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ABSTRACT

Development of Safety Analysis Codes and Experimental Validation for a Very High Temperature Gas-Cooled Reactor

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The proposed research focuses on development of new **Advanced Computational Methods** for safety analysis codes for Very High Temperature Gas-Cooled reactors (VHTGR), and **numerical & experimental validation of these computer codes.** The research proposes to improve two wellrespected light water reactor transient response codes (RELAP5/ATHENA and MELCOR) in the modeling of molecular diffusion and chemical equilibrium, and to validate these codes against VHTGR accident data, i.e., air ingress and others from literature. The VHTGR is intrinsically safe, has proliferation resistant fuel cycle, and many advantages relative to light water reactors (LWRs). This study consists of six tasks: (a) development of computational methods for VHTGR, (b) theoretical modification of aforementioned computer codes for molecular diffusion (RELAP5/ATHENA) and modeling CO and CO_2 equilibrium (MELCOR), (c) development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power distributions and decay heat deposition rates, (d) reactor cavity cooling system experiment, (e) graphite oxidation experiment, and (f) validation of these codes.

The VHTGRs are those concepts that have average coolant temperatures above 900°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat application in addition to power generation. While all the High Temperature Gas Cooled Reactor (HTGR) concepts have sufficiently high temperature to support process heat applications, such as coal gasification, thermochemical **hydrogen production**, desalination or cogenerative processes, the VHTGR's higher temperatures allow broader applications. However, due to the high temperature operation, this reactor concept can be detrimental if accidents occur by a loss-of-coolant accident (LOCA) or a pipe breaks due to seismic activities and others. Following the loss of coolant through the break and coolant depressurization, air will enter the core through the break by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structure and fuel. The oxidation will accelerate heatup of the reactor core and the release of toxic gasses (CO and CO_2) and fission products. Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. As of today, the world does not have reliable numerical tools to analyze this event. The INEEL has investigated this event for the past three years for the HTGR. The new code development, improvement of these codes, and experimental validation are imperative to narrow the gap between predicted knowledges on this type of accident and the real phenomena occurring in the reactor.

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This project promotes the development of advanced numerical schemes and technologies that will enhance the safety and economics of a range of reactor designs. Innovative concepts, methodology, and data that will be obtained from this study include:

- development of a benchmark safety code.
- incorporation of diffusion model into RELAP5/ATHENA code.
- incorporation of chemical equilibrium model into MELCOR code
- development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power distributions and decay heat deposition rates
- code validation using data to be collected in this study and additional data from German NACOK or Chinese HTR-10 experiments.

This project will validate computational methods using new experimental data to be collected in 2003 and data from Germany or China. The most significant issue for the U.S. Nuclear Regulatory Commission (NRC) licensing of VHTGRs is the V&V of computer codes used in the neutronic and safety analysis of plant performance. At present, such capability is extremely limited. Validation of the wellknown computer codes will facilitate the licensing process.

Project tasks have been defined to take advantage of key capabilities of this international team. Our highly esteemed and experienced experts in Korea on the high temperature gas cooled reactor system bring code development, scaling test and relevant tests for this project that enable production of quality work. In support of DOE programs and of the nuclear power industry, the INEEL has long been an international leader in treating transient reactor thermal hydraulic behavior, both experimentally and numerically. Based on its large-scale experiments at the Water Reactor Research Test Facility, INEEL has developed the world's leading code (RELAP5/ATHENA) for transient analyses of hypothesized reactor accident scenarios. That same experimental expertise is employed for this project. In addition, the INEEL's long history of collaboration with international and academic research organizations ensures a strong research team as a leading organization.