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MHTGR-NUCLEAR ISLAND ENGINEERING FINAL SUMMARY REPORT FOR THE PERIOD NOVEMBER 30, 1987 THROUGH DECEMBER 1, 1988

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LIST OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
AIA	Automatic Image Analyzer
BNI	Bechtel National, Inc.
B ₄ C	boron carbide
CEA	Commissariat à l'Energie Atomique
C-E	Combustion Engineering, Inc.
DOE	Department of Energy
ECA	Energy Conversion Area
FA	functional analysis
FH&SS	fuel handling and storage system
FOAK	first-of-a-kind
FSV	Fort St. Vrain
GA	General Atomics
HTS	heat transport system
H ³	tritium
IB	instrument block
IFMU	in-core flux monitor unit
IPS	investment protection system
ISI	in-service inspection
JAERI	Japan Atomic Energy Research Institute

KFA	Kernforschungsanlage Jülich GmbH (a German national laboratory)
LEU/Th	low-enriched uranium/thorium
LT-MED	low-temperature horizontal tube multieffect distillation
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MW(e)	megawatt (electric)
MW(t)	megawatt (thermal)
NI	Nuclear Island
NIE	Nuclear Island Engineering
NOAK	nth-of-a-kind
NRC	Nuclear Regulatory Commission
O	oxygen
OPDS	overall plant design specification
ORNL	Oak Ridge National Laboratory
O&M	operations and maintenance
PCDIS	plant control, data and instrumentation system
PDCO	Plant Design Control Office
PF	process flow
PSID	Preliminary Safety Information Document
PSSAR	Preliminary Standard Safety Analysis Report
PyC	pyrolytic carbon
P&I	piping and instrumentation
RCCS	reactor cavity cooling system
RFP	request for proposal
RPS	reactor protection system
SCHE	shutdown cooling heat exchanger
SCS	shutdown cooling system
SCWS	shutdown cooling water system
SER	Safety Evaluation Report

SLPP	Summary Level Program Plan
SSC	structures, systems, and components
SSE	safe shutdown earthquake
UCO	uranium oxycarbide
UKAEA	United Kingdom Atomic Energy Authority
US/FRG	United States/Federal Republic of Germany
WBS	work breakdown structure
WIN	waste information network

1. INTRODUCTION AND SUMMARY

This report summarizes the Modular High-Temperature Gas-Cooled Reactor (MHTGR) - Nuclear Island Engineering (NIE) design and development work performed by General Atomics (GA) for the period November 30, 1987 through December 1, 1988 under the Department of Energy (DOE) Contract DE-AC03-88SF17367. The scope of the report includes work performed by Bechtel National Inc. (BNI), Combustion Engineering Inc. (C-E), and James Howden Company, as major subcontractors to GA.

Prior major accomplishments in the MHTGR program include the submittal of the Preliminary Safety Information Document (PSID) to the U.S. Nuclear Regulatory Commission (NRC) for the purpose of obtaining a licensability statement and the completion of the conceptual design phase. The preliminary design phase of the MHTGR-NIE was initiated at inception of this contract period. The major goals set for the one-year MHTGR-NIE effort were:

1. To obtain NRC's Safety Evaluation Report (SER), to continue interaction on licensing issues, and to prepare Preliminary Standard Safety Analysis Report (PSSAR) outline.
2. To initiate preliminary design addressing key technical issues identified during the conceptual design and assessment efforts.
3. To continue ongoing base technology-related activities.

The NRC staff completed review of the PSID in 1988 and the draft SER was submitted to the Advisory Committee on Reactor Safety (ACRS). An ACRS letter on the SER, sent to the NRC Commissioners in October

1988, generally supported the favorable conclusions of the NRC staff. Several issues remain to be resolved, one issue of a policy nature pertaining to the acceptability of reactor plant designs not having pressure-retaining containments.

The focus in the preliminary design effort was to further the design definition of Type 1 (necessary for licensing or requiring technology program support) systems and structures, with design work on Type 2 (has major interfaces with Type 1 and/or major influence on plant capital cost) systems and structures limited to establishing design interface requirements with Type 1 systems and structures. No work was performed on Type 3 (other systems) systems and structures.

Base and supporting technology activities continued in the areas of graphite and metals characterization and fission product transport behavior. Major accomplishments in fuel development included the demonstration of TRISO coating runs with full production batch size and particle defect detection technique.

The specific scope of work performed under this contract was as prescribed in the Summary Level HTGR Program Plan (SLPP) for FY 1988 (DOE-HTGR-8801, Revision 1).

The following sections of this report summarize the accomplishments for the contract 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 comprehensive list of documents representing contract deliverables.

2. HTGR TECHNOLOGY (WBS 1000)

The objective during this contract period was to continue 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 U.S. Department of Energy/Japan Atomic Energy Research Institute (JAERI) Technology Exchange, and the Siemens-Interatom Technical Review.

The US/FRG subprogram managers meetings [March 8 and 9 at Kernforschungsanlage Jülich GmbH (KFA) and May 9 and 10 at GA] were held to review cooperative programs on fuels, fission products, and graphite pertinent to MHTGR development. The US/FRG Umbrella Agreement Quarterly Progress Report for the Fuel, Fission Products, and Graphite Subprogram, period ending March 30, 1988, and Revision 1 of the US/FRG Safety Subprogram Plan were issued.

U.S. DOE/JAERI technology exchange meetings were held to review the status of Annex 3, Cooperative Program on Graphite Technology, and Annex 5, Cooperative Program on Design Verification and Support Technology, on March 23 through 25 at GA. The status of ongoing cooperative experiments, previous tests, and technical areas for possible additional

cooperation under Annex 3 were discussed. Component design methods, licensing criteria, and support testing were compared for the Japanese and U.S. HTGR programs for potential cooperation.

The scope of the technical review to be performed by Siemens-Interatom on the MHTGR design was defined. The scope included review of core outlet helium mixing, helium purification, operating and maintenance (O&M) assessment, lower core region shielding, radionuclide release and transport, and offsite dose assessment.

Closure on principles for a graphite technology exchange between the U.S. DOE and the United Kingdom Atomic Energy Authority (UKAEA) was confirmed in a meeting held in June 1988.

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)

Data received from HRB-17/18 capsules defining the hydrolysis process in uranium oxycarbide (UCO) and the related radionuclide release were analyzed. The analysis of the transport of cesium and silver in the R2-K13 irradiation tests were completed. The comparison between predicted and measured transport was within a factor of 10 for most data points.

The status of models and data used for predicting radionuclide retention within the confinement building for the MHTGR was reviewed and documented. The report describes the radionuclide transport models, currently used in MHTGR safety analyses, which encompass natural retention phenomena such as gravitational settling, condensation, turbulent deposition, and plateout. The report also describes the experimental database associated with the current models.

The fuel specifications for the COMEDIE test DB-1 were issued. The DB-1 samples contain reference TRISO UCO particles, about 10% of which were "designed-to-fail" particles with only a thin pyrolytic carbon (PyC) coating. Early failure of these designed-to-fail particles provides a well-defined source of fission products. U.S. government authorization was received to export reactor grade H-451 graphite to Commissariat à l'Energie Atomique (CEA) in France for use in fabrication of fuel and reflector components for the COMEDIE test BD-1.

Graphite Base Technology (WBS 1602)

The Graphite Design Handbook was issued. The primary purpose of the handbook is to assemble graphite physical, chemical, and structural properties data for ready use by the designer.

Metals Base Technology (WBS 1603)

The Metals Design Handbook was issued. Data on several alloy materials including nil-ductility temperature shift, creep properties, and long-term behavior at elevated temperature and coolant chemistry exposure were included in the handbook.

Safety Technology (WBS 1604)

A report describing the US/FRG safety criteria and compliance procedures was prepared and issued.

2.3. SUPPORTING TECHNOLOGY (WBS 1700)

Work under the supporting 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)

The in-core flux monitor unit (IFMU) drive concept layout drawing was completed in support of the IFMU test plan. This new concept, which employs a centrally located top-head mechanism that operates to raise and lower a fission detector assembly to traverse the central reflector, eliminates the need to test the IFMU in the Fort St. Vrain (FSV) reactor. The simulated insertion test of the Toshiba fission chamber successfully demonstrated the ability of the unit to traverse bends during insertion into a reactor without damage.

A core flow-induced vibration assessment was performed. Test plans covering core fluctuations and core flow-induced vibration were issued. The results of the assessment showed that if all the core columns were keyed at the top and bottom (metallic plenum elements and core support floor blocks), flow-induced vibrations of any significance are unlikely. Since some columns are not keyed at the top, further confirmation is required that flow-induced vibration cannot occur.

Compacts containing PyC-coated boron carbide (B_4C) were characterized for total boron content. Improved methods were developed to measure the fraction of B_4C kernels exposed by failed coatings.

A status report on the UCO kernel process development was issued. The major accomplishment has been the determination of a set of process parameters resulting in much shorter times for completion of kernel

heat treatment. This allows economy of single-shift operation to be utilized.

Demonstration TRISO coating runs were made with the full production batch size of 5 kg U. Product results confirm the successful development of a process for production of MHTGR-quality TRISO fuel.

Fabrication of the 40-hole fuel compact mold and piston assembly was completed at Micron Machine Company. The assembly was QC-inspected and found to be in compliance with design requirements. The assembly will be used in conjunction with the existing DESMA compact press to manufacture MHTGR-quality fuel compacts.

The Automatic Image Analyzer (AIA) equipment manufactured by CDR Corporation was subjected to an acceptance test and found to meet design requirements. Standard radiograph plates of TRISO-coated particles containing known amounts of missing buffer and fuel dispersion defects were scanned by the AIA and the defective particles were always located by the machine.

The quality assurance plan for MHTGR fuel manufacture was issued. The plan contains an overview of the manufacturing process, quality control techniques, sample size and sampling frequency, and includes the management organization and administrative requirements for quality assurance in MHTGR fuel production.

The development of an improved SiC defect detection technique was documented in a status report. The candidate technique to be calibrated and qualified for MHTGR production quality control will utilize high-pressure mercury to penetrate defective SiC and make such coatings visible by radiograph and automatic image analysis.

Reactor Cavity Cooling System Design Support (WBS 1716)

A report, "Development of Conduction Cooldown Requirements/ Alternatives," was issued for draft review by program participants. A systematic process for developing test requirements for the conduction cooldown of the MHTGR is proposed that focuses on meeting the top-level requirements. The most limiting requirement is in the area of investment protection. The need for meeting all requirements with verified and validated analytical models is addressed. Alternative approaches are delineated that range from a completely analytical assurance of goal compliance, separate effects tests, to a full-scale integrated test of the conduction cooldown thermal performance.

Reactor Protection System Design Support (WBS 1731)

The Helium Mass Flow Measurement Test Requirements Report was issued. The report reviews MHTGR flow requirements, includes considerations relative to flow measurement, and concludes that three basic types of tests are necessary (design verification, qualification, and production acceptance tests). These test requirements are needed in support of the circulator design and testing program.

3. MHTGR DESIGN (WBS 5000)

The objective during this contract period was to initiate 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 Design Data Needs, and review of corresponding technology development plans and participation in design reviews.

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

Development and documentation of plant requirements through functional analysis and allocation of requirements to systems to support the plant design were performed under this task. Plant-level and multi-system 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)

Input was provided in support of the revision of the Overall Plant Design Specification (OPDS) for use in preliminary design of the MHTGR. The Nuclear Island (NI) arrangement drawings, including the reactor module general arrangement drawing, were updated to incorporate design developments and issued. The Functional Analysis (FA) report was revised to make the FA trees consistent with the FA worksheet effectiveness list and to incorporate FA updates by program participants.

A plant parameter trade study was performed at the beginning of preliminary design phase. The results of the study and its recommendation were reviewed by program participants. This review led to a consensus that, although there were several benefits to be derived primarily to the steam generator, the benefits were not sufficiently compelling reasons to change the reference plant parameters for preliminary design.

Plant-Level Analysis (WBS 5102)

Update of the MHTGR Thermal Performance Requirements report was completed, resulting in a design change proposal pertaining to the thermal performance envelope requirements at 100% feedwater flow.

The Plant Design Basis Transient Analysis report was updated to incorporate recently completed analyses of transient events affected by changes in requirements and design configurations.

Development of the MHTGR plant seismic criteria was completed and documented for use in preliminary design. An assessment of seismic interaction of adjacent silos was completed. The study concluded that the seismic response spectra associated with the reactor silo will have to be increased by as much as 20% to account for the presence of nearby silos, and that increasing the spacing between silos to reduce this increase does not appear to be warranted. Development of a conceptual design of the reactor module that may be acceptable at high-seismic sites was completed. The design approach introduces changes in the reactor building and reactor supports to isolate the reactor and steam generator from the horizontal ground motion by means of an isolation slab supported by special seismic bearings. Other components can be easily designed for high seismicity and need not be isolated.

An assessment of the pipe break hazard protection was performed. The report provides guidance for the initial layout of fluid piping systems to limit the risk to both safety and owner's investment, methodology to assess the effects of pipe break, and the design of required protection features.

The crossduct vessel cooling trade study was completed and reported. The study concludes that cooling by an external reactor cavity cooling system (RCCS) bypass cooling loop along the crossduct vessel is not needed to maintain the temperature below 371°C (700°F) during a pressurized conduction cooldown event. An approach to insulation design for the crossduct vessel evolved from the results of the study. The RCCS Decay Heat Removal Analysis report was updated to incorporate additional conduction cooldown analyses and sensitivity studies on reactor vessel temperatures during pressurized conduction cooldown.

Availability/Reliability Assessments (WBS 5103)

A status report on the ongoing investment risk studies was issued. The status report includes identification of investment risk events, studies of selected key events, development of key risk models, and scoping assessment of plant-forced outage. Based on partial results, the MHTGR's ability to meet the investment risk goals is summarized as follows: the design goal of less than 10% forced outage rate appears achievable even when multiple system failures are included, and the likelihood of transients which might cause very long outages appears to be very low. This protection against major loss essentially precludes the type of severe financial loss experienced at Three Mile Island.

A review of the need for automatic purification system pumpdown in the event of primary coolant leaks was completed. Automatic pumpdown is not necessary to meet either investment risk or safety top-level plant requirements, and that operator-initiated action is the better choice.

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

The NI In-Service Inspection (ISI)/Surveillance Assessment report was issued. ISI and surveillance requirements were developed for safety-related NI systems only. Requirements included ISI examination and test schedules, examination categories and methods, acceptance standard, and extent and frequency of examination.

A preliminary study on the approach to inspection of the RCCS was completed. The existing ASME Section XI ISI codes do not appear to be applicable to the RCCS. An approach is proposed using the RCCS failure modes and effects scenarios as a guide for determining inspection requirements necessary for continued operation of the system without diminished reliability.

Control of Radionuclide Release (WBS 5106)

A review of the safety classification of MHTGR structures, systems, and components (SSC) was completed using the previously developed methodology and considering additional design and analysis information available since January 1987. Several SSCs were identified with potential for either reclassification as "safety-related" or for deletion of "safety-related" designation. Additional analyses are required before a complete update to the safety classification of SSCs can be implemented. The review provides a preview of the type of changes in equipment classification that could be anticipated in the future.

A status report on the safety risk assessment for certain low-power and noncore accidents was issued. Assessments were made for control rod withdrawal at source power, loss of forced cooling during refueling, loss of offsite power during refueling, gas waste storage release, and low-temperature adsorber release. Based on point estimates of risk, it was found that the accidents analyzed fell well within the bounds of the MHTGR top-level safety goals. In addition, the low-power and noncore

events considered do not significantly contribute to the overall risk envelope for MHTGR accidents which occur at full power.

The verification and validation plan for radionuclide control methods, which describes the status and work needed to demonstrate that the computer codes (including fuel/fission products methods) utilized to assess MHTGR radionuclide control during off-normal events are verified and validated to meet program requirements for licensing, was issued.

3.2. MHTGR SYSTEM DESIGN (WBS 5200)

MHTGR preliminary design was initiated 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 system and structures. No work was performed on Type 3 (other systems) systems and structures.

Reactor System Design (WBS 5211)

The Reactor System Design Description was reformatted and issued. Major physical interfaces of the reactor system with other systems were established and identified in an Interface Control Document. The reactor system arrangement drawing was updated.

A trade study on the selection of a control rod drive motor was completed. This study, along with the thermal analysis of the neutron control assemblies, resulted in the selection of a relatively small dc torque motor operating through a harmonic drive speed reducer to obtain the required performance.

The Component Design Specification for the metallic reactor internals was issued. Arrangement drawings were completed on the metallic reactor internals, the upper plenum shroud, and metallic core support structure. A stress analysis report for the metallic reactor internals was also completed.

The Component Design Specification for the graphite reactor internals was issued. A status report on the graphite reactor internals design was issued, including assembly drawings of the graphite core

support and permanent side reflector. Analysis substantiating the core support design was documented in a stress report.

The Component Design Specification for the reactor core was issued. A design status report, including the updated core arrangement drawing, was completed documenting the mechanical design and thermal/hydraulic performance on the reactor core. Analyses substantiating the core preliminary design were documented in an updated fuel element stress report. The core nuclear design report was updated to incorporate recently completed preliminary design nuclear analyses.

The analysis of tritium (H^3) production, transport, and environmental release from the MHTGR was completed. Results indicate that most of the H^3 produced in the reactor is retained by the fuel particles and the core structural graphite. The single largest source of H^3 is ternary fission in the fuel particles. The H^3 released from the core and produced by activation is largely removed by the helium purification train.

Plateout distribution analysis for the MHTGR primary circuit components was completed. The analysis accounts for changes in the steam generator parameters and the reference sorption isotherms since the plateout analysis was performed in 1986.

A fuel performance analysis was conducted for the MHTGR preliminary core design in order to predict the core temperature distributions, fuel particle failure, and gaseous and metallic fission product release under normal operating conditions at full power. The results were compared with the previous fuel performance analysis for the conceptual core design. The fuel performance of the MHTGR preliminary core design is slightly better than for the conceptual core design. This is due to the improvements in the core physics design which result in flatter radial power profiles, lower fuel and graphite temperatures, and, consequently, lower fuel particle failure and fission product release.

The MHTGR fuel element corrosion analysis was completed. The primary coolant chemistry specification for impurity levels in the MHTGR specify 2 ppm of H₂O, 2 ppm of CO₂, and 5 ppm of CO. This is equivalent to a total of 11 ppm of O (oxygen) in the coolant. The water-leak rate that would result in these reaction products is equal to about 1.0 lb/h.

The core physics methods verification and validation plan, which describes the status and work needed to demonstrate that the computer codes utilized for core physics analysis are verified and validated to meet program requirements for licensing, was issued. The plan was the subject of an informal design review by the Plant Design Control Office (PDCO). The plan was well received by the review panel.

An informal design review of the nuclear instrumentation was conducted by a PDCO review panel. No major issues were identified; review comments will be considered in the course of the design process.

Vessel System Design (WBS 5212)

The Vessel System Design Description was reformatted and issued. The code inquiry on vessel material elevated temperature application was submitted to the ASME Code committee for review and endorsement. The scoping duty-cycle analysis of the reactor vessel flange under conduction cooldown, with emphasis on prediction of creep in support of the ASME Code inquiry, has been completed; no design concerns are apparent. The pressure stress analysis of the steam generator vessel steam outlet nozzle was completed. Preliminary evaluation of resulting stresses indicates that modifications to the nozzle layout may be necessary; however, the extent of modifications is not expected to affect interfaces.

Heat Transport System Design (WBS 5213)

The Heat Transport System (HTS) Design Description was reformatted and issued. Major interfaces with the HTS were established and identified in an Interface Control Document.

Pressure, flow, and temperature distributions throughout the primary coolant loop for full-load and partial-load conditions and for pressurized and depressurized shutdown were calculated. A significant result is that the expected main loop pressure drop at 100% power design conditions has been reduced by 1.5 psi due to various design changes. The HTS process flow (PF) diagram was revised to reflect the results of these calculations.

Analyses of major moisture ingress events were completed. Amounts of moisture in the primary coolant system and in the reactor core and the time required for removal of the moisture have been evaluated for various plant conditions and event scenarios.

The primary coolant chemistry specification was completed. Three sets of values have been identified: expected, design, and operating limits. The first set of values represents the "long-term" impurity levels to be used to define the reference atmosphere for the MHTGR materials test program. The second set of values is to be used in the design of the components exposed to the primary coolant environment, and the third set is to be used to establish plant-operating limits and procedures.

The Component Design Specification for the main circulator was completed and included in the request for proposal (RFP) for circulator design support. James Howden and company was selected from several vendors that responded to the RFP. Phase I of the design study was completed; Howden's recommended circulator configuration resembled the reference design, the primary features being an induction motor drive

(4500 rpm), a two-stage axial flow compressor impeller, and a magnetic bearing with mechanical catcher system. The circulator design report was updated to reflect the latest design status.

Alternative steam generator concepts using single-material (Alloy 800H) tubing were assessed. The conclusion from the metallurgical evaluation is that the Alloy 800 series material cannot be recommended for the low temperature liquid phase region of the tube circuit due to concern over the corrosion performance since the database required to support the material choice does not appear to exist. The recommendation was to continue to develop the reference design steam generator (2-1/4 Cr - 1 Mo/800H tubing), recognizing that substantial improvements (steam generator design simplification and wide latitude in the choice of plant parameters) are possible if a single material can be used for the tube bundle. It was also recommended to develop and implement a plan to clarify the material corrosion performance of the Alloy 800 series material for specific MHTGR requirements and design configuration. An independent design review of the steam generator reference design was performed by Sulzer Brothers. Sulzer's review concluded that the MHTGR steam generator design is feasible, achievable, within the HTGR experience base, fabricable, and operable. The steam generator design status report was issued, documenting design progress in FY 1988.

The Component Design Specification for the hot-duct was issued. The hot-duct assembly drawing was completed for a modified design of a one-piece longitudinal section duct welded to the core barrel, with bellows that can be removed or replaced separately through the circulator penetration in the steam generator vessel. C-E performed an independent assessment of the hot-duct design and concurred with the current approach of more welded connections and the one-piece design for the main hot-duct longitudinal section.

A design status report and layout drawing for the steam generator vessel upper plenum shrouds were completed. Design changes to the steam generator vessel upper shroud have resulted in a significant reduction in primary circuit pressure loss.

Shutdown Cooling System Design (WBS 5214)

The Shutdown Cooling System (SCS) Design Description was reformatted and issued. Major interfaces with the SCS were established and identified in an Interface Control Document. PDCO performed a formal design review of the SCS. The review focused on design concerns related to the location of the SCS at the bottom of the reactor vessel and to maintenance/replacement issues.

The trade study on the shutdown cooling water system (SCWS) - single versus multiloop SCWS - was completed. The study resulted in a design change to a four-loop SCWS configuration with an air-cooled heat exchanger from the previous single-loop SCWS configuration with a service water-cooled heat exchanger.

A study was performed to evaluate a possible reduction in surface area of the shutdown cooling heat exchanger (SCHE). It was shown that an area 70% of the present design would be marginally acceptable; however, it was recommended that the design not be changed in order to retain margin to accommodate variables in hot streaks and peak conduction cooldown conditions. An assessment of the SCHE design for the hot startup transient condition was completed; a change in the material for the shrouds was recommended. A design status report, including updated arrangement and interface drawings, for the SCHE was issued.

The Component Design Specification for the SCS circulator was issued.

The shutdown cooling system control logic was revised to reflect the four-loop SCWS configuration and a revised instrument block (IB) diagram was issued.

Shutdown Cooling Water System Design (WBS 5215)

The SCWS Design Description was reformatted and issued. Major system interfaces with the SCWS were established and identified in an Interface Control Document.

Revised SCWS PF and piping and instrumentation (P&I) diagrams were issued to reflect the design changes associated with the four-loop SCWS configuration using air-cooled heat exchangers.

Reactor Cavity Cooling System Design (WBS 5216)

The RCCS Design Description was reformatted and issued. Major system interfaces with the RCCS were established and identified in an Interface Control Document. The RCCS trade study was completed, confirming the reference design selection. The RCCS PF and P&I diagrams were revised to incorporate design evolutionary changes. The RCCS failure modes and effects analysis was completed. The status of the computer-based three-dimensional analytical modeling of the RCCS and the computer code for RCCS performance evaluation was documented. PDCO performed a formal design review of the RCCS. No major design issues were identified by the PDCO review panel.

Fuel Handling and Storage System Design (WBS 5221)

The Fuel Handling and Storage System (FH&SS) Design Description was reformatted and issued. Trade studies related to the definition of fuel handling interfaces at the fuel storage wells were completed. The layout drawing of the fuel handling equipment at the upper end of the reactor vessel was revised. Two new layouts that illustrate the conceptual

design of the new fuel handling and storage equipment were completed and issued. Major system interfaces with the FH&SS were established and identified in the Interface Control Document.

Spent Fuel Cooling Water System Design (WBS 5222)

An updated PF diagram of the spent fuel cooling water system was issued.

Helium Services System Design (WBS 5224)

Design requirements for the helium purification train, related to continuous and periodic duties, were established and documented as input to the Helium Services System Design Description.

Reactor Protection System Design (WBS 5231)

With the restructuring of the former plant protection and instrumentation system into the reactor protection system (RPS) and the investment protection system (IPS) for preliminary design, the RPS Design Description was reformatted and issued. Major system interfaces with the RPS were identified in an Interface Control Document.

Results of analysis of eight unprotected transients were documented. The transients analyzed may be broadly grouped into control rod withdrawal events (at 102% power and source level) and circulator speed mismatch events (-15% and maximum speed). The control rod withdrawal events both from source level and full power require protection to limit steam generator temperatures. Circulator underspeed at 100% power is a relatively benign event; however, underspeed from 25% power may necessitate protection of the turbine generator due to the resultant decrease in main steam temperature. Circulator overspeed at 100% power is an

extremely benign event. Overspeed at 25%, however, challenges the reactor vessel relief valve setpoint. These results will form the basis and justification of several of the RPS trips.

Steam and Water Dump System Design (WBS 5232)

Major system interfaces with the Steam and Water Dump system were identified in an Interface Control Document. The PF and P&I diagrams were updated and issued.

Investment Protection System Design (WBS 5233)

With the restructuring of the former plant protection and instrumentation system into the RPS and IPS for preliminary design, the IPS Design Description was reformatted and issued. Major system interfaces with the IPS were identified in an Interface Control Document.

An approach and methodology to address the role of the operator and control philosophy were developed. Design requirements that affect the role of the operator were reviewed to identify basic quantifiable requirements. Design alternatives based on five levels of manual controls that could be provided in the control room were developed. The report issued on the role of the operator addresses: (1) the safety role of the operator, and (2) the operator role in control, operation, and plant investment protection. An evaluation of how the location of manual control capability affects satisfying the top-level design requirements of the MHTGR is included.

Steam generator dump alternatives were evaluated; alternatives considered the reference design (using two valves in two parallel paths) with 18-month test intervals, and an alternative design (using two normally closed valves and one normally open test valve) with 18-month test intervals and only testing the normally closed valves at 18-month test intervals. Eighteen-month test intervals decrease the execute unavailability by a factor of 10 or more compared to 6-month test intervals.

However, not testing the normally open test valve makes the alternative design execute unavailability about the same as the reference design with 6-month test intervals.

SCS isolation/dump alternatives were also evaluated; the alternatives considered were the reference design (using two valves for inlet and outlet isolation and dump with one of two logic) and an alternative design using one valve for each function. The reference design has an unavailability about 10 times better than the alternative design.

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

A schematic for the NI control layout was completed and an update of the NI control logic was issued. An analysis of the load-step transients to determine whether the revised OPDS load-step requirements are compatible with the conceptual control design was completed. With modification, the pressure control deadband proposed in the Energy Conversion Area (ECA) conceptual design is acceptable. The NI control design requirements were developed as input to the plant control, data and instrumentation system (PCDIS) Design Description. NI control interfaces were identified for incorporation in the PCDIS Interface Control Document.

Reactor Complex Design (WBS 5251)

The Reactor Complex Design Description was reformatted and issued. The study report was issued that describes the evaluation and improvements made in the reactor complex design with emphasis on the reactor building. The layout of the upper portion reactor building was modified in which the efficiency of the arrangement was improved by creating a second level for instrumentation and electronic cabinets. Developments related to reactor building systems and components were incorporated along with potential modification to the RCCS panel and duct configuration. Reactor building and equipment location drawings were issued.

A comprehensive layout study on the locations for the various items of fuel handling and reactor service equipment within the reactor service building was completed. The layout shows proposed locations and interfaces for about 20 different components along with layout information for many other components.

Nuclear Island HVAC System Design (WBS 5281)

PF and P&I diagrams for the NI HVAC system were issued. Major system interfaces with the NI HVAC system were established and identified in an Interface Control Document.

Essential AC Electrical System Design (WBS 5291)

The single-line diagram for the essential ac electrical system was issued. The block diagram for the NI electrical power systems and the single-line diagram for the NI nonessential 125 V dc and 120 V ac system, which form the basis for the electrical system design, were also issued.

Essential DC Electrical System Design (WBS 5294)

The single-line diagram for the essential dc electrical system was issued.

Plant Security System Design (WBS 5297)

The preliminary plant security assessment for the MHTGR was completed. The assessment concludes that the risk posed by radiological sabotage can be acceptably limited by meeting the intent, relative to public health and safety and owner's investment, of relevant existing regulatory criteria. Substantial recognition is given to the plant design which inherently provides significant protection against the threat of radiological sabotage or theft.

3.3. DESIGN MANAGEMENT AND COST DEVELOPMENT (WBS 5900)

Management direction, support, and control were performed for the MHTGR-NIE program, including major subcontract work performed by BNI and C-E. Support for the development and maintenance of MHTGR program cost and schedule, including fuel-cycle economics, was provided.

The MHTGR-NIE program management plan and the quality assurance program plans for conducting MHTGR-NIE design activities at GA, BNI, and C-E were issued soon after inception of contract.

First-of-a-kind (FOAK) and Nth-of-a-kind (NOAK) plant cost estimates for the NI were updated. New unit costs for FY 1988 were developed for the low-enriched uranium/thorium (LEU/Th) (once through) MHTGR design and for potential alternative MHTGR cycle designs based on high-enriched uranium (HEU) -235 and/or HEU-233 fuels. These unit costs were estimated utilizing current DOE cost ground rules for advanced nuclear power technologies. These revised unit costs were based on escalating the FY 1986 unit costs. Fresh fuel fabrication costs for FY 1988 actually decreased (by ~4%) from the FY 1986 values due to reduced graphite costs (~20% reduction) over the last two years. The revised unit handling costs, as well as revised DOE estimates of fuel cost assumptions, were used to obtain FOAK, NOAK, and replica plant fuel-cycle cost estimates which, in general, were nearly the same ($\pm 2\%$) as for the 1986 values.

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.

Program Development, Planning, and Control (WBS 9100)

Requirements for an electronic communications system were established based on home office capabilities of MHTGR program participants. The recommended system was the Oak Ridge National Laboratory (ORNL) waste information network (WIN). DOE approval was obtained and the WIN system implemented.

Utility/User Requirement and Design Evaluation (WBS 9200)

A plan was prepared to address incremental utility/user requirements on: increased operational life from 40 to 60 years, increased safe shutdown earthquake (SSE) from 0.3 to 0.5 g, and reactor-turbine configuration.

An evaluation of the technical, economic, and institutional viability of an MHTGR nuclear desalination plant in Southern California for the production of fresh water and electric power was completed. The study consisted of (1) performing an assessment of future needs for new water and power additions; (2) specifying the requirements for a plant which would meet a reasonable fraction of the projected need; (3) selecting a desalination process for use with the MHTGR; (4) developing a conceptual plant design and cost estimate; (5) investigating

the safety and institutional issues associated with the plant; and
(6) developing a project plan.

An attractive dual purpose desalination plant design is based on the reference MHTGR plant. It consists of four 350-MW(t) reactor modules, two 273-MW(e) (gross) steam turbine energy conversion systems, and eight 13.3-million gallons per day low-temperature horizontal-tube multieffect distillation (LT-MED) desalination trains. The net water output of the plant is approximately 100,000 acre feet per year and the net electrical power output is 466 MW(e). The cost of desalinated water is lowest in a dual-purpose plant arrangement in which low-temperature [68° to 74°C (155° to 165°F)] turbine exhaust heat is utilized by the desalination process. This approach has led to a product water cost reduction of nearly a factor of two compared to previous dual-purpose plant designs with higher process temperatures. Relatively few modifications to the reference MHTGR are required for the desalination application. The large-scale LT-MED desalination equipment has been demonstrated and is commercially available.

Licensing (WBS 9300)

Support of NRC's review of the MHTGR PSID and the Probabilistic Risk Assessment continued. The planned series of review meetings with the NRC was concluded with the last two meetings being held in November 1987 and March 1988 to discuss the NRC's "bounding event sequences" (i.e., events beyond the design basis), fuel performance, and vessel integrity during accidents. NRC requests for additional information resulting from these meetings were formalized and documented in PSID Amendment 9, the submittal of which in April 1988 completed supporting documentation for NRC's preparation of the SER on the MHTGR.

NRC staff submitted the draft SER for review by the ACRS subcommittee on Advanced Reactors in June 1988, followed by the ACRS full committee review. Several ACRS meetings pertaining to the review of the SER

were attended in support of NRC staff presentations. An ACRS letter on the SER was sent to the NRC Commissioners in October 1988. The generally favorable conclusions of the NRC staff were largely supported by the ACRS. Several issues remain to be resolved; the principal issue involved in the NRC deliberation is of a policy nature dealing with the acceptability of reactor plant designs not having pressure-retaining containments.

Preparation of the PSSAR was initiated. An outline for the PSSAR was prepared, reviewed, and issued. Issues to be addressed in the application phase of MHTGR licensing and planned continuation of interaction with the NRC were formulated by the Safety and Investment Task Group.

5. CONTRACT DELIVERABLES

The documents listed in Table 5-1 constitute the deliverables for Contract DE-AC03-88SF17367.

TABLE 5-1
LIST OF CONTRACT DELIVERABLES

SLPP No.	Document No.	Revision	Date	Title	Remarks
<u>Technology Transfer (WBS 1400)</u>					
1400.1.01	*	--	3/88	Trip report; fuel fission product and graphite meeting in Netherlands and FRG	*GA letter to DOE, dated 4/12/88
1400.1.02	GA/DOE-121-88	--	5/88	Minutes from US/FRG safety subprogram managers meeting	
1400.1.03	GA/DOE-083-88	--	3/88	Meeting minutes on US/JAERI Annex 5 collaboration	
1400.1.05	GA/DOE-092-88	--	4/88	Updated Interatom work packages	
<u>HTGR Base Technology (WBS 1600)</u>					
1601.2.05.02	DOE-HTGR-88148	0	7/88	Cesium and silver transport in R2K13	
1601.2.12.01	DOE-HTGR-88236	0	11/88	Review of radionuclide retention models and data applicable to the reactor building of the MHTGR	
1601.2.15.07	DOE-HTGR-88240	0	11/88	Fuel specification for COMEDIE test BD-1	
1602.3.02.01	DOE-HTGR-88111	0	9/88	Graphite Design Handbook	
1603.1.02.01	DOE-HTGR-88106	0	7/88	Metals Design Handbook	
1604.2.01.01	DOE-HTGR-87102	1	7/88	Study of U.S. and FRG criteria for use in HTGR safety evaluation	
<u>Supporting Technology (WBS 1700)</u>					
1711.1.09.01	GA/030595	1	11/88	IFMU layout drawings	
1711.1.09.02	DOE-HTGR-88104	0	6/88	Test report - Toshiba fission chamber Model HS-3 simulated insertion test	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
1711.3.09.02	DOE-HTGR-88123	0	9/88	MHTGR core fluctuation test plan	
1711.3.10.01	DOE-HTGR-88124	0	9/88	MHTGR core flow-induced vibration test plan	
1711.3.11.01	GA/DOE-014-89	--	10/88	Letter report on preparation of compacts containing coated B ₄ C particles	
1711.5.02.01	DOE-HTGR-88165	0	9/88	Critical process parameters for UCO kernel production	
1711.5.02.02	GA/DOE-035-89	--	10/88	Letter report on UCO kernel pilot line demonstration	
1711.5.03.02	GA/DOE-046-89	--	11/88	Letter report on status of demonstration of production of TRISO coating process	
1711.5.04.01	DOE-HTGR-88092	0	5/88	Power cylinder array for MHTGR mold assembly	
1711.5.04.03	GA/DOE-016-89	--	10/88	Letter report on compact process development using laboratory press	
1711.5.04.04	GA/DOE-049-89	--	11/88	Letter report on completion 40-hole compact mold piston assembly	
1711.5.05.01	*	--	6/88	Installation of automatic image analyzer (AIA)	*P.O. 095784
1711.5.05.02	DOE-HTGR-88091	0	11/88	DOE MHTGR fuel quality assurance plan	
1711.5.05.03	GA/909802	0	11/88	Capability of the AIA system	
1711.5.05.04	DOE-HTGR-88247	0	11/88	SiC coating defect detection technique development	
1716.1.01	DOE-HTGR-88221	0	11/88	Development of conduction cooldown test requirements/alternatives	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
1731.1.03.01	DOE-HTGR-88237	0	11/88	Helium mass flow measurement test requirements report	
<u>MHTGR Plant-Level Design and Analysis (WBS 5100)</u>					
5101.1.05.02	GA/PDCO-057-88	--	6/88	Comments on overall plant design specification, DOE-HTGR-86004, Rev. 7 draft	
5101.1.06.01	DOE-HTGR-88250	0	9/88	Reactor complex (51) - reactor building - plan below elevation 35 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88251	0	9/88	Reactor complex (51) - reactor building - plan below elevation 12 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88252	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)6 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88253	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)18 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88254	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)31 ft-0 in. and (-)73 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88255	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)100 ft-0 in. and (-)140 ft-0 in.	BNI
5101.1.06.01	DOE-HTGR-88256	0	9/88	Reactor complex (51) - reactor building - Section A-A	BNI
5101.1.06.01	DOE-HTGR-88257	0	9/88	Reactor complex (51) - reactor building - Section B-B	BNI
5101.1.06.01	DOE-HTGR-88258	0	9/88	Reactor complex (51) - reactor building - Section C-C	BNI

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5101.1.06.02	DOE-HTGR-88112	0	6/88	Plant parameter trade study	
5101.1.07.01	DOE-HTGR-86002	5	10/88	Functional analysis report standard MHTGR plant	
5102.1.01	DOE-HTGR-86030	3	9/88	MHTGR module thermal performance requirements	
5102.2.02	DOE-HTGR-86121	2	11/88	MHTGR plant design basis transient analysis	
5102.3.01	DOE-HTGR-88085	0	8/88	Plant level seismic criteria - MHTGR plant	BNI
5102.3.02	DOE-HTGR-88249	0	11/88	Assessment of seismic interaction of adjacent silos	BNI
5102.3.03	DOE-HTGR-88248	0	11/88	Preliminary high seismic design concept for the MHTGR plant	BNI
5102.4.01	DOE-HTGR-88202	0	9/88	Pipe break hazard protection and assessment	BNI
5102.6.01	DOE-HTGR-87082	1	11/88	Reactor cavity cooling system decay heat removal analysis report	
5102.6.02	DOE-HTGR-88139	0	8/88	Trade study on crossduct vessel cooling	
5103.1.01	DOE-HTGR-88235	0	11/88	Status of investment risk studies	
5104.3.01	DOE-HTGR-86026	0	11/88	Nuclear Island in-service inspection/surveillance assessment modular HTGR	C-E
5106.1.01	GA/DOE-030-89	--	11/88	Update of safety classification of MHTGR structures, systems, and components	
5106.1.02 (5211.5.10)	DOE-HTGR-88125	1	11/88	Verification and validation plan for radio-nuclide control methods	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
<u>System Level Design (WBS 5200)</u>					
5211.1.01	DOE-HTGR-88033	0	6/88	Reactor system interface control document	
5211.1.02	DOE-HTGR-88136	0	9/88	Reactor system arrangement drawings	
5211.1.03	DOE-HTGR-86035	2	4/88	Reactor system design description	
5211.3.01	DOE-HTGR-88164	0	10/88	Reactor metallic internal component specification	
5211.3.02	DOE-HTGR-88138	0	9/88	Metallic internals arrangement	
5211.3.03	DOE-HTGR-88191	0	9/88	Metallic reactor internals stress analysis	
5211.4.01	DOE-HTGR-88135	0	10/88	Graphite core support component design specification	
5211.4.02	DOE-HTGR-88229	1	11/88	Preliminary graphite core support structural analysis	
5211.5.01	GA/DOE-044-89	--	11/88	MHTGR reactor core design status letter report	
5211.5.02	DOE-HTGR-88098	0	5/88	Tritium distribution in the MHTGR	
5211.5.03	DOE-HTGR-88099	0	5/88	Plateout distribution analysis	
5211.5.04	DOE-HTGR-88245	0	11/88	Fuel performance and fission product transport analysis for preliminary core design	
5211.5.06	DOE-HTGR-88134	0	10/88	Core components design specification	
5211.5.07	DOE-HTGR-87063	1	11/88	MHTGR core nuclear design	
5211.5.08	DOE-HTGR-88179	0	9/88	MHTGR core physics validation plan	
5211.5.09	DOE-HTGR-88243	0	11/88	MHTGR fuel element corrosion analysis	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5212.1.04	DOE-HTGR-86125	2	2/88	Vessel system design description	C-E
5213.2.01	DOE-HTGR-88035	0	6/88	Heat transport system interface control document	
5213.2.02	GA/028943	2	10/88	MHTGR process flow diagram for the heat transport system	
5213.2.03	DOE-HTGR-86020	4	4/88	Heat transport system design description	
5213.3.01	DOE-HTGR-88031	0	4/88	Main circulator component design specification	
5213.3.02	DOE-HTGR-88241	0	11/88	Main circulator design status	
5213.3.03	*	--	5/88	RFP for circulator design support	*Proposal No. 6300 7052 DC.588.04
5213.3.04	GA/DOE-189-88	--	8/88	MHTGR main circulator vendor selection	
5213.4.04	DPS-88-637	0	11/88	MHTGR steam generator design status report for FY 1988	C-E
5213.4.05	ADD-88-233	--	8/88	MHTGR program foreign trip report. "Summary of April 1988 Technical Exchange Meeting Between Combustion Engineering Inc. and Sulzer Brothers, Ltd. Winter, Switzerland."	C-E
5213.4.06	DOE-HTGR-88233	0	9/88	Assessment of alternate MHTGR steam generator concepts	C-E
5213.5.01	DOE-HTGR-88196	0	11/88	Hot duct assembly	
5213.5.02	DOE-HTGR-88149	0	10/88	Hot duct component design specification	
5213.5.03	GA/DOE-042-89	0	11/88	Status of hot duct assembly design	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5213.5.04	ADD-88-312	--	10/88	Letter report - C-E review of hot duct assembly design	C-E
5213.5.05	SE-7085-800-007	0	11/88	Steam generator vessel upper internals assembly	C-E
5213.5.06	ADD-88-347	--	11/88	FY 1988 MHTGR program steam generator vessel upper internals design assessment report	C-E
5214.1.01	DOE-HTGR-88119	0	6/88	Trade study report shutdown cooling system configuration	
5214.1.02	DOE-HTGR-88036	0	6/88	Shutdown cooling system interface control document	
5214.1.03	DOE-HTGR-86028	3	4/88	Shutdown cooling system design description	
5214.2.01	DOE-HTGR-88127	0	10/88	Shutdown cooling system circulator component design specification	
5214.4.02	E-9417-C087-016	1	9/88	Shutdown cooling heat exchanger subsystem arrangement	C-E
5214.4.02	E-9417-C087-017	1	9/88	Shutdown cooling heat exchanger subsystem interfaces	C-E
5214.4.03	DPS-88-646	0	11/88	MHTGR shutdown cooling heat exchanger design status report for FY 1988	C-E
5215.1.01	DOE-HTGR-88182	1	9/88	Shutdown cooling water system flow diagram	BNI
5215.1.01	DOE-HTGR-88183	0	9/88	Shutdown cooling water system P&I diagram	BNI
5215.1.02	DOE-HTGR-88037	0	11/88	Shutdown cooling water system interface control document	BNI
5215.1.03	DOE-HTGR-87078	2	11/88	Shutdown cooling water system design description	BNI

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5216.1.01	DOE-HTGR-88155	0	8/88	Reactor cavity cooling system configuration trade study	BNI
5216.1.02	DOE-HTGR-88038	0	9/88	Reactor cavity cooling system interface control document	BNI
5216.1.03	BL-8811-0200	--	11/88	Modeling of the reactor cavity cooling system (letter report)	BNI
5216.1.04	DOE-HTGR-88153	1	8/88	Reactor cavity cooling system flow diagram	BNI
5216.1.04	DOE-HTGR-88154	1	8/88	Reactor cavity cooling system P&I diagram	BNI
5216.1.05	DOE-HTGR-87068	3	11/88	Reactor cavity cooling system design description	BNI
5221.1.01	DOE-HTGR-88039	0	6/88	Fuel handling and storage system interface control document	
5221.1.02	DOE-HTGR-86098	2	4/88	Fuel handling and storage system design description	
5221.1.03	GA/028980	1	9/88	Core refueling equipment interface study	
5221.1.03	GA/030425	0	9/88	New fuel drum positioner	
5221.1.03	GA/030426	1	10/88	New fuel storage facility study	
5222.1.01	DOE-HTGR-88160	1	8/88	Spent fuel cooling system flow diagram	BNI
5224.1.01	GA/909789	0	11/88	Helium purification train input to helium service system design description	
5231.1.01	DOE-HTGR-88004	0	3/88	Reactor protection system design description	
5231.1.02	DOE-HTGR-88044	0	6/88	Reactor protection system interface control document	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5232.1.01	DOE-HTGR-88102	A	5/88	Steam and water dump system flow diagram	BNI
5232.1.01	DOE-HTGR-88103	A	5/88	Steam and water dump system P&I diagram	BNI
5232.1.02	DOE-HTGR-88045	0	9/88	Steam and water dump system interface control document	BNI
5233.1.01	DOE-HTGR-88005	0	3/88	Investment protection system design description	
5233.1.02	DOE-HTGR-88046	0	6/88	Investment protection system interface control document	
5233.1.03	DOE-HTGR-88166	0	9/88	Role of operator and manual control capability in the MHTGR	
5234.2.01	GA/029993	1	9/88	Nuclear Island control scheme and interfaces instrumentation block diagram	
5234.2.02	GA/909749	0	9/88	Nuclear Island control input to plant control, data, and instrumentation system design description	
5251.1.01	BL-8808-0136	--	8/88	FY 1988 reactor building trade study report	BNI
5251.1.02	DOE-HTGR-88211	0	9/88	Reactor complex (51) - reactor building - plan below elevation 35 ft-0 in.	BNI
5251.1.02	DOE-HTGR-88212	0	9/88	Reactor complex (51) - reactor building - plan below elevation 12 ft-0 in.	BNI
5251.1.02	DOE-HTGR-88213	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)6 ft-0 in.	BNI
5251.1.02	DOE-HTGR-88214	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)18 ft-0 in.	BNI

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5251.1.02	DOE-HTGR-88215	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)31 ft-0 in. and (-)73 ft-0 in.	BNI
5251.1.02	DOE-HTGR-88216	0	9/88	Reactor complex (51) - reactor building - plan below elevation (-)100 ft-0 in. and (-)140 ft-0 in.	BNI
5251.1.02	DOE-HTGR-88217	0	9/88	Reactor complex (51) - reactor building - Section A-A	BNI
5251.1.02	DOE-HTGR-88218	0	9/88	Reactor complex (51) - reactor building - Section B-B	BNI
5251.1.02	DOE-HTGR-88219	0	9/88	Reactor complex (51) - reactor building - Section C-C	BNI
5251.1.03	DOE-HTGR-88009	0	6/88	Reactor complex building design description modular HTGR plant	BNI
5281.1.01	DOE-HTGR-88168	0	8/88	Reactor building/auxiliary building HVAC flow diagram	BNI
5281.1.01	DOE-HTGR-88169	0	8/88	Reactor service building HVAC flow diagram	BNI
5281.1.01	DOE-HTGR-88170	0	8/88	Personnel service building HVAC flow diagram	BNI
5281.1.01	DOE-HTGR-88171	0	8/88	Helium storage structure/remote shutdown building HVAC flow diagram	BNI
5281.1.01	DOE-HTGR-88176	0	8/88	Radioactive waste management building HVAC flow diagram	BNI
5281.1.01	DOE-HTGR-88172	0	8/88	Reactor building/auxiliary building HVAC P&I diagram	BNI

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5281.1.01	DOE-HTGR-88173	0	8/88	Reactor service building HVAC P&I diagram	BNI
5281.1.01	DOE-HTGR-88174	0	8/88	Personnel service building HVAC P&I diagram	BNI
5281.1.01	DOE-HTGR-88175	0	8/88	Helium storage structure/remote shutdown building HVAC P&I diagram	BNI
5281.1.01	DOE-HTGR-88177	0	8/88	Radioactive waste management building HVAC P&I diagram	BNI
5281.1.02	DOE-HTGR-88071	0	10/88	Nuclear Island HVAC interface control document	BNI
5291.1.01	DOE-HTGR-88158	0	9/88	Single line diagram - essential ac electrical system	BNI
5294.1.01	DOE-HTGR-88159	0	9/88	Single line diagram - essential dc electrical system	BNI
5297.1.01	DOE-HTGR-88244	0	11/88	Preliminary security assessment for standard MHTGR	BNI
<u>Design Management and Cost Development (WBS 5900)</u>					
5900.1.01	HTGR-85028	4	4/88	DOE HTGR programs management plan December 1987 through November 1988	Per contract reporting requirements checklist
5900.1.03	QAPD-6300	A	2/88	Quality assurance program document - MHTGR-NIE program	Per contract reporting requirements checklist

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
5900.1.04	BL-8803-0029	--	3/88	Quality assurance program plan for conceptual design of the standard MHTGR	Per contract reporting requirements checklist (BNI)
5900.1.05	1388-ADD-QAPD-1	1	3/88	MHTGR quality assurance program document	Per contract reporting requirements checklist (C-E)
5900.1.06	DOE-HTGR-88259	0	12/88	MHTGR-NIE final summary report for the period November 30, 1987 through December 1, 1988	Per contract reporting requirements checklist
5900.2.01	*	--	9/88	Interim design cost estimate	*GA/GCRA-024-88
5900.2.02	GA/GCRA-001-89	--	11/88	FY 1988 capital cost estimate - equipment within GA scope of supply	
5900.2.07	BNI/GCRA:89:003	--	12/88	FY 1988 cost estimate update - lead plant	BNI
5900.3.01	DOE-HTGR-86092	1	11/88	FY 1989 fuel cycle economic and unit cost assumptions for MHTGR alternative cycle designs	
<u>Program Development, Planning, and Control (WBS 9100)</u>					
9100.7.01	GA/DOE-079-88	0	3/88	Implementation plan for MHTGR electronic communication	
<u>Utility/User Requirements and Design Evaluation (WBS 9200)</u>					
9200.1.01	PC-000277	0	4/88	Plan to address incremental utility/users requirements (draft)	

TABLE 5-1 (Continued)

SLPP No.	Document No.	Revision	Date	Title	Remarks
9200.3.01	GA-A19476	0	11/88	MHTGR desalination for Southern California (draft)	
<u>Licensing (WBS 9300)</u>					
9300.1.01	HTGR-86024	9	10/88	Preliminary safety information document for the standard MHTGR	PSID Amendment 9
9300.1.01	HTGR-86011	5	6/88	Probabilistic risk assessment for the standard MHTGR	
9300.1.02	BL-8806-0101	--	6/88	PSSAR outline strawman	BNI
9300.1.03	BL-8810-0167	--	10/88	Standard MHTGR draft PSSAR outline	BNI

APPENDIX A
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