December 20, 2004

MEMORANDUM TO: Farouk Eltawila, Director
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

THRU: Jack E. Rosenthal, Chief ***/RA/***
Advanced Reactors and Regulatory Effectiveness Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

FROM: Stuart D. Rubin, Senior Technical Advisor for Advanced Reactors ***/RA/***
Advanced Reactors and Regulatory Effectiveness Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: TRIP REPORT FOR THE INTERNATIONAL TOPICAL MEETING ON HIGH TEMPERATURE REACTOR TECHNOLOGY AND THE IAEA INTERREGIONAL WORKSHOP ON SAFETY DEMONSTRATION AND MARKET POTENTIAL OF HIGH TEMPERATURE GAS COOLED REACTORS BEIJING, CHINA

The purpose of this memorandum is to inform you of subject foreign trip, which involved participation in both the 2nd International Topical Meeting on High Temperature Reactor (HTGR) Technology on September 22-24, 2004 and the International Atomic Energy Agency (IAEA) Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors on September 27-30, 2004. The topical meeting provided a forum for technical experts from international organizations involved in HTGR technology research and development (R&D), safety assessment or commercial deployment to present papers on their plans and recent significant progress with their programs. Presentations included R&D in the areas of HTGR fuel and fuel cycle, nuclear and thermal-fluid analysis, systems analysis, engineering design-analysis, materials and components, safety and licensing, new experimental studies and reactor operations. The purpose of the IAEA workshop was to provide a forum for technical experts from international organizations involved in these areas to the status of current HTGR designs and national R&D programs, core and power conversion system options, safety and licensing issues and potential market applications. The workshop also included observation of a safety demonstration test of the HTR-10 reactor, which is located at the Institute for Nuclear Energy and Technology (INET), and a tour of selected INET HTGR R&D facilities. During the meeting and workshop, informal discussions were held with a number of the participants to explore specific opportunities for the NRC to engage in HTGR bilateral information exchange and safety research cooperation. Overall, participation in the meeting and workshop effectively enabled NRC to maintain and update its cognizance of developments in safety-related aspects of HTGR technology. The attached trip report documents the significant items of interest from the technical presentations and discussions.

The technical papers from the meeting and presentation slides from the workshop can be made available upon request.

The content of this trip report is not likely to be of interest to the Commission.

Attachment: As stated

cc w/att:
A. Thadani, OC
W. Dean, AO
W. Kane, DEDH
M. Virgilio, DEDMRS
E. Merschoff, DEDR
C. Paperiello, RES
J. Craig, RES
A. Szukiewicz, RES
J. Dyer, NRR
M. Culllingford, NRR
J. Dunn Lee, OIP
T. Sherr, NMSS
T. Rothschild, AGCLR

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NRC INTERNATIONAL TRIP REPORT

**Subject** 2nd International Topical Meeting on High Temperature Reactor Technology
International Atomic Energy Agency Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors

**Dates of Travel, Countries and Organizations Visited**

September 22-24, 2004 and September 27-30, 2004
Beijing, China
International Atomic Energy Agency and the Institute for Nuclear Energy and New Energy Technology

**Author, Title, and Agency Affiliation**

Stuart Rubin, Senior Technical Advisor, RES

**Sensitivity** Not applicable

**Background/Purpose**

The purpose of the trip was to participate in the 2nd International Topical Meeting on High Temperature Reactor (HTGR) Technology and the International Atomic Energy Agency (IAEA) Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors (HTGRs). The topical meeting objective was to provide a forum for technical experts from international organizations involved in HTGR technology research and development (R&D), safety assessment or commercial deployment to present papers on their plans and recent significant progress with their programs. The purpose of the workshop was to provided a forum for technical experts from international organizations involved in these areas to discuss the status of current HTGR designs and national R&D programs, core and power conversion system options, safety and licensing issues and potential market applications. As part of the workshop, I observed a safety demonstration test at the Institute for Nuclear Energy and New Energy Technology (INET) HTR-10 reactor. During the meeting and workshop, informal discussions were held with a number of the participants to explore specific opportunities for the NRC to engage in HTGR bilateral information exchange and safety research cooperation.

**Abstract: Summary of Pertinent Points/Issues**

The 2nd International Topical Meeting on High Temperature Reactor Technology and the IAEA Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors involved presentations and discussions on HTGR technology R&D, design-analysis and safety analysis being conducted internationally. The information from the meeting ans workshop are very useful to maintaining and updating the staff’s cognizance of domestic and international developments in safety-related aspects of HTGR technology. During the meeting and workshop, informal discussions were held with a number of the

Attachment

participants to explore specific opportunities for the NRC to engage in HTGR bilateral

information exchange and safety research cooperation of potential future value to an HTGR pre-application review and infrastructure development activities.

## Discussion

On September 22-24, 2004, I participated in the 2nd International Topical Meeting on High Temperature Reactor Technology and, on September 27-30, 2004, I participated in the International Atomic Energy Agency (IAEA) Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors (HTGRs). The international topical meeting was attended by about 140 participants from 12 counties while the interregional workshop was attended by 60 participants from 23 countries. Both took place in Beijing, China.

The objective of the international topical meeting was to provide a forum for international organizations and technical experts involved in HTGR research and development, safety assessment and commercial deployment activities to present papers on their plans and recent significant progress across a broad range of technical and programmatic areas. A total of ninety-six papers were presented on HTGR research and development associated with fuel and fuel cycle; nuclear and thermal-fluid analysis; systems and engineering design and analysis; materials and components; safety and licensing; new experimental studies and test reactor operations. The widening scope and accelerating pace of international HTGR R&D activities in the past two years is reflected by the 2½-fold increase in the number of papers presented at the 2nd International Topical Meeting compared to the 1st International Topical Meeting held in Petten, Netherlands in 2002.

The purpose of the IAEA HTGR workshop was to provide a forum for technical experts from organizations involved in HTGR research, development, design and deployment to present and discuss with representatives from developing countries and with technical peers, current HTGR designs, INPRO, national programs, licensing, market applications, core and power conversion system design options. In all, thirty-three presentations were made at the workshop. The workshop concluded with a visit to the HTR-10 reactor, located at the Institute for Nuclear Energy and Technology (INET). The purpose of the visit was to observe an HTR-10 safety demonstration test and to tour selected INET HTGR R&D facilities.

The conference and the workshop were used to discuss areas and opportunities for NRC to engage in HTGR bilateral information exchange and HTGR cooperative safety research. The attached trip report and related attachments detail the significant items of interest from the presentations and these discussions. A copy of the International Topical Meeting agenda is included as Attachment A and a copy of the international workshop program is included as Attachment B. Copies of the papers for the International Topical Meeting and the presentation materials for the IAEA Workshop can be made available upon request.

### 2nd International Topical Meeting on High Temperature Reactor Technology

Highlights from the International Topical Meeting of potential significant value to HTGR knowledge management activities and of potential future use for HTGR pre-application review and infrastructure development activities, are as follows:

International HTGR Projects and Programs

Mr. D. Matzner (South Africa) presented: (1) the evolution of the Pebble Bed Modular Reactor (PBMR) design, (2) the current status of the PBMR licensing in the Republic of South Africa (RSA) and efforts to secure RSA government funding for PBMR fabrication and construction programs in RSA, (3) activities in preparation for a potential bid in the Next Generation Nuclear Plant (NGNP) - Hydrogen Co-Generation project. Significant recent PBMR design changes were discussed. These included replacement of the dynamic central reflector column (i.e., moving graphite pebbles) with a fixed (and replaceable) central graphite structural column. This change reduces core bypass flow and lowers fuel operating temperatures and also allows for central reactivity control elements to be included for improved shutdown. The core thermal power has been increased from 268 MWt to 302 MWt and finally to 400 Mwt. This will reduce the number of reactor modules and improve overall plant costs. The three vertical shaft turbines in the power conversion system (PCS) have been redesigned into a single horizontal shaft with turbo-compressors coupled to the main power turbine generator. This proven design approach is intended to eliminate turbine blade transient over heating and electromagnetic bearing penetrations, improve equipment maintainability and resolve the turbine rotor failure trajectory safety issues. PBMR project management has been transferred to the RSA Department of Trade and Industry with a new corporate board of directors with anticipated government financing of the project. PBMR licensing milestones in RSA include: agreement on the resolution approach to Key Licensing Issues (12/04); decision and disposition of environmental impact assessment issues 12/04 and submittal of the PBMR safety analysis report to the RSA regulator (following RG 1.70) by 1/06. Site preparation of the PBMR demonstration plant is to start by the second quarter of 2007 and construction to begin by 4/07. PBMR has defined a technology development program to extend PBMR demonstration reference plant capabilities. The development program involves: fuel modeling, advanced fuel design, advanced metallic and ceramic materials, component improvements and testing, advanced modeling and design capability, high-level waste minimization.

Dr. N. Fujimoto (Japan) presented the status of the Japan Atomic Energy Research Institute (JAERI) High Temperature Engineering Test Reactor (HTTR) operations and safety testing, ongoing design and development work for coupled process heat applications and design and development work on Japan's first HTGR power reactor, the GTHTR 300. The HTTR reached full power (30MWt) operations and reactor outlet temperature of 850ºC in December 2001 and 950ºC in April 2004. Power escalation, normal operation and limited system transient tests have been completed in the first phase of testing. Fuel performance monitoring via $^{88}$Kr release to birth ratio at full power indicates a fuel particle failure rate three orders lower than the 1% failure rate limit, which is about $5 \times 10^{-4}$. Reactor performance evaluation and reactor simulation code validation activities are focused on: (a) reactor physics in relation with thermal response and control system, (b) thermal analysis for fuel, reactor internals and high temperature components, (c) fuel performance on fission product release, (d) structural integrity of reactor internals and high temperature components, (e) decay heat and residual heat removal system characteristics. Two phases of safety tests to begin in 2006 are currently being planned. The tests are designed to demonstrate and evaluate the inherent safety characteristics of the HTTR with progressively more challenging transient and accident scenarios. To produce hydrogen from water, the thermo-chemical iodine-sulfur (IS) process was successfully achieved in a small-scale experimental facility. Coupling a large IS process system to the High Temperature Engineering Test Reactor (HTTR) may occur by 2011. Design and development of the direct cycle gas turbine GTHTR300, electric power plant is based on

design and operational experience in conventional gas turbine system and the HTTR. Tests of a major helium power conversion system components have been completed and performance testing of a closed-cycle helium gas turbine system is planned. Commercial deployment of GTHTR300 is expected after 2010.

Mr. Y. Wang (China) discussed plans by the government of China to promote the design, development and widespread deployment of HTGR electric power generating plants in China. The plant design will build on current HTGR technology development activities being pursued at the Institute for Nuclear Energy and New Energy Technology (INET), Tsinghua University. As a first step toward achieving this goal, the state-owned electric utility China Huaneng Group, the China Nuclear Engineering & Construction Corporation and Tsinghua University, have signed a joint agreement to design, construct and operate a demonstration pebble bed power reactor. The objective is to commercialize HTGR modular power plants in China utilizing pebble bed power reactor modules that build on the HTR-10 test reactor design and technology. The reactor design, construction and safety objectives include a core damage frequency (CDF) less than $10^{-6}$/r-yr., no need for offsite emergency planning, and a four-year construction period for reactor modules. Current targets include submitting the project proposal by the end of 2004, approving the feasibility report by 2005, starting construction in 2007 and generating electric power by 2010. The project team is currently preparing the feasibility study and considering various alternative sites for the demonstration plant.

Mr. M. Lecomte (France) discussed work currently under way at Framatome ANP to design and development a versatile high temperature reactor very high temperature reactor (HTGR/VHTR). The Framatome HTGR/VHTR design, referred to as ANTARES (A New Technology with Advanced Reactor for Energy Supply), is an indirect combined cycle HTGR plant which will be capable of electric power generation, process heat generation (for hydrogen production) and co-generation. Framatome is drawing upon its previous HTGR design-analysis experience, including its participation in the design of both the German HTR-Module pebble bed power reactor and collaboration with GA on the HTGR and GT-MHR designs. In mid-2003, Framatome set up a dedicated ANTARES project organization with responsibility to organize and accelerate development of the plant design. Project participants include AREVA workers from Germany, France and the US. AREVA's primary goal is to develop a commercial electric plant but will also compete for the contract to design and build the DOE Next Generation Nuclear Plant (NGNP). The ANTARES reactor will utilize a prismatic graphite block fuel element design similar to the GT-MHR and the coated fuel particle kernels will be either $UO_2$ or UCO. The ANTARES plant will utilize 600 MWt reactor modules, and an intermediate heat exchanger coupled to a closed secondary nitrogen gas cycle. The arrangement is intended to provide flexibility to support any of the aforementioned applications and minimize radiological contamination of secondary side components. The reactor primary system will operate at a lower core inlet temperature than other contemporary modular HTGRs to lower vessel temperatures and heat losses during normal operations and to improve decay heat removal.

Mr. D. Barbier (France) discussed French Atomic Energy Commission (CEA) research and development program to support the HTGR/VHTR and Gas Cooled Fast Reactor (GFR) initiatives in France. This program is organized into five R&D technical areas: design and safety, fuel and fuel cycle, materials and components, hydrogen production, helium turbine and balance of plant. In the design and safety area the focus will be on the analyses of safety features and the development of analytical tools. Analytical tool development is to include models and methods for reactor physics analysis, steady-state and transient thermal-hydraulic

analysis and accident (e.g., graphite oxidation) analyses, fuel system analysis and radionuclide transport analysis. Principles and guidelines for HTGR/VHTR safety assurance and safety analyses are also being developed. For example, a leak-before-break methodology for gas cooled reactors is being developed and will include instrumentation specifications on the required level of detection and will utilize data from planned experiments involving break geometries and associated leakage flow rates. In the fuel and fuel cycle area research and development activities are focused on fuel fabrication, fuel irradiation testing, fuel performance modeling, and advanced coated fuel particle designs associated with the higher burnup and higher temperature conditions of VHTR fuels. In the materials and components area research and development is focused on the qualification of various reactor pressure vessel materials for application to VHTR pressure, fluence and temperature for 60 year lifetimes. Assessments of the suitability of material and physical properties of base material and welds are included in the program.

Dr. D. Hittner (European Commission) discussed the plans and status of the European Commission HTGR Technology Network (HTR-TN) research and development program in the areas of fuel technology, reactor physics, components, materials, safety framework and waste management. About twenty industrial, research and education organizations from of eight countries and the Joint Research Centre (JRC) of the European Commission are involved. HTR-TN R&D activities are intended to establish the technology base for commercial deployment in Europe of HTGRs in the next decade and VHTRs in the longer-term. In the fuels area the limits of German, US and Chinese reference $UO_2$ fuel to perform in VHTR conditions is being explored via irradiation tests in the HFR test facility in Petten and in post-irradiation accident simulation heat-up tests. Fission gas release will be monitored during the HFR irradiations. Fuel models are now being developed which will integrate a Monte-Carlo statistical sampling analysis with the existing deterministic 3-D coated particle fuel code. The added capability will allow an assessment of the statistical failure probability of the thousands of fuel particles in each a fuel element. Fuel particle layer materials properties data for changes due to irradiation is also being developed for use in the fuels code. Research into fuel kernel manufacture, particle coating technology and advanced coating material (i.e., ZrC vs SiC) are also continuing. Fuel particles from an HTR-TN pilot fuel fabrication line will be available for irradiation by early next year. In the reactor physics area, HTGR models and methods are being improved based on HTTR cold criticality experimental data. Based on HTR-10 and HTTR investigations further modeling work is still needed to resolve criticality prediction issues for these plants. In the components area, research into main turbine blade and rotor disc failures has resulted in the conclusion that the containment of turbine missiles within the turbine casings cannot be assured. This led HTR-TN program managers to strongly recommend that PBMR and GT-MHR re-orient the turbine- generators from vertical to horizontal. Although this has been done for the PBMR, it has not been done for the GT-MHR. In the materials area, irradiation testing was initiated in 2004 on 5 graphite grades from 3 manufacturers. The purpose of the irradiations is to develop, by 2006, graphite physical and materials property data that are needed for HTGR thermal and structural design-analysis and are critical to the selection of the optimum grade of graphite for modular HTGR graphite core structures. In the safety framework area, work on a proposed safety framework for HTGRs was presented in September 2004 to national regulatory authorities for countries that might be involved in an

HTGR license application. In the waste area, research is being conducted into graphite decontamination methods with the goal of minimizing the graphite waste volume.

Dr. M. Feltus (U.S.) provided an overview of the status and planned activities for the DOE advanced gas reactor fuel development and qualification program, which is being conducted in support of the design and licensing for the Next Generation Nuclear Plant (NGNP). Based on a comparison of fuel performance margins at high temperature and high burnup, cost, manufacturing risk and schedule risk, a UCO kernel (a mixture of $UO_2$ and UC) rather than a $UO_2$ kernel has been selected for the reference NGNP fuel particle. The goal of the program is to develop fuel particles with properties that result in the requisite very low fuel failure rates at the high temperatures and burnups associated with the NGNP core design. The program is science and technology-based to provide a comprehensive and quantitative understanding of the relationship between the fuel particle manufacturing process parameters and fuel particle manufactured properties and, fuel particle properties and fuel particle performance and failure under NGNP service conditions including postulated accidents. The scope of program development activities includes fuel materials, manufacture and fabrication quality control technology; fuel irradiation, accident simulation and post-irradiation examination; fuel particle materials properties and performance modeling; fuel fission product transport models and methods; and fuel qualification testing required for licensing. A total of 8 (multi-cell capsule) irradiations are planned for the Advanced Test Reactor (HTR) at INTEL. The tests will provide data for developing the fuel manufacturing process, the fuel fission product transport models and the fuel performance model and for qualifying the fuel for licensing. Initial irradiations are expected to begin in FY 2005 with final irradiations ending in FY 2010. The schedule supports and NGNP operations by 2017.

HTGR Technical Sessions

Two days of the three-day HTGR topical meeting involved presentations of technical papers in four parallel (i.e., simultaneous) sessions. In all, there were six technical sessions over the two days. These sessions involved research and development associated with: (1) fuel and fuel cycle, (2) nuclear and thermal-fluid analysis, (3) systems and engineering design and analysis, (4) materials and components, (5) safety and licensing, and (6) new experimental studies and test reactor operations. A total of 96 papers were presented by representatives from thirteen countries and the IAEA. Copies of individual papers are available upon request.
The following provides selected highlights, significant information and observations for each of the sessions.

Fuel and Fuel Cycle

Fuel Performance Analysis Codes

A significant number of countries and organizations presented papers on the development and benchmark of HTGR (TRISO particle) fuel performance codes. These codes are to be used by HTGR designers to optimize the design and specifications of fuel particles for particular HTGR applications and by safety analysts to quantify the failed particle fraction within the core during normal operations and postulated core heat-up accidents. The codes are also to be used to evaluate the fraction of fission product releases during normal operation and accident conditions and to provide input to the fission product transport (source term) codes. R&D work is focused on 2-D finite element models because of the complexity of the particle behavior and the number and the complexity of the particle failure mechanisms. Additionally, the adverse effects of chemical interactions at higher burnups are leading several code developers to add chemical interaction models to the base of thermal-mechanical interaction models. Statistical

models are being added as well to account for the statistical distribution of particle parameters from manufacture and irradiation conditions (e.g., pebble bed cores, irradiation data uncertainties) and to allow improved benchmark (statistical treatment) of particle failure data from irradiation tests. Codes with such modeling have begun to be developed only in the last few years. Presentations indicate that most development and benchmarks so far have involved code-to-code and code-to-data based on historical irradiations for fuel particles whose properties were not fully known or characterized. Current fuel irradiation tests are aimed at addressing these issues to provide better data for benchmarks. Nevertheless, additional research and development is being planned and conducted to improve the best-estimate prediction accuracy of these codes. These activities are focusing on refining the models for the particle layer failure mechanisms, conducting irradiations to develop irradiation-dependant layer materials properties data, conducting post-irradiation examinations to assess failure mechanisms, running fuel irradiations and fuel accident simulation heatup tests to higher levels to obtain more data on failure points and failure margins. Similar work is underway to improve the best estimate models for fission product transport within intact and failed particles. Although these modeling shortcomings will delay fuel design optimization goals they can generally be addressed for license applications with (larger) model uncertainty bands (i.e., defense-in-depth). However, it is evident that the R&D continues to make steady progress in further understanding fuel particle behavior and failure mechanisms and quantifying fuel performance.

<u>Fuel Fabrication Technology</u>

The number of potential HTGR fuel suppliers actively pursuing HTGR fuel particle fabrication technology R&D is increasing. Countries involved in these programs include South Africa, France, China, Japan, USA, Korea and India. The R&D is intended to re-establish and master past German fuel fabrication experience and expertise and in some cases to further develop (optimize) HTGR fuel fabrication methods in order to optimize fuel performance (e.g., properties, dimensions, materials) for specific HTGR design applications. All are needed for licensing the fuel and HTGR plant designs. Japan has successfully demonstrated its fabrication capability in completed fuel irradiation tests and currently on a large scale in the HTTR. China is currently conducting fuel qualification irradiations to allow HTR-10 testing at full power. However, some of the fuel irradiations have resulted in significant fuel failure fractions which have been attributed to fuel irradiation capsule (rather than fuel manufacturing) problems. Although planned, none of the other countries has yet reached the stage of development where they are ready to manufacture TRISO fuel for purposes of qualification irradiation testing of the manufactured fuel. In a few cases (e.g., South Africa, Framatome) these irradiation campaigns will begin in 2006.

Significant unresolved back-end fuel cycle issues for both HTGR pebble and prismatic block fuels forms. These include compatibility with the design specifications of transportation containers and the monitored geologic repository. The large spent fuel volume coupled with the large volume ratio of fuel (i.e., particles, compacts) to graphite (pebble graphite matrix, prismatic graphite blocks) makes separating the fuel from the graphite attractive. However, limits on heat output (11.8kW per standard package) to the repository, the need to separate fuel and graphite offsite and NGNP specifications for onsite storage, is leading most HTGR designers to design their facilities with sufficient onsite storage capacity to accommodate all of the fuel high level waste (HLW) and graphite waste that would be generated over the 60-year life of an HTGR plant. Even so, PBMR, CEA and others are investigating cost-effective

technologies that would allow the fuel particles to be de-consolidated from the graphite matrix. Additional investigations are being conducted in removing the particle coatings from the fuel kernels, which is also a key challenge to HTGR fuel reprocessing.

Nuclear and Thermal-fluid Analysis

A significant number of papers involved HTGR design studies, test predictions and operational analyses that utilized a range of HTGR core neutronic analysis codes, thermal-hydraulic analysis codes and coupled neutronic thermal-hydraulic systems analysis codes.

Several of the HTGR core neutronic studies focused on predicted versus actual (e.g., $K_{eff}$, the number of required fuel pebbles) for the HTR-10 initial criticality and initial power escalation steps up to full power. These tests provided good opportunity to fill the gaps in data for the validation of codes and physics methods used for HTGR core design. VSOP predictions utilized computational errors at lower steps and, with this method, had a computational error of about 2%. The core physics benchmarks with TRIPOLI4 considered two theoretical pebble bed packing geometries and three particle arrangements inside the fuel pebbles. The study concluded that pebble packing assumption is important for Monte Carlo methods, while the system reactivity is insensitive to the assumed fuel particle distribution. Additionally, TRIPOLI4 benchmark studies of pebble core critical experiments that were conducted at the Paul Scherrer Institute, Switzerland, showed improved agreement between experimental and calculated $K_{eff}$ values and reactions rate distribution, if the new recommended graphite reflector effective absorption cross section of 4.47 mbarn is used instead of the former recommended value of 4.09 mbarn, that had been reported in the IAEA technical documents.

Some historical (e.g., AVR) and very limited contemporary (e.g., HTR-10) operational or experimental data is available to develop or validate coupled neutronic thermal-hydraulic HTGR systems analysis codes. However, the studies presented generally involved code-to-code comparisons and sensitivity analyses (e.g., related to CRP-5). Work is continuing to improve code predictive capabilities and to identify and assess the key phenomena and analytical parameters that will provide the basis for further code improvements until more extensive steady-state and transient experimental data are available. Two developmental studies were presented for pebble bed cores with selected system model simplifications. One involved the use of the PARCS core simulator code and 3-D NEM core simulator code coupled to two versions of the THERMIX thermal-hydraulic code to simulate a number of steady-state and transient core conditions based on given cross sections and defined parameters (i.e., standard benchmark problems) to facilitate later code-to-code comparisons. Another developmental study involved KIND, a 2D neutron diffusion code, coupled to THERMIX (Juelich) for fast transient analysis and coupled to the steady-state ZIRKUS (Framatome) code for initial conditions.

Studies were also presented on the use of the Flownex, a two-dimensional computational fluid dynamics (CFD) code for modeling the reactor system coupled to the power conversion system (PCS), with all major PCS components modeled. The verification and validation (QA) program for the PBMR Flownex Nuclear computational program was also presented. PBMR Flownex Nuclear predicts the mass flows, heat transfer and pressures during normal operation and accidents in a PBMR reactor coupled directly to a PCS with a recuperated Brayton cycle. (PBMR Flownex currently utilizes a zero-dimensional point kinetics approach that calculates a single normalized power level for the core based on reactivity feedback equations which take

account of the average fuel and moderator temperatures xenon concentration and control rod positions.) Three dimensional computational fluid dynamics (CFD) studies were presented for partial coolant flow channel blockages in a prismatic block core fuel and assessment of the temperature distribution within the core exit hot gas mixing chamber and connected hot gas duct. The former study showed that the maximum fuel particle temperatures could significantly exceed the 1600° C limit if several coolant flow channels are blocked. The latter study showed the effectiveness of the mixing to the design to mixing chamber. Effective mixing is needed to avoid having the hotter and cooler helium flow core exit streams causing thermal stress limits of the downstream power conversion system metallic components to be exceeded. Finally, a study was presented on a postulated helium gas-turbine “de-blading” accident using the CEA CATHARE system transient thermal-hydraulic code. Turbine de-blading is a concern due to the potential for very high energy loose parts penetrating the turbine casing and impacting the reactor vessel system and for the potential for damage to the graphite lower core support structures due to the rapid increase in pressure drop across the core. The study concluded that turbine de-blading without a reactor trip would result in an abrupt rise in core pressure drop with likely damage and deformation of the graphite core support structure.

Systems and Engineering Design and Analysis

A significant number of the papers presented related to the design and analysis of important power systems and components. These papers focused on the engineering design-analysis, thermodynamic analysis and thermal-hydraulic design-analysis of first-of-a-kind power conversion system (PCS) equipment and components for both direct cycle and indirect cycle HTGR plant designs. Recent design-analysis issues, equipment development risk assessments and nuclear safety considerations have resulted in some HTGR designers changing from a vertical shaft main turbine-generator power conversion system to a traditional horizontal shaft arrangement. The latter layout allows proven turbo-machinery technology to be utilized while reducing the potential for collateral turbine missile damage. Most recently PBMR made this design change. Papers presented included: engineering design-analysis of active components supporting for direct-cycle vertical turbo-machinery; key parameters influencing PCS design and characteristics when directly coupled to the reactor system and plant simulator development.

Materials and Components

Several papers were presented on the subjects of HTGR nuclear graphite, metallic high-temperature, pressure-retaining components and advanced composite materials for in-core components. HTGR nuclear graphite studies included: friction coefficients and wear rates in a high temperature helium environment; back-end waste management strategies and treatments of contaminated and irradiated graphite structures; fracture behavior of nuclear graphite; methods to correlate graphite structures and density with mechanical properties and behavior; the relationship of graphite strength properties to the modulus of elasticity; and a graphite damage mechanics failure model that can be used for predicting graphite structural failure loads in a finite element structural analysis code. Another paper described the planned irradiation of 7 graphites with the objective of developing improved irradiation graphite property data for use in the selection and analysis of HTGR nuclear graphite structures. Candidate reactor pressure vessel (RPV) materials reviews for the PBMR indicate that although SA 508 Grade 3 Class 1 would be suitable for a PBMR with operating temperatures of about 300° C, materials such as Mod 9Cr1Mo will be needed for a very high temperature reactor (VHTR)

PBMR RPV that would be used for hydrogen generation. However, developmental work will be needed to establish manufacturing capabilities of large ring forgings using Mod 9Cr1Mo. Another paper submitted by the European Union (i.e., HTR-M) had similar views. Similarly, although Type 316H stainless steel can be used for the PBMR core barrel, Incoloy 800H would be needed due to the significantly higher in-reactor operating temperatures of a VHT-PBMR. Other presentations describe materials assessments for power conversion system components such as heat exchangers, recuperators and gas-turbine discs, while another presentation described performance testing of HTGR pressure boundary over pressure protection safety valves in a helium environment.

## Safety and Licensing

A number of papers were presented on a broad range of HTGR safety and licensing topics. Papers presented included an analysis of HTR-10 reactivity events such as caused by a control rod withdrawal, pebble densification due to an earthquakes and He circulator RPM increase; A paper on a sensitivity study (with the THERMIX code) of a VHTR fuel and vessel temperature transient response for a depressurized conduction cooldown for alternative design operating inlet/outlet temperatures and variations in decay heat, graphite conductivity, surface emissivity, and reactor cavity cooling system temperature (i.e., CRP-3 benchmark problem) was also presented. The results indicate that although nominal fuel temperatures remain below 1600$^o$ C inclusion of calculation uncertainties could result in exceeding the fuel temperature limit. Code-to- code comparison with the results from other countries that participated in CRP-3 show significant differences in the predicted transient/peak fuel temperatures and the transient/peak vessel temperatures. A paper presented by IAEA indicated that point kinetic models for neutronics analysis of modular HTGRs may not be adequate, especially for reactivity events, due to the large height-to-diameter ratio and the strong spatially heterogeneous thermal feedback effects. Two papers were presented on the application of leak-before-break (LBB) procedure to the timely detection of through-wall cracks in the HTGR cross-connect piping/vessel. The papers rely on the view that, although the cross-connect piping/vessel would be subject to additional types of loads/stresses than reactor pressure boundary piping, they are relatively small, and would not invalidate the application of the methodology, and the detection of LBB, if the He leakage detection scheme is sufficiently sensitive. The assumptions and conclusions of both assessments would require experimental validation. Two papers (KAERI, MIT) were presented on the development and benchmark of computational tools for the analysis of HTGR air ingress events. Benchmarks included data available from HTGR air-ingress experiments conducted with the JAERI inverse U-Tube and Juelich NACOK test facilities. Papers were presented (PBMR) on both the PBMR containment system design basis and the approach to safety classification of SSCs and the approach to defense-in-depth. On the latter topic, there was considerable discussion and differing views among the session participants.

## New Experimental Studies and Test Reactor Operations

A limited number of papers were presented on experimental programs and facilities that have or are being used to develop improved data and knowledge and improved designs for systems, structures and components applicable to HTGR systems design, operation and analysis. A presentation (CEA) described a helium technology program involving test facilities that will be used to develop data on the operational behavior and performance for a broad array of HTGR SSCs in a high temperature, high pressure helium environment.

Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors

Interregional Workshop on Safety Demonstration and Market Potential for High Temperature Gas Cooled Reactors is the first of a series of IAEA projects on innovative reactor concepts. About 60 participants from 24 countries, including 19 developing counties attended the workshop. The purpose of the workshop was to: (1) enable technical experts to provide current HTGR information to technical representatives from developing countries, as well as peers, in the HTGR technical areas of: current design options, INPRO and international activities, safety and licensing issues, process applications and market potential; and to observe a safety demonstration test of the HTR-10 reactor. Highlights from the IAEA Workshop on Safety Demonstration and Market Potential for HTGRs of value to HTGR technology knowledge management activities and of potential future use for HTGR pre-application review and infrastructure development activities are described in the paragraphs below:

IAEA and INET Activities

INET is the focal point in China for the HTGR research and development and design activities. These activities involve HTR-10 operations and safety demonstration testing and data collection activities for INET's safety analysis code development efforts. Completed safety tests include a helium circulator trip (loss of flow) transient without a reactor trip and a single control rod withdrawal (reactivity insertion) transient. More challenging HTR-10 transients and accidents are planned to demonstrate HTR-10 safety characteristics and to obtain data. These activities are aimed at improving analytical capabilities in support of the design of the HTR-PM which will be a pebble bed reactor with an annual core with a steam turbine power cycle. The current plan is to have a central graphite pebble dynamic column although the design could be revised to a fixed graphite central column (i.e., like the PBMR). Although electric power generation by 2010, is the current focus, INET is also conducting R&D to enable the reactor to be used in the future for hydrogen production.

IAEA and National HTGR Programs

The IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) has developed an integrated standard methodology and computer code to assess the potential of future new/innovative reactor designs meeting Generation IV and other objectives (i.e., sustainable, economic, safe and reliable, proliferation resistant, low environmental impact, efficient use of resources, supported by an adequate infrastructure) in different countries and needed R&D (IAEA TECDOC 1362). The method is currently being used on a trial basis to assess new design concepts and their suitability in several countries and regions. IAEA, plans to expand its HTGR safety and development activities to include, respectively, a safety guide for HTGR safety analysis and, beginning in 2006, a coordinated research project (CRP) on potential process heat applications.

China is expanding its HTGR research, development, design and deployment activities beyond the HTR-10 to utilize modular HTGRs to: supplement LWRs for commercial electric power production and to provide process heat for heavy oil recovery, coal gasification and liquefaction and hydrogen production. HTR-10 R&D activities reflect the decision to progress from a steam-turbine cycle BOP design to a direct gas-turbine generator for electric power production. The design of the first Chinese HTGR power reactor module will be a steam-turbine cycle pebble

bed reactor (HTR-PM) with a core design and power output similar to the 2002 vintage PBMR. INET has the significant challenge of leading the HTR-PM design activities with a first module delivery schedule that involves issuance of a power reactor operating license and initial power operation by 2010. It is expected that considerable international cooperative R&D will be required.

France, led by AREVA, is focusing on a designing a prototype prismatic block HTGR/VHTR and conducting related R&D activities to meet Generation IV goals and to progress beyond the EPR to meet their future energy needs. The standard AREVA plant design will utilize an indirect gas cycle which incorporate an intermediate heat exchanger (IHX). Current areas of focus are fuel technology development (including advanced VHTR fuel) and the development and qualification of a suite of gas reactor computer codes for safety and licensing analyses. The preliminary reactor vessel materials choice is 9Cr-1Mo because of its strength and creep resistance, chemical compatibility with impure He and industrial experience with large forgings and welding. Several metal and graphite materials are being considered respectively for the IHX and fuel blocks and core structures. Large loops will be constructed to test components in high temperature and high pressure helium.

Japan, led by JAERI, is the first national HTGR program to achieve a $950^{O}$ core exit operating temperature. This is the temperature required for efficient hydrogen generation. The HTTR R&D and design activities are providing a platform for VHTR R&D and demonstration in the areas of high temperature alloys, core graphite materials and structural design guides, reactor physics and T&H analysis, fuel development and safety analysis. JAERI is currently planning, with international participation, the next phase of HTTR safety demonstration transient tests which will be used to demonstrate the reactors passive safety characteristics and to collect data to benchmark reactor physics and thermal hydraulic and systems transient models and methods. The HTTR is designed with core exit instrumentation (not included on the HTR-10) which is expected to provide good data for code development and benchmarks. JAERI is currently implementing an R&D and design program for the GTHTR300, which will be the first HTGR commercial electric power reactor in Japan and also capable of hydrogen production. It will be based on HTTR technology. The effort will be completed and turned over to the private sector around 2008.

OKBM (Nizhny Novgorod) is the prime Russian contractor of the GT-MHR power conversion unit (PCU). OKBM is conducting R&D and design work for the major components for the power conversion unit. The PCU is an integrated vertical turbine-generator with coolers and a high efficiency heat exchanger (recuperator) housed in a single large pressure vessel which is connected to the reactor vessel via a relatively short cross connect hot gas vessel. Significant progress continues to be made on the design of the PCU components. A significant safety concern was raised during the presentation involving the acceptability of locating a large high energy, high mass, rotating vertical turbine-generator in close proximity to the reactor vessel and separated by a concrete wall. (The PBMR PCU design was the most recent to change from a vertical to a horizontal, radially oriented turbine-generator to address both economic risk issues and safety concerns associated with severe reactor vessel damage that might be caused by a turbine-generator shaft missile).

DOE is leading a US program to develop a VHTR, which is expected to be based on either a prismatic block HTGR or a pebble bed HTGR design. The primary impetus of the reactor design is for both process heat (i.e., hydrogen production) and electric power generation with

objectives of demonstrating its economy, reliability and safety (i.e., licensing) for commercial application. A key to success is to develop, fabricate and qualify fuel which performs at the requisite level of reliability for service conditions (e.g., operating temperature) which significantly exceed that of a modular HTGR designed only for electric power generation. (The US fuels R&D program was presented at the HTR 2004 meeting).

The Republic of South Africa (RSA), led by PBMR Pty. Ltd., after significant evolutionary design changes starting in 1996, has settled a 400 MWt pebble bed reactor unit, with a fixed, replaceable graphite central (annular) column core, a reactor outlet temperature of 900$^{O}$ C and a conventional horizontal single shaft power conversion unit with dry gas seals and oiled bearings. Significant R&D prototype, validation and qualification testing is being conducted to support the design. Prototype testing of first of a kind equipment includes fuel sphere counter, flow and burn-up system tests. Analysis code benchmark test facilities include: construction of a new annular core heat transfer facility, the use of the existing ASTRA and NACOK facilities and continued use of the PBMR, Pty scaled PBMR integral system (PBMR Micro Model) facility. Qualification tests are focused on fuel and graphite irradiation testing and testing of SSCs in a large high temperature, high pressure, helium loop. Construction of the first demonstration PBMR module in the RSA at the Koeberg Nuclear Plant site is scheduled to start in 2007 with synchronization in 2010.

Workshop Technical Sessions

Technical sessions at the workshop included: core design, fuel management and PCU options; core physics; thermal hydraulic and fuel performance analysis methods (national programs and IAEA research coordination); safety and licensing issues; and market potential and market interests. Selected points of interest are described below.

Safety and Licensing

The National Nuclear Regulator (NNR), RSA has identified key technical issues for PBMR licensing in RSA. These were identified in a process similar to a full scope pre-application review and include: assuring shutdown capability; V&V of computer codes; high temperature metallic materials, codes and ISI; containment and source term, control room design, chemical attack, defense-in-depth, classification of SSCs; fuel qualification and performance; reactivity control and measurement; aircraft and external events; graphite behavior; event selection and analysis; PRA; and codes and standards. Response and resolution for the key issues will be addressed within the PBMR safety analysis report and the associated NNR review.

IAEA recently initiated an effort to organize an expert working group to write a TECDOC for HTGR accident analyses. The purpose of the TECDOC is to document: accident analysis methods (and related references) that have been applied to HTGRs; key safety analysis issues to be addressed; and opportunities for improvement. It will include aspects such as: event categorization and acceptance criteria, event selection process, analysis tools and methods, including analysis assumptions, representative analysis results, and recommendations on required technology development (i.e., models and data) and V&V. Also presented and discussed was a draft IAEA TECDOC, issued in 2004, which describes a technology-neutral, risk-informed approach to develop safety requirements for innovative reactors such as modular HTGRs. The approach is similar in a number of respects to the NRC technology-neutral framework that is being developed for new plant licensing. A notable difference is in the area of

defense-in-depth. The IAEA draft TECDOC uses the “levels of defense” concept documented in IAEA INSAG-10.

Also discussed was the safety analysis report (NRC has a copy) for the German licensing review of the HTR-Module. The HTR-Module was a pebble bed power reactor with a steam cycle turbine. The safety analysis approach was deterministic and was based on the application of German regulatory authority LWR safety rules (similar to 10 CFR Part 50) to the extent appropriate and supplemental by HTGR safety and design requirements that were established for the design.

<u>HTGR Market Potential</u>

Based on the number of developed and developing nations represented at the workshop and the remarks of their representatives, it is evident that there is significant world-wide interest in modular HTGRs for a range of commercial and non-commercial applications. These include: electric power generation; low temperature process heat applications (e.g., saltwater desalinization, district heating), high temperature applications (e.g., hydrogen production, coal gasification); surplus plutonium disposition; and actinide waste destruction (i.e., deep burning).

**<u>Pending Action/Planned Next Steps for NRC</u>**

No pending action is planned.

<u>International Atomic Energy Agency (CRP-6 and HTGR Safety Criteria)</u>

Ongoing IAEA activities in the HTGR fuels technology arena are being conducted under Coordinated Research Project 6 (CRP-6). Side discussions with European Commission (EC) representatives participating in CRP-6 included discussion and a request for NRC input to the plans for the HTR accident condition simulation (i.e., heat-up) tests that are to be performed on previously irradiated German archive (Juelich) fuel pebbles. The tests will be performed with the recently refurbished KUFA (“cold finger”) heating apparatus obtained from the Juelich Research Center. The planned tests are part of the HTR-F fuel technology development program and the test results will be provided to countries participating in CRP-6. (Recently NRC suspended its participation in CRP-6 for budgetary reasons). The HTR-F is interested in NRC input to the fuel test plans in response to the HTGR fuel qualification testing methods issues that were raised by the staff in connection with the Exelon PBMR pre-application review. These issues related to the conservatism of the time-temperature heat-up profile historically used for fuel qualification testing. Follow-up planned is to contact Dr. E. Toscano, EC, who is the HTR-F lead for the planned heat-up tests.

Discussions were held with Mr. Yoshi Makihara, IAEA Nuclear Safety Division, on the IAEA activities to organize an international panel of experts to write a safety guideline for the design and operation of modular HTGRs. I discussed a similar ongoing effort by the ANS 28 subcommittee to write ANS Standard 53.1, Nuclear Safety Criteria for Modular Gas-Cooled Reactors and indicated that I would endeavor to arrange for an invitation to participate as an IAEA representative on the ANS standards writing subcommittee. Follow-up actions included arranging for Mr. Makihara to participate in the November meeting of the ANS subcommittee. He is now a member of the subcommittee.

Japan Atomic Energy Research Institute

Discussion were conducted with Dr. Kazuhiko Kunitomi of the Japan Atomic Energy Research Institute (JAERI) on potential bilateral HTGR cooperative research and information exchange associated with HTTR transient testing. During the discussions, Dr. Kunitomi requested input and ideas for the next phase of the HTTR test program. The first phase involved traditional reactor startup and power escalation testing. Following an extended period of power operation, the next phase is intended to involve HTTR system transients which will develop performance data to assess the inherent safety characteristics and passive safety systems of the reactor. The tests are also intended to provide data that is useful to the benchmark of HTGR nuclear, thermal hydraulic and systems transient analysis models. JAERI's interest now is to obtain ideas and suggestions for the tests. Potential costs sharing or other information exchange arrangements would be addressed later. I indicated that the NRC would endeavor to provide suggestions and ideas by the end of the year, in coordination with appropriate NRC management channels and would seek to obtain the input of the NRC's HTGR systems transient analysis contractor (Syd Ball, ORNL) if possible. Dr. Kunitomi was also very interested in obtaining Commission policy papers and Commission decisions on non-LWR containment functional performance requirements, since it would aid JAERI in evaluating and deciding on the containment design requirements for the GTMHR300. The GTMHR300 is a commercial power plant version of the HTTR and is now in the final stages of preliminary design and early stages of systems and component development and testing. Dr. Kunitomi also indicated a strong interest in obtaining informal comments from me and any other technical staff at NRC on the GTMHR300 preliminary safety analysis report. He indicted that he would translate the SAR into English and send it to me by the end of the calendar year. I indicated that I would investigate the possibility of providing personal comments informally early in CY 2005.

Institute of Nuclear and New Energy Technology

An informal meeting was held by representatives from INET, DOE and NRC to discuss opportunities for NRC and DOE research cooperation and information exchange on HTGR technology. Meeting attendees included Dr. Madeline Feltus, DOE, Profs. Yuanhui Xu, INET and Yuliang Sun, INET, myself and staff from INEEL and ORNL. INET/DOE discussions included DOE proposals to: (1) exchange fuel materials (e.g., TRISO coated particles) for characterization by DOE and INET using agreed upon procedures with subsequent exchange and discussion of the characterization results; (2) analyze upcoming HTR-10 transient tests with the ORNL transient analysis code and the INEEL transient analysis code provided sufficient information is provided by INET; (3) provide high temperature incore instrumentation (e.g., temperature and flux measurements) to provide additional data and to demonstrate the capability of the instruments; (4) apply advanced fuel pebble burnup measurement capability developed by North Carolina State University and the University of Cincinnati; (5) analyze HTR-10 pebble flow, core temperature and power distributions using the INEEL PEBBED code for code-to-code benchmark comparison with the INET calculations using the VSOP code; (6) measure HTR-10 in-core temperatures using pebbles with melt-wires; and (7) use the INEEL fuel performance code (PARFUME) to predict fuel performance of the INET qualification fuel currently being irradiated in IVV-2M reactor in Russia. INET expressed sufficient interest in the above areas to support further discussion.

Professor Xu expressed an interest in obtaining any NRC documentation that could be provided on design or safety criteria for HTGRs. I explained that the NRC is developing a technology-neutral, risk-informed safety framework for new plant licensing. The framework is similar to the PBMR concept of controlling risk, and the NRC staff expects to generate technology-specific regulatory guides including those that would be applicable to HTGRs. I indicated that the proposed framework would be submitted to the Commission by December 2004 and that proposed functional performance requirements criteria for containment would be included. I indicated that the SECY paper could be useful for INET. Also mentioned was the documentation produced on the HTGR steam cycle design (e.g., the Preliminary Safety Information Document, the Probabilistic Risk Assessment, and the draft Preliminary Safety Assessment Report).

**<u>Points for Commission Consideration or Items of Interest</u>**

The content of this trip report is not likely to be of interest to the Commission.

Attachments: As stated