2 man-rem per event. The shutdown cooling system circulator, which is located at the bottom of the reactor vessel is estimated to be removed and replaced in 59 calendar hours with a total personnel dose exposure of 0.3 man-rem per event.

In-Service Inspection (ISI) (WBS 5104.3). Development of ISI sequences was completed for reactor internals graphite components (permanent side reflector and core support structure) and metallic components (upper plenum shroud, core lateral restraint, and core support structure). These ISI sequences include the ISI requirements, type of examination, steps in the sequences, and the radiation dose. Special equipment needed to perform these examinations was identified.

Occupational Radiation Exposure Assessment (WBS 5104.4). Input to the Onsite Radiation Protection Design Manual was completed. Several sections of the planned manual were prepared containing information on: material exposure limits, neutron and gamma sources, activated components, primary coolant sources, personnel access requirements, radioactivity analysis codes, and list of future radiation protection tasks. The manual will provide a common basis for all participants in this task (GA, BNI, and ORNL) to develop the personnel radiation doses and plant radiation zones.

Control of Radionuclide Release (WBS 5106)

Safety Consequence Assessment (WBS 5106.1). Qualitative and quantitative comparisons of the US and FRG fuel failure and radionuclide release models as developed at GA and at KFA/ISF were completed and reported. The work was conducted under PWS S-6, "Fission Product Retention in Fuel," as part of the Safety Research Subprogram Plan of the US/FRG Umbrella Agreement. Results of this work have provided a better understanding of the differences in US and FRG release models, leading to the development of consistent models. The comparison of the exposed kernel release models and the US/FRG standard particle failure and retention models provides valuable support to the verification and validation of those models within the independently developed computer codes of GA and KFA/ISF.

Preparation of the POLO User's Manual was completed. The POLO computer code determines the initial and time-dependent activity sources transported from the vessel to the reactor building for use in dose assessment calculations.

Probabilistic Risk Assessment (WBS 5106.2). The report on the allocation of investment and safety requirements for the MHTGR was issued. Based on plant-level analyses assessing the MHTGR's compliance with the top-level requirements quantifying Goals 2 and 3, key functions

of the plant were determined and the required "functional performance" determined at the system level. This requisite performance is allocated, as lower-level requirements, to the various systems where it provides the direction ensuring design efforts are in fact addressing the top-level goals. In addition to a listing of allocated requirements, the document includes definition of key terms used in the plant-level assessments and allocations, a description of the methodology, and a requirements traceability index.

The informal design review conducted by PDCO and its consultants of the forced outage and investment risk areas was supported. Review comments were reflected in the preparation of the aforementioned document.

3.2. MHTGR SYSTEM DESIGN (WBS 5200)

Preliminary design of the MHTGR continued at the system-level and below. System-level functional analysis, trade studies, design, design analysis, and documentation were performed on Type 1 (necessary for licensing or requiring technology program support) systems and structures. Work on Type 2 (has major interfaces with Type 1 and/or major influence on plant capital cost) systems and structures was limited to establishing design interfaces with Type 1 systems and structures. No work was performed on Type 3 (other systems) systems and structures.

Reactor System Design (WBS 5211)

Reactor System Design (WBS 5211.1). An evaluation of the effect of fuel block bowing and the resulting nonuniform fuel column gaps on core primary coolant flow distribution was completed. In this analysis, which was performed as part of the overall core thermal/hydraulic analysis, the effect of gravity forces on columns of bowed blocks in causing the gaps to collect at specific locations in the core were calculated. The results show that the nonuniform gaps result in an increase in total gap flow compared to the idealized case of uniform

gaps by only a small amount (approximately 1% in the top part of the active core and decreasing to essentially no difference at the core exit). This small difference would not affect significantly fuel temperatures or core exit hot streaks. Locally, however, the gap flows were as much as five times higher for the nonuniform gap case, which could affect fuel element stresses.

The core temperature measurement requirements trade study was completed. The conclusion from the study is that there is no compelling justification for additional core instrumentation for normal operation or licensing basis events.

The reactor system arrangement drawings were updated. The drawings were updated to incorporate changes as a result of trade studies and analysis of the reactor system and components.

Neutron Control Design (WBS 5211.2). Requirements for the control rod drive motor controller design were developed based on data provided by suppliers. Acceptable velocity control accuracy necessitates use of a brushless tachometer as a feedback device for the controller. The use of the controller to hold the control rod in a fixed position potentially leads to rod drift. This fixation may be separated from the controller and performed with a separate holding circuit. Additional evaluations are planned.

Metallic Reactor Internals Design (WBS 5211.3). The fluid flow leakage and thermal analysis of the metallic core support structure was completed. Results show that peak metal temperatures approach 427°C (800°F). Provisions for cooling flow will be incorporated into the metallic core support structure configuration.

Drawings of the metallic reactor internals were revised. The metallic core support structure was modified to incorporate the design of the interface flange to the shutdown cooling system heat exchanger

shroud. The upper plenum shroud was modified to reflect the addition of neutron shielding to meet vessel top head fluence limits.

Graphite Reactor Internals Design (WBS 5211.4). The core support stabilization trade study was completed. Several promising anti-fluctuation features were selected for incorporation into the core fluctuation test evaluation.

The graphite reactor internals drawings were updated to incorporate design evolutionary changes. The following drawings were revised: graphite reactor internals arrangement, permanent side reflector assembly, and reactor core support structure assembly.

The stress analysis of the MHTGR graphite reactor internals to assess oxidation effects was completed. The results of the analysis showed that the graphite core support meets stress requirements when continuously exposed to the design requirement of 2.3 ppmv H₂O . (A similar analysis of the graphite fuel elements concluded that the fuel elements meet stress requirements when continuously exposed to 3.5 ppmv H₂O.)

Reactor Core Design (WBS 5211.5). The core stabilization trade study to develop upper plenum constraint concepts to preclude core fluctuation was completed. The selected concepts will be included in the core fluctuation test evaluation. The core (fuel/reflector block) thermal/hydraulic analysis to establish lateral pressure forces on the core columns was completed. Results show that the lateral pressure forces are low. Estimates of stability indicate that the columns will not fluctuate.

The reactor core arrangement drawings were revised.

The structural design criteria document for replaceable graphite core elements was updated to incorporate more detailed design requirements and modifications to the design limits. The modifications to the design limits are based on the conclusions of the damage model study which was completed earlier in the year.

The MHTGR fuel product specification was updated. The most significant changes were the requirement for a 95% (instead of 50%) confidence level on the segment mean particle defect fractions and the addition of the protective outer pyrolytic carbon (OPyC) coating layer to the particle.

The core physics validation plan was revised to reflect comments resulting from the informal PDCO and participant review. Available physics data were reviewed and ranked as to their usefulness in the current validation program. Physics evaluations of available DRAGON reactor experiment measurements have been initiated. A joint GA/AVR report describing AVR physics measurements, measurement techniques, and uncertainties was completed.

A fuel performance analysis of the Fort St. Vrain (FSV) core was performed using the reference fuel performance and fission gas release methods. The purpose of the analysis was to compare the predicted fission gas release with data taken as part of the FSV radiochemistry surveillance program. The comparison of results, which shows very good agreement between the predicted and measured fission gas release for the key isotopes Kr-85m and Xe-138 during the entire operating period, supports validation of the reference GA fuel failure and fission gas release methods.

The validity of plateout distribution methods was assessed using plateout data from the FRG LAMINAR out-of-pile loop and JAERI in-pile loop OGL-1. The agreement between the predictions and the measured data

was within the required factor of ten for both the LAMINAR and OGL-1 loops.

The core nuclear design report was revised to include power distribution and control analytical results.

An analysis of radionuclide control for the MHTGR was completed. The results of this analysis indicated that the overall fuel performance of the MHTGR preliminary core design was generally acceptable except that the predicted releases of Cs and Ag exceeded the core release criteria. Due to changes in the axial power profiles and increases in the core bypass flow fraction, the predicted fuel and graphite temperatures for the bottom half of the core are higher than for the previous conceptual design, resulting in the higher fission metal release. Acceptable design alternatives will be developed in the course of continuing design effort.

The calculations of nominal fuel hydrolysis and fuel element graphite burnoff were completed.

Reactor Service Equipment (WBS 5211.6). Top-level preliminary design layouts for the auxiliary service cask and transporter assembly were completed.

Heat Transport System Design (WBS 5213)

Heat Transport System (HTS) Design (WBS 5213.2). Update of the HTS interface control document, incorporating current design configuration on the interface drawings, was completed.

The sensitivity study of hot streaks at the core exit, the hot duct entrance, and the steam generator entrance to bypass flow at various locations was completed. Results show that predominant bypass flows are from the cold leg to the hot leg. One bypass flow from the hot leg to

the cold leg is from the steam generator inlet to the steam generator upper internals at the tube sheet/vessel interface. This bypass flow has potential for local hot streak impingement on the vessel wall; an effective seal needs to be designed to control this bypass flow.

Main Circulator Design (WBS 5213.3). James Howden and Company issued the circulator design status report. James Howden and its sub-contractors have advanced the design of the main circulator subcomponents (e.g., motor and bearing system) and completed a preliminary overall machine layout. Compressor performance maps were completed for pressurized and depressurized modes of operation. The thermal analysis of the circulator within the vessel confirmed that cooling of the motor is satisfactory at all operating conditions.

Laurence Scott and Electromotors, the motor designer, have achieved the desired distance between bearing centers to meet rotor critical speed requirements, thus providing values for the first three critical speeds which allow Magnetic Bearing Inc. to define a suitable bearing control philosophy. The preliminary size and geometry of the rotor have been established. A preliminary selection of the magnetic bearing system has been made. Corresponding critical speed analysis, thermal loading, control and power, and interface data have been completed. Magnetic bearing geometry, rotor and motor details, and catcher bearing arrangement have been incorporated into the circulator general arrangement.

HTS Internals Design (WBS 5213.5). A study of the removability/replaceability of the hot duct was completed. Results of the hot duct removal/replacement study showed that the interface diameter between the circulator seal and the steam generator upper internals must be enlarged to the same diameter as the circulator penetration in the steam generator vessel to allow installation of the hot duct horizontal segments, bellows, and elbow. The proposed replaceability scheme requires that insulation on the inside of the horizontal section of the hot duct must

be removable independently from the horizontal cylindrical shell. Additional detailed study is planned.

Shutdown Cooling System Design (WBS 5214)

Shutdown Cooling System (SCS) Design (WBS 5214.1). The SCS assembly drawing and interface control document were completed. These drawings include the latest design for the bellows seal interface between the SCS and the reactor system, the flange interfaces between the SCS and the shutdown cooling water system (SCWS), and the space requirements for the SCS control equipment.

The SCS component removal/replacement study was completed. A review of the component removal procedures and the necessary handling equipment showed that removal and replacement of both the circulator and heat exchanger are feasible and economically practical.

As part of the study, the core bypass flow was calculated for HTS decay heat removal with the SCS circulator and the heat exchanger removed. For the case in which the SCS circulator is removed one day after shutdown, 57% of the main circulator flow bypasses the core, and the core outlet temperature is 414°C (777°F). If the removal of the circulator is done seven days after shutdown, the core bypass is reduced to 46% and the core outlet temperature is 167°C (333°F).

Removal and replacement of the SCS heat exchanger is not an expected operation as the component is designed for a 40-year life. However, if it is done seven days after reactor shutdown, the core bypass flow, with the SCS circulator also removed is 67% of the main circulator flow and the core outlet temperature is 221°C (430°F).

Additional work is planned to evaluate the foregoing replacement strategies versus availability goals.

Fuel Handling and Storage System Design (WBS 5221)

Fuel Handling System Design (WBS 5221.1). Layout drawings defining specific interfaces between the fuel handling components and the reactor building were completed. Layout drawings include: fuel handling equipment positioner study, floor valve installation layout, services assembly, and plug actuator study. Layout drawings defining the maximum operating capabilities of the fuel handling machine during in-core maintenance were also completed.

The structural analysis of the fuel handling equipment mounted over the reactor vessel was completed. Results were obtained for both static and dynamic loading conditions. Seismic loads, stresses, and deflections at major points of the fuel handling equipment and support points were established.

Helium Services System Design (WBS 5224)

Helium Purification Design (WBS 5224.1). The process flow diagram for the helium purification train was updated to include revisions to the flow, pressure, and temperature parameters under part load, shutdown, and refueling conditions.

Reactor Protection System Design (WBS 5231)

Reactor Protection System (RPS) Design (WBS 5231.1). A review of protection system compliance with independence and separation requirements was completed. The review concluded that: (1) the reference design involving separating the redundant channels of the RPS and investment protection system (IPS), and locating both the RPS and IPS redundant instrumentation cabinets in a single safety class structure (room) is adequate; (2) IEEE-384, "Independence Criteria," is consistent with the MHTGR top-level requirements for high reliability and will provide necessary criteria and guidance for use in subsequent detail

design; and (3) RG 1.75, "Physical Independence of Electric Systems," and RG 1.97, "Accident Monitoring," are out of date or inappropriate for MHTGR.

Results of analysis of selected plant duty cycle transients without protection (unprotected transients) to establish design basis requirements for the protection system were assessed. The transients which were analyzed include complete loss of main loop circulator from 100% and 25% module load. Based on the results, protection is required for the loss of feedwater transients.

Development of algorithms defining the RPS design for use in MHTGR plant dynamic analysis transients was completed.

Drawings of the RPS equipment and reactor service instrument cabinets, and their locations, were completed. A review of the RPS cabinet arrangement in the reactor building was completed. It was found that the cabinets cannot all be located at the (-) 12 ft-6 in. elevation due to space constraints. The general allocations given in the reference building design are reasonable, although the arrangement within the areas need refinement.

The RPS interface control document was revised to include updated sensor instrumentation interface requirements.

Investment Protection System Design (WBS 5233)

Investment Protection System (IPS) Design (WBS 5233.1). A review of requirements for status and bypass monitoring for the MHTGR protection systems was completed. The review concluded that: (1) the reference design for including the RPS and IPS status and bypass monitoring within the IPS is adequate; (2) the requirements for status and bypass monitoring given in IEEE-603, "Safety System Criteria" and IEEE-497, "Accident Monitoring Criteria," are appropriate for MHTGR; and (3) the

design selection for IPS status and bypass monitoring is a computer system.

Development of algorithms defining the IPS design for use in MHTGR plant dynamic analysis transients was completed.

A drawing of the IPS equipment and locations was completed. A review of the IPS cabinet arrangement in the reactor building was completed in conjunction with the RPS arrangement study.

An unprotected transient analysis was completed for loss of circulator, loss of feedwater, and the maximum increase in feedwater flow transients for cases beginning at 100% module load.

The IPS instrumentation block diagram was revised. Revisions include deletion of automatic primary coolant pumpdown and deletion of the valve redundancy for the shutdown cooling heat exchanger automatic isolation and drain.

The document, "Role of the Operator and Manual Control Capability in the MHTGR," was revised to incorporate participant review comments. The general conclusion remains unchanged, i.e., there is an economic incentive for locating the manual control and protection capability in the control room.

General accident monitoring requirements were established for the IPS to support the system design work in a manner that ensures that plant performance will comply with plant top-level goals. Plant-wide assessments, such as the safety and investment risk studies, which provide a measure of goal compliance were used to establish the IPS accident monitoring requirements. The IPS interface control document was updated to incorporate accident monitoring requirements.

Plant Control, Data, and Instrumentation System (PCDIS) Design
(WBS 5234)

Nuclear Island (NI) Control Design (WBS 5234.1). The NI control input to the PCDIS interface control document was updated to reflect latest space, power, and heating, ventilation, and air conditioning (HVAC) requirements of the NI control equipment.

The analysis of NI control enveloping transients was completed and documented. Seven transient events were analyzed to evaluate adequacy of control configuration/gains affected by the incorporation of revised system/component design data in the transient analysis model. Three events included limiting conditions for control rod operation to provide interfacing requirements data for the control rod design. Results show that defined control limits are well maintained during expected/nominal conditions and the behavior to abnormal conditions/malfunctions is very forgiving with the algorithms adopted. Portions of the analyses performed to determine requirements for the control rod system show significant time allowable for control rod bank reshimming under worst case assumptions. The control rod requirements analyses also indicated a potential for significant reduction of required rod speeds.

Development of NI control algorithms was completed. NI control algorithm development was performed to enhance the stability margin for turbine trip and reactor trip transients. A dual rate feedwater flow runback schedule following turbine trip and a steam generator inlet helium temperature compensation of the main steam temperature control gains following reactor trip were shown to substantially improve the control. The latter algorithm will also improve stability margin for other duty cycle events involving transient mismatch of reactor heat generation and heat removal rates.

Response data were prepared for control rods, feedwater-flow, and power change at both high and low load conditions. Data were

obtained for power, steam temperature, and steam pressure response to these inputs. The previously chosen configuration was substantiated by the response data. Only marginal controller improvement over results using previously established controller gains could be attained by gain adjustments to tune to the slightly shifted plant response. The need for rod control to accommodate long-term (Xenon) changes was established.

3.3. DESIGN MANAGEMENT AND COST DEVELOPMENT (WBS 5900)

Management direction, support, and control were performed for GA's scope of work in MHTGR-NIE program. Support for the development and maintenance of MHTGR program cost and schedule, including fuel-cycle economics, was provided.

MHTGR Design Management (WBS 5900.1). The MHTGR-NIE program management plan and the quality assurance program document for conducting MHTGR-NIE design activities at GA were issued soon after inception of contract.

4. PROGRAM SUPPORT (WBS 9000)

Program support was provided in developing annual and long-term plans for the MHTGR-NIE, participating in design reviews and assessments, integrating utility/user input within the MHTGR-NIE, licensing interaction with NRC and developing the PSSAR. Support of the PDCO continued.

Utility/User Requirements and Design Evaluation (WBS 9200)

Desalination Studies (WBS 9200.3). The MHTGR desalination study final report was issued. A further study using a vertical tube evaporator (VTE) in an MHTGR desalination plant was performed. Results of the study showed that the low temperature multieffect distillation (LT-MED) process was more promising than the VTE concept on the basis of suitability, plant performance, and water cost.

Licensing (WBS 9300)

Draft PSSAR (WBS 9300.2). The proposed method for developing the technical specifications for the MHTGR was assessed. The proposed methodology is consistent with the Integrated Approach, and provides a systematic approach to identifying the specification to ensure that the "as-operated" safety characteristics of the plant remain within the basis for the plant license, that is in compliance with 10CFR100.

The Top-Level Regulatory Criteria document was revised. The revisions include a general update of the document and some expansion of the dose and risk criteria to event (frequency) criteria.

The procedures used for determining Licensing Basis Events (LBEs) for the MHTGR during conceptual design were reviewed and compared to the current method. It was found that the basic procedure is valid and that only two significant changes pertaining to the factor to be applied to the frequency of events for certain analyses and the initial conditions for analysis of safety-related conditions would be incorporated in the modification of the procedure to provide more conservatism. Additionally, better explanation of the details of the methodology will improve its understandability in future applications. Assessment of the LBE selection methodology was completed.

The methodology for equipment classification was completed. An assessment of the motivations for having equipment classification, including the pros and cons of several alternatives was performed. While no compelling reason is found for the MHTGR program to adopt a safety-relevant classification scheme, the advantages and disadvantages underscore the net benefit derived from an equipment classification approach similar to that used in the Preliminary Safety Information Document.

5. DELIVERABLES

The documents listed in Table 5-1 constitute the SLPP milestone deliverables for Contract DE-AC03-89SF17885 for the contract performance period December 12, 1988 through September 30, 1989.

TABLE 5-1
LIST OF SLPP MILESTONE DELIVERABLES - FY-89
(Contract DE-AC03-89SF17885)

5-2

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
<u>HTGR Base Technology (WBS 1600)</u>					
1601.1.02	DOE-HTGR-85107	A	3/89	US/FRG Accident Condition Fuel Performance Models	
1601.1.19	DOE-HTGR-88331	N/C	9/89	HRB-21 Test Specification	
1601.2.01	DOE-HTGR-87095	A	9/89	Specification for COMEDIE Test BD-1	
1601.2.02	DOE-HTGR-88278	0	3/89	Capsules HFR-B1 PIE Test Procedure	
1601.2.23	DOE-HTGR-88120	1	9/89	Pretest Analyses in Support of AVR Depressurization Program	
1601.2.24	DOE-HTGR-88240	N/C	6/89	Fuel Specification for COMEDIE BD-1 Test	
1601.4.01	DOE-HTGR-88302	--	4/89	Draft Test Specification for MIT DABLE Loop	
1602.3.01	GA/DOE-256-89	--	8/89	Updated Graphite Technology Development Plan, Sections 4.1 and 4.2	
1603.1.01	GA/DOE-257-89	--	8/89	Metals Technology Development Plan, Sections 4.1 and 4.2	
<u>Supporting Technology (WBS 1700)</u>					
1711.5.04	DOE-HTGR-88359	0	9/89	Operating Procedure for SiC Defect Detection	
1713.1.01	DOE-HTGR-88281	0	3/89	Test Specification for Air Flow Test	

TABLE 5-1 (Continued)

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
<u>MHTGR Plant-Level Design and Analysis (WBS 5100)</u>					
5101.1.01	DOE-HTGR-88232	11	9/89	MHTGR Reactor Module General Arrangement	
5102.1.02	DOE-HTGR-88086	1	7/89	Primary Coolant Chemistry Control Requirements	
5102.1.05	DOE-HTGR-86030	*	3/89	Thermal Performance Envelope Requirements (at 25% Feedwater Flow)	*DCP-GA-076 disapproved by PDCO, 5/89
5102.2.03	DOE-HTGR-86029	*	3/89	Plant Design Duty Cycle	*DCP-GA-069 submitted for approval, 4/89
5102.3.06	GA/CE-041-89	--	4/89	Core Seismic Model Update MHTGR-Reference Design	
5102.6.02	DOE-HTGR-88306	0	8/89	Conduction Cooldown 2-D Model Description	
5103.2.01	GA/BNI-041-89	--	5/89	Letter Report - Input to the MHTGR-NI Scheduled Outage Assessment Report	
5104.4.01	GA/BNI-046-89	--	5/89	Letter Report - Input to the MHTGR Onsite Radiation Protection Design Manual	
5106.1.03	DOE-HTGR-88293	0	5/89	Comparison of US/FRG Accident Condition Fuel Failure and Release Models	
5106.1.04	DOE-HTGR-88332	0	7/89	POLO Users Manual	
5106.2.01	DOE-HTGR-87008	1	8/89	Allocations of Investment Protection and Safety Requirements for the Standard MHTGR	

TABLE 5-1 (Continued)

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
<u>System Level Design (WBS 5200)</u>					
5211.1.01	DOE-HTGR-88283	0	3/89	MHTGR Core Flow Analysis for Expected (Nonuniform) Core Gaps	
5211.1.02	DOE-HTGR-88136	1	9/89	Reactor System Arrangement	
5211.1.03	DOE-HTGR-88307	0	9/89	MHTGR Core Temperature Measurement Trade Study	
5211.2.01	GA/DOE-200-89	--	6/89	Requirements for Control Rod Motor Controller	
5211.3.02	DOE-HTGR-88194	1	9/89	Core Support Structure, MHTGR	
5211.3.02	DOE-HTGR-88197	1	9/89	Upper Plenum Shroud, MHTGR	
5211.4.03	DOE-HTGR-88137	1	9/89	Graphite Reactor Internals Arrangement	
5211.4.03	DOE-HTGR-88203	1	9/89	Permanent Side Reflector Assembly	
5211.4.03	DOE-HTGR-88204	1	9/89	Reactor Core Support Structure Assembly	
5211.4.04	DOE-HTGR-88275	0	6/89	Core Support Stabilization Trade Study	
5211.5.02	DOE-HTGR-88140	0	7/89	MHTGR Fuel Product Specification	
5211.5.03	DOE-HTGR-88152	1	9/89	Reactor Core Arrangement	
5211.5.05	DOE-HTGR-88150	1	9/89	Structural Design Criteria for Replaceable Graphite Core Elements	
5211.5.08	DOE-HTGR-88335	0	8/89	Description of AVR Reactor Physics Test Data and the Methods and Techniques of Measurement	
5211.5.20	DOE-HTGR-87063	2	8/89	MHTGR Core Nuclear Design	

TABLE 5-1 (Continued)

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
5211.5.25	DOE-HTGR-88245	1	8/89	Radionuclide Control for MHTGR	
5211.5.26	DOE-HTGR-88358	0	9/89	Radionuclide Methods Validation with FSV Data	
5211.6.01	GA-030759	0	6/89	Auxiliary Service Cask and Transporter Layout	
5211.7.01	DOE-HTGR-88033	0*	6/88	Reactor System Interface Control Document	*No change required
5213.2.01	DOE-HTGR-88035	1	8/89	Heat Transport System Interface Control Document	
5213.2.02	GA/DOE-243-89	--	8/89	Effect of Core and Loop Bypasses on Hot/Cold Streaks	
5213.3.03	TN7361	B	9/89	MHTGR Gas Circulator Status Report	James Howden and Company report
5213.5.08	DOE-HTGR-88347	0	9/89	Hot Duct Removability/Replaceability	
5214.1.01	DOE-HTGR-88036	1	7/89	Shutdown Cooling System Interface Control Document	
5214.1.03	DOE-HTGR-88323	0	7/89	Shutdown Cooling System Assembly Drawing	
5214.1.04	DOE-HTGR-88313	0	7/89	SCS Components Removal/Replacement Study	
5214.3.01	DOE-HTGR-88036/ 88078	--	7/89	Shutdown Cooling System Interface Drawing	
5221.1.01	GA-028875	2	7/89	Fuel Handling Equipment Positioner Study	
5221.1.01	GA-030835	0	7/89	Study - Services Assembly, Plug Actuator	
5221.1.02	DOE-HTGR-88338	0	8/89	Fuel Handling Equipment Support Structure Seismic Analysis	

TABLE 5-1 (Continued)

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
5224.1.02	DOE-HTGR-88343	0	9/89	Flow Diagram-Helium Purification Train (Helium Service System)	
5231.1.02	DOE-HTGR-88044	1	9/89	Reactor Protection System Interface Control Document	
5233.1.01	DOE-HTGR-88166	1	6/89	The Role of the Operator and Manual Con- trol Capability in the MHTGR	
5233.1.05	DOE-HTGR-88046	1	9/89	Investment Protection System Interface Control Document	
5234.2.01	DOE-HTGR-88310	0	7/89	NI Control Algorithm for Enveloping Transients	
5234.2.02	GA-030488	1	4/89	PCDIS (NI Control Only) Interface Drawing	
<u>Design Management and Cost Development (WBS 5900)</u>					
5900.1.01	DOE-HTGR-85028	7	3/89	Management Plan for NI Engineering-MHTGR Preliminary and Final Design	Per contract "Reporting Requirements Checklist"
5900.1.02	QAPD-6300	B	3/89	Quality Assurance Program Document - MHTGR-NIE Program	Per Contract "Reporting Requirements Checklist"
5900.2.03	GA/DOE-020-89	--	7/89	Review of 1988 MHTGR Cost Report (Final Draft)	
<u>Utility/User Requirements and Design Evaluation (WBS 9200)</u>					
9200.3.02	GA-A19476	0	12/88	MHTGR Desalination for Southern California	

TABLE 5-1 (Continued)

SLPP M/S No.	Document No.	Revision	Date	Title	Remarks
<u>Licensing (WBS 9300)</u>					
9300.2.02	GA/DOE-284-89	--	9/89	Letter Report - Proposed Method for MHTGR Technical Specifications	
9300.2.06	DOE-HTGR-85002	3	9/89	Top-Level Regulatory Criteria for the Standard MHTGR	
9300.2.07	GA/DOE-279-89	--	9/89	Letter Report - Licensing Basis Event Selection and Use	
9300.2.08	GA/DOE-280-89	--	9/89	Letter Report - Method for Equipment Safety Classification for the MHTGR	