NRC FOREIGN TRIP REPORT

Subject:

Research Coordination Meeting of the International Atomic Energy Agency Coordinated Research Project 6, "Advances in HTGR Fuel Technology Development"

Dates of Travel:

May 3-7, 2010

City and Country Visited

Vienna, Austria

Author, Title and Agency Affiliation

Stuart D. Rubin, Senior Technical Advisor for Advanced Reactors
Division of Systems Analysis
Office of Nuclear Regulatory Research

Sensitivity:

Official Use Only

Background and Purpose:

On May 3-7, 2010 I participated in the sixth research coordination meeting (RCM) of the International Atomic Energy Agency (IAEA), Coordinated Research Project 6 (CRP-6), "Advances in HTGR Fuel Technology Development." The meeting purpose was to: (1) discuss the contributions and final results of the research and development (R&D) tasks that had been implemented by the organizations which had participated in the CRP; (2) present and discuss key findings and conclusions that could be drawn from the completed R&D and, (3) conduct a detailed chapter-by-chapter review the draft IAEA TECDOC that had been written to document the R&D that had been conducted.

Information and insights obtained from the meeting will be input to NRC's HTGR knowledge management program; support the development of the NRC NGNP accident analysis evaluation model and HTGR fuel performance model; support effective implementation of the NGNP licensing strategy and the NRC/DOE MOU on NRC participation NGNP Project related to fuel performance and accident source term calculation and; provide insights and information on HTGR coated fuel particle manufacturing methods and QC characterization techniques.

Summary of Pertinent Points and Issues:

Participants in the 6th RCM involved representatives from the US, China, France, Germany, Ukraine, Turkey and IAEA. The traveler participated in the RCMs as a contributing investigator and to provide regulatory and safety perspectives related to assuring HTGR fuel safety performance.

m/3

Overview of National HTGR Fuels R&D Activities

China: The HTGR fuel R&D activities in China are currently focusing on: upgrading existing fuel fabrication process and manufacturing equipment, conducting fuel qualification irradiation testing for ZrC coating and constructing the HTR-PM fuel fabrication plant. Mr. Tang presented the current status of HTR-PM project and the fuel fabrication facility project. The detail design for the HTR-PM is currently underway. First concrete pour for the HTR-PM plant is planned to be implemented in 2010. The upgrade of the existing lab-scale fuel manufacturing facility at INET will be finished by the end of CY 2010, and construction of the HTR-PM Fuel Plant will be started this year. The production capability of the HTR-PM fuel fabrication plant will be 300,000 fuel spheres per year. The design and construction of the fuel fabrication plant will be based on upgrading and scaling-up China's existing laboratory-scale fuel fabrication process and equipment which was developed by INET. It is expected that the prototype fuel fabrication plant will begin to produce fuel spheres for the first core loading of the HTR-PM in 2011. Also presented were the irradiation test results for two Chinese fuel spheres which were irradiated in HFR-EU1 and research results on the fabrication of coated fuel particles with a zirconium carbide (ZrC) layer instead of a silicon carbide (SiC) via the chemical vapor deposition process.

Outside of Scope

Outside of Scope

Overview of IAEA CRP-6 HTGR Fuel R&D Activities

The main objectives of the sixth and final RCM for CRP-6 was to discuss the results of the round-robin exercise on HTGR coated fuel particle characterization techniques, the results of the benchmark calculations for the fuel accident condition simulation heat-up problems and tests and, to review the CRP-6 draft TECDOC report and develop an action list for finalizing the report. The major R&D items for the CRP – the coated fuel particle characterizations and, the benchmark calculations for fuel fission product release for normal operation and accident conditions – were successfully concluded. The TECDOC chapter for the normal operation benchmarks has been drafted. The chapters for the accident condition benchmark problems and coated fuel particle characterizations, using advanced characterization methods, are currently being drafted and under review. The complete fourth draft of the TECDOC was distributed at the meeting to all RCM participants for discussion. All sections of the TECDOC have had significant contributions although some additional inputs are still needed. The draft of a new section on "Regulatory Perspectives on HTGR Fuel Safety and Licensing" was recently added. The chairman noted that IAEA formally concluded CRP-6 in December 2009 although IAEA has extended the completion date for submitting the final TECDOC.

Round-Robin Coated Fuel Particle Characterization Benchmark Exercise

As part of CRP-6, participating members received samples of TRISO coated particles to perform a number of fuel characterizations which are typically required as part of a fuel quality control program during the manufacture of TRISO fuels. The samples for the round-robin benchmark study were produced with zirconia micro-spheres rather than UO_2 or UCO fuel kernels. Four organizations participating in the CRP provided the surrogate fuel particle samples. These were: (1) Babcock and Wilcox Company (B&W); (2) Korea Atomic Energy Research Institute (KAERI); (3) Pebble Bed Modular Reactor, Pty. Ltd (PBMR) and; (4) Oak Ridge National Laboratory (ORNL). The detailed results of the benchmark characterizations were presented earlier at the 5th RCM for CRP-6. The purpose of the benchmark exercise was to compare both the characterization methods and the characterization results for each of the participating members.

Overview of Fuel Fission Product Release Benchmark Exercises and TECDOC Status

In September 2009 a workshop was held for each participating member to present, discuss and compare their results in the subject round robin fuel fission product release benchmark exercises (RREs). During the 6th RCM a summary of the results for all RREs was presented together with the draft of Chapter 4 of the TECDOC, which covered the RREs. A comparative analysis of the results was also presented. Also presented was the suggested final format and content for Chapter 4. Action items were identified to complete the chapter.

Fuel Fission Product Release Benchmark Exercises

In earlier CRP-6 RCMs, the fuel geometry, boundary conditions, steady-state conditions and transient conditions for the accident benchmark exercises were developed and documented. The benchmark exercises involves three parts: (1) sensitivity studies of fission product releases from (a) a bare spherical fuel kernel, (b) a spherical fuel kernel with a buffer layer and an IPyC coating layer and, (c) a complete TRISO fuel particle with all coating layers; (2) post-test calculation of historical fuel heat-up experiments and; (3) a pre-test predictions of planned fuel heat-up tests that were to be conducted during CRP-6 , and (4) a predictive study for a future HTR-PM fuel element. A total of 15 benchmark cases were involved in the sensitivity study. Additionally, 9 heating experiment cases were involved. The required outputs for the transient benchmarks involved calculating the release of Cs-137, Sr-90, Ag-110m, and I-131 from the coated particles and the fuel sphere, respectively. Calculated cesium release was an area of focus.

Analysis of Accident Condition Fuel Fission Product Release Benchmark Results

A comparison of the code-to-code predictions for the fission product releases from a single bare kernel during a heat-up accident generally showed good agreement. It was suggested that the predicted releases could be further improved if the Booth formula were used for accident condition (and normal operating condition) fission product releases from the kernel.

A comparison of the code-to-code predictions for the fission product releases from a single intact TRISO coated fuel particle during a heat-up accident generally showed agreement during only parts of the heat-up transient. For the German (FRESCO) code, the Cs diffusion coefficient in SiC was observed to be different from the recommended diffusion coefficient. It was observed that several predictions provided by the GA code were generally higher than the other code predictions while the AREVA code predictions were frequently lower than the others.

A comparison of the code predictions with actual post-irradiation heat-up (i.e., code validation) typically showed an over-prediction of the measured metallic releases for all of the codes. For strontium, the measured releases were over-predicted by all of the codes by several orders of magnitude. It was concluded that the over prediction by all codes was due to the diffusion coefficients for Sr through SiC layer being overly excessively conservative. The diffusion coefficients used for the benchmarks were taken from IAEA TECDOC 978. As a result, it was recommended that a cooperative research program or working group be initiated to establish a more realistic strontium transport coefficient. The cesium releases calculated by the codes also were higher than the measured releases but to a lesser degree than Sr. It was also observed that during the heating tests the measured silver releases remained constant and relatively low, which were not expected. It was proposed that the heating tests be further investigated and future tests planned as part of a future cooperative research program.

AVR Melt-Wire Re-Analysis

A detailed re-analysis was presented of the German AVR core operating temperatures based on the data collected from the "melt wire" pebbles. The melt-wire pebbles were passed through AVR core in 1986. The re-analysis of the AVR melt-wire experiments resulted in an "inferred" fuel pebble temperature frequency distribution (i.e., fuel pebble temperature census) for the AVR core at a time when it was operated at a measured average core outlet temperature of 950° C. During the AVT experiment, graphite spheres containing thin wires (rather than coated fuel particles) with a range of metallic composition (and therefore a range of melting points) were dropped onto the top of the pebble core at the center drop point and the outer drop points. The re-analysis concluded that the spherical fuel elements which were passed through the center region of the AVR core must have had a distribution with a mean pebble surface temperature of 1100°C and a standard deviation of 66°C. The spherical fuel elements passing through the outer region of the AVR core must have had a distribution with a mean pebble surface temperature of 1220°C and a standard deviation of 100°C. These temperatures are associated with fuel pebbles at the very top of the active core, where the highest (exit) temperatures occurred during AVR operation. The would indicate that that some of the fuel pebbles at the top of the core likely operated at temperatures as high as 1450°C which was well-above the maximum fuel operating temperatures that were reported in the AVR licensing safety analysis report.

Further, based on core bypass flow sensitivity studies, which had been conducted by PBMR, it was concluded that the much higher than expected fuel temperatures could best be explained by much higher than predicted coolant bypass flows for the AVR core. Analyses conducted by PBMR show that the much higher than expected maximum fuel temperatures at the top of the AVR core must have been due to an additional 10% "non-engineered" (unknown) core bypass flow caused by the widening of the narrow gaps between each of the outer graphite reflector blocks. This widening, which occurred over the plant operating life, was caused by graphite shrinkage/distortion which increased with increasing neutron dose. Thus about 10% "non-engineered" (unknown) core bypass flow developed over time which added to the 9%

"engineered" (known) coolant bypass flow. This bypass flow is designed to pass through the control rod insertion holes in the outer graphite blocks in order to cool the control rods.

Potential Follow-on IAEA-Organized CRPs

At the end of the RCM, two recommendations were discussed for follow-on IAEA-organized CRPs.

The first recommendation involved IAEA organizing an effort among HTGR fuels experts to re-review and possibly reanalyze the existing historical data for metallic fission product transport and release from accident condition testing experiments and to develop recommendations for improved techniques and methods for deriving the diffusion coefficients. This would involve international experts in the areas of fuel post-irradiation testing methods, fuel post-irradiation examination methods and analytical methods for deriving fuel fission product transport coefficients from the data collected from such methods.

The second recommendation involved IAEA organizing a follow-on CRP focused on modeling and code benchmarks of core-wide fuel fission product release during HTGR normal power operations and HTGR accident conditions. This would involve small interdisciplinary (e.g., nuclear, thermal fluids, fuels) teams of experts from national organizations which are currently involved in the development, verification and validation of HTGR accident analysis methods.

Both recommendations were to be documented for review and consideration by the IAEA

Points for Commission Consideration or Items of Interest

None.

Attachments:

The presentations for the CRP-6 RCM are to be placed in ADAMS under Project Number 748, Next Generation Nuclear Plant.