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International Nuclear Energy Research Initiative

2003 Annual Report

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Foreword

The International Nuclear Energy Research Initiative (I-NERI) was established by the U.S. Department of Energy (DOE) in fiscal year (FY) 2001 as a mechanism for conducting collaborative R&D with international partners in advanced nuclear energy system development. I-NERI was created in response to recommendations of the Presidents' Committee of Advisors on Science and Technology (PCAST) in the Committee's 1999 report entitled Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation.

The I-NERI program focuses on creating collaborative research with participating countries which currently include the Republic of Korea, France, the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), the European Union, Canada, and Brazil. Participating in collaborative research allows the Department to leverage its resources and expand its knowledge in nuclear science and engineering and make valuable contacts with other countries' researchers. In FY 2003, under the U.S./Republic of Korea agreement, five research projects were awarded involving national laboratories, industry, and universities. U.S. universities were collaborators in four of these projects. The overall U.S. participation in the I-NERI program consists of five national laboratories, eleven universities, and two industry contributors. All I-NERI projects currently funded directly support the Generation IV Nuclear Energy Systems Initiative (Generation IV), the Advanced Fuel Cycle Initiative (AFCI), or the Nuclear Hydrogen Initiative (NHI) programs.

FY 2003 marks the second full year of the I-NERI program. The program has achieved the following goals:

- Completed FY 2002 annual project performance reviews under both U.S./France and U.S./Republic of Korea collaborations and approved continuation of the projects.
- Completed awards for the five proposals selected in the FY 2002 U.S./Republic of Korea competitive procurement.
- Completed new international cooperative agreements with the European Commission, Canada, and Brazil.
- Leveraged U.S. contributions with international contributions of approximately $11.2M for Republic of Korea and $7M for France as a result of collaborating with international counterparts.

This annual report includes FY 2003 programmatic accomplishments and summarizes research progress based on information submitted by the principal investigators for I-NERI projects initiated in FY 2001, FY 2002, and FY 2003.

William D. Magwood IV, Director
Office of Nuclear Energy, Science and Technology
U.S. Department of Energy

Contents

1.0 Introduction

The International Nuclear Energy Research Initiative (I-NERI) supports the *National Energy Policy* by conducting research to advance the state of nuclear science and technology in the United States. I-NERI sponsors innovative scientific and engineering research and development (R&D) in cooperation with participating countries to address the key issues affecting the future of nuclear energy and its global deployment by improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of future nuclear energy systems.

The *International Nuclear Energy Research Initiative 2003 Annual Report* serves to inform interested parties of progress and future planning, as well as research progress made in individual I-NERI projects in the second full year of the I-NERI program.

Section 2 of this report provides background on the motivation and events that led to the creation and implementation of I-NERI, a discussion of the goals and objectives of the program, and an overview of the key players in the current I-NERI collaborations.

Section 3 provides an overview of the current established I-NERI collaborations, a summary of the workscopes of current collaborative projects, a summary of research project awards through the end of fiscal year (FY) 2003, highlights of FY 2003 accomplishments and planned FY 2004 activities.

Section 4 provides a summary of programmatic accomplishments to date including the number of new and existing projects in the current collaborations, an overview of the current funding profiles of each of the collaborations, the status of new I-NERI collaborations established in FY 2003, and new collaborations anticipated in FY 2004.

Sections 5, 6, and 7 provide, for the U.S./France, U.S./Republic of Korea, and U.S./Organization for Economic Cooperation and Development (OECD) collaborations, respectively, an overview of the R&D workscopes of current projects.

Appendixes A, B, and C provide for the respective collaborations, an index of projects, and summaries of technical accomplishments in FY 2003 for each current I-NERI project.

2.0 Background

In January 1997, the President of the United States requested his Committee of Advisors on Science and Technology (PCAST) to review the current national energy R&D portfolio and provide a strategy to ensure that the United States has a program to address the Nation's energy and environmental needs for the next century. In its November 1997 report responding to this request, the PCAST Energy R&D Panel determined that ensuring a viable nuclear energy option to help meet our future energy needs is important, and recommended that a properly focused R&D effort should be implemented by the U.S. Department of Energy (DOE) to address the principal obstacles to achieving this option, including improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of nuclear energy systems.

In 1999, in response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI) to help overcome the principal technical and scientific issues affecting the future use of nuclear energy in the United States. Information on the NERI program, including an overview of the NERI program, accomplishments, a summary of progress, and planned activities for projects initiated in FY 2000, FY 2001, and FY 2002 are provided in the *Nuclear Energy Research Initiative 2003 Annual Report*.

Recognizing the importance of a focused program of international cooperation, PCAST issued a June 1999 report entitled *Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation*, which highlights the need for an international component of the NERI program to promote "bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management, and proliferation resistance of nuclear fission energy systems." The report further states that "The costs of exploring new technological approaches that might deal effectively with the multiple challenges posed by conventional nuclear power are too great for the United States or any other single country to bear, so that a pooling of international resources is needed..."

The I-NERI component of NERI was established in FY 2001 in response to the PCAST recommendations. The I-NERI activity is enhancing DOE's ability to leverage its limited research funding with nuclear technology research funding from other countries.

To date, three I-NERI collaborative agreements have been fully implemented: the first between DOE and the Commisariat a l'Énergie Atomique (CEA) of France; the second between DOE and the Republic of Korea Ministry of Science and Technology (MOST); and the third with the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA). The primary U.S. client for the OECD/NEA program is the Nuclear Regulatory Commission (NRC), with DOE as a contributing partner. Since program inception, five projects with France, eleven with the Republic of Korea, and one with NEA have been initiated. New cooperative agreements were signed in FY 2003 with Brazil, Canada, and the European Union. Actions to implement these new collaborations will be taken during FY 2004. Discussions are ongoing with Japan, the Republic of South Africa, and the United Kingdom with the intent that at least two additional I-NERI collaborations will be established during FY 2004. Lists of the currently funded I-NERI projects are provided in Appendixes A, B, and C. Abstracts of funded I-NERI projects are maintained on the I-NERI website, http://www.nuclear.gov.

3.0 I-NERI Program Description

Mission

I-NERI sponsors innovative scientific and engineering R&D, in cooperation with participating countries, to address the key issues affecting the future use of nuclear energy and its global deployment by improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of future nuclear energy systems.

Goals and Objectives

In accomplishing its assigned mission, the following overall objectives have been established for the I-NERI program:

- ♦ Develop advanced concepts and scientific breakthroughs in nuclear energy and reactor technology to address and overcome the principal technical and scientific obstacles to the expanded use of nuclear energy worldwide.
- ♦ Promote collaboration with international agencies and research organizations to improve development of nuclear energy.
- ♦ Promote and maintain a nuclear science and engineering infrastructure to meet future technical challenges.

Over the past three years, the Office of Nuclear Energy, Science and Technology (DOE-NE) has coordinated wide-ranging discussions among governments, industry, and the research community worldwide on the development of next-generation nuclear energy systems known as the Generation IV Nuclear Energy Systems Initiative (Generation IV). During FY 2004, DOE plans to restructure I-NERI to become a key collaboration mechanism for conducting research with international partners centered on R&D workscopes closely linked to Generation IV and the other major R&D programs of DOE-NE's Office of Advanced Nuclear Energy Research, Nuclear Hydrogen Initiative, and the Advanced Fuel Cycle Initiative.

Scope of Work—FY 2002-2003

In their current form, the workscopes of projects selected to be supported by I-NERI are broadly based in the following general areas:

- ♦ **Next-generation (i.e., Generation IV) nuclear energy and fuel cycle technology**. The focus of this R&D area is on assessing key enabling technologies of Generation IV reactor concepts. Approaches include physical and numerical modeling of structural, thermal hydraulic, chemical, and nucleonic behavior under the demanding conditions presented by advanced reactor concepts, including severe accident scenarios.
- ♦ **Next-generation nuclear plant designs with higher efficiency, lower cost, and improved safety and proliferation resistance**. This topic assesses the noted performance tradeoffs of selected Generation IV concepts. Current I-NERI collaborations focus on modeling and assessing of advanced power cycles for high-temperature light water, supercritical water, gas-cooled, and liquid-metal-cooled reactors.
- ♦ **Innovative nuclear plant design, manufacturing, construction, operation, maintenance, and decommissioning technologies**. Key elements in this R&D topic include advanced sensors, instrumentation, algorithms, and controls for optimized operations.
- ♦ **Advanced nuclear fuels and materials**. Current work includes modeling particle fuels and evaluating advanced zirconium (Zr) alloys and nano-composited steels for high burnup applications and metals for use in reduction of fuels by molten salt extraction.

- **Fundamental nuclear science**. Key topics in this area include experiments to improve data for neutronic predictions, experimental evaluation of sulfur-iodine (S-I) water splitting chemistry for H_2 gas production in a high-temperature reactor, and core melt/water interactions under severe accident conditions.

The specific workscope of each I-NERI collaboration is established by agreement between DOE and the respective agency of the collaborating international country. Figure 1 provides an overview of the I-NERI program as it has functioned through FY 2003.

Program Organization and Control

Bilateral I-NERI agreements are normally established under existing or new "umbrella" agreements between the collaborating countries. The U.S. element of I-NERI is managed by DOE-NE, who receives guidance from the Nuclear Energy Research Advisory Committee (NERAC). A counterpart agency of the collaborating country manages their participation.

Currently, each I-NERI collaboration is directed by a Bilateral I-NERI Steering Committee (BINERIC) made up of representatives from the United States and the collaborating country. The BINERIC identifies specific research areas for mutually beneficial collaboration and makes decisions on other bilateral cooperation issues such as required agreements, eligibility for participation, project selection processes, joint funding structure, and contractual vehicles. Each BINERIC operates according to guidelines approved by the collaborating countries. Executive Agents (EA), one from each country, administer the I-NERI program under the guidance of the BINERIC. The structure for control and administration of the I-NERI program is illustrated in Figure 2.

Funding

The I-NERI program provides an effective means for international collaboration on a leveraged, cost-shared *quid pro quo* basis. Each country in an I-NERI collaboration provides funding for their respective project participants. Actual cost-share amounts are determined for each jointly selected project. The program has a goal to achieve approximately a 50-50 matching contribution from each partner country. Funding provided by the United States can be spent only by U.S. participants. I-NERI projects are typically for a duration of three years and are funded annually through grants and cooperative agreements.

4.0 I-NERI Program Accomplishments

The I-NERI program effectively began in the second quarter of FY 2001, with an initial focus on developing international collaborations, program planning, and project procurements. Awards for the first set of I-NERI projects were made on the French collaboration at the end of FY 2001. I-NERI program progress for FY 2001, FY 2002, and FY 2003 are reported briefly here.

Programmatic Accomplishments

The primary programmatic accomplishments in FY 2001-2003, and planned accomplishments for FY 2004, are briefly described as follows:

FY 2001 Programmatic Accomplishments

- DOE signed collaborative I-NERI agreements with the Republic of Korea (May) and France (July).
- The U.S./France collaboration started with seven proposals resulting in the award of three projects in September 2001 and another in January 2002.
- The U.S./Republic of Korea program conducted a competitive procurement resulting in 21 proposals from which six projects were selected for FY 2002 awards.

FY 2002 Programmatic Accomplishments

- DOE and the Republic of Korea MOST completed awards for six U.S./Republic of Korea projects involving 13 U.S. and nine Republic of Korea participants from 15 universities, four national laboratories, and four industry partners.
- Added collaboration with the OECD/NEA, under which one new project was awarded with funding provided by the NRC, DOE, and the Electric Power Research Institute (EPRI).
- Added one new project in the U.S./French collaboration on nuclear-based, thermo-chemical production of hydrogen, bringing total funded U.S./French projects to five.
- Conducted competitive procurement in the U.S./Republic of Korea collaboration resulting in 22 proposals from which five projects were selected for FY 2003 awards.

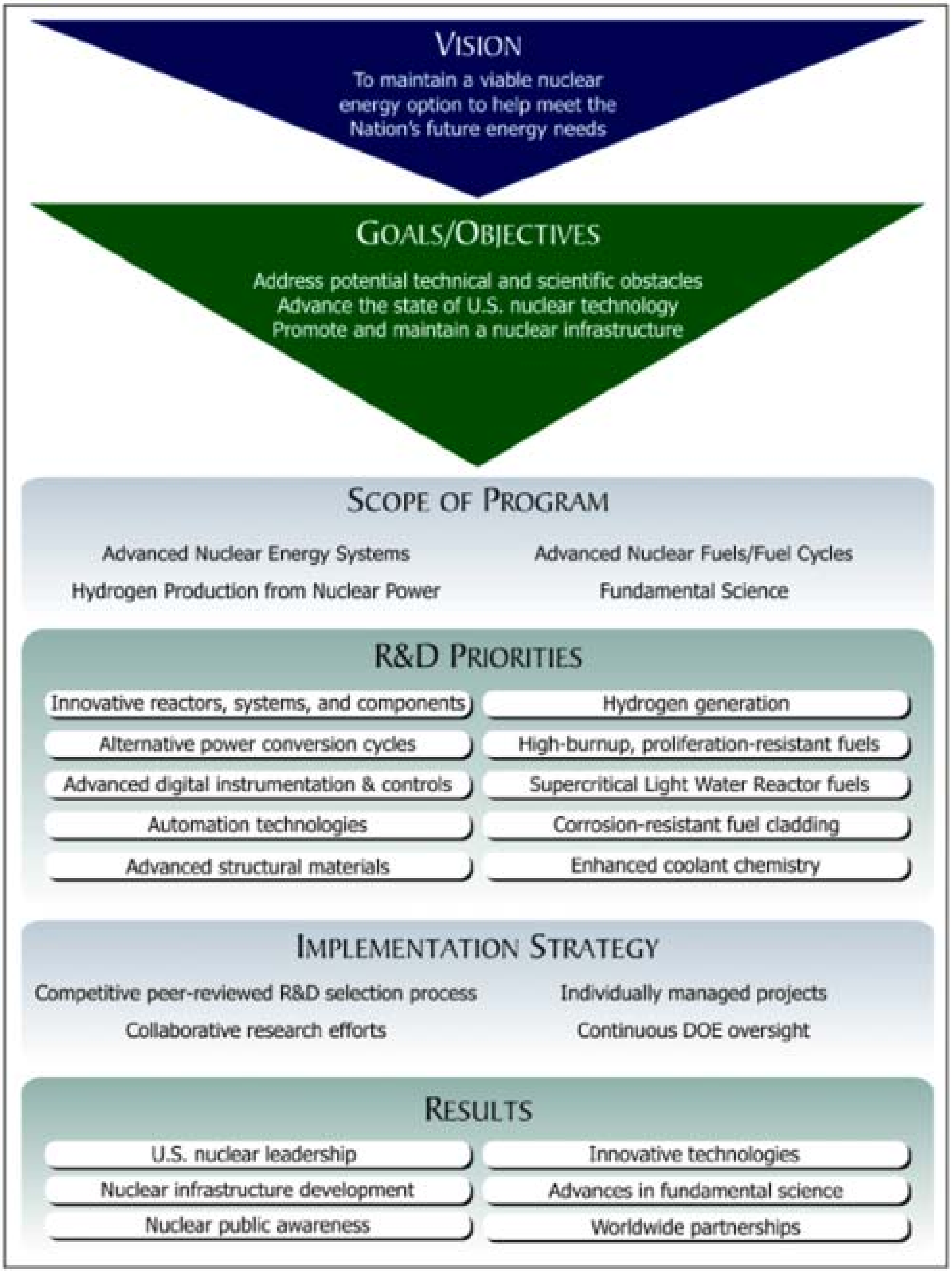


***Figure 1**. Overview of I-NERI Program*

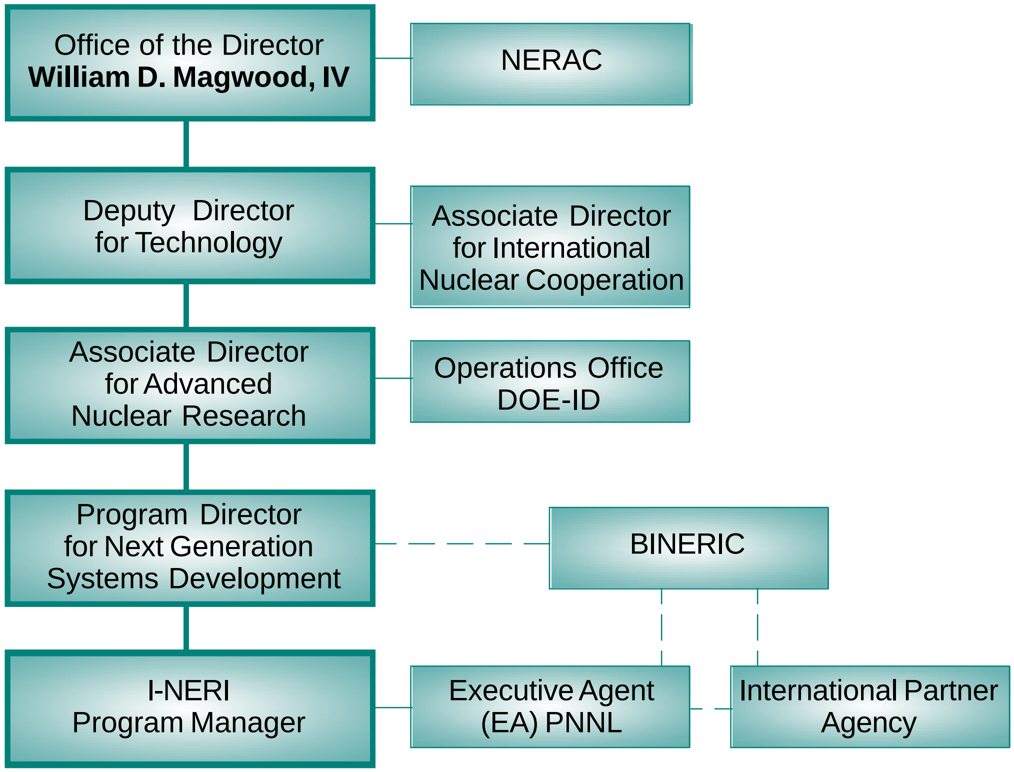


Figure 2. *Office of Nuclear Energy, Science and Technology I-NERI Organizational Chart*

FY 2003 Programmatic Accomplishments

- Completed FY 2002 annual project performance reviews for both U.S./France and U.S./Republic of Korea collaborations and confirmed projects approved for ongoing support.
- Completed awards to the five proposals selected in the FY 2002 U.S./Republic of Korea competitive procurement.
- Signed new I-NERI cooperative agreements with the European Commission, Canada, and Brazil.

Planned FY 2004 Programmatic Accomplishments

- Complete FY 2003 annual project performance reviews for both U.S./France and U.S./Republic of Korea collaborations and confirm projects approved for ongoing support.
- Complete three projects in the U.S./France collaboration that are currently planned for completion in FY 2004.
- Initiate two new I-NERI collaborations with new international partners.

Current I-NERI Collaborations

The establishment and successful management of the three existing international collaborations was the primary accomplishment of DOE in the reporting period. Brief descriptions of the current I-NERI collaborations are as follow:

United States/France Collaboration

The collaborating agency in France is the *Commisariat a l'Énergie Atomique* (CEA). The U.S./France collaboration focuses on developing Generation IV advanced nuclear system technologies that will enable the United States and France to move forward with cutting-edge generic R&D that can benefit the range of anticipated future reactor and fuel cycle designs.

United States/Republic of Korea Collaboration

The participating agency in the Republic of Korea is the Ministry of Science and Technology. The U.S./Republic of Korea collaboration focuses on advanced technologies for improving the cost, safety, and proliferation resistance of nuclear energy systems. The U.S./Republic of Korea I-NERI projects have been selected competitively from researcher-initiated proposals based upon the results of independent peer evaluation processes.

United States/OECD Collaboration

The United States has teamed with the Nuclear Energy Agency of the OECD and a number of its 30 member states to conduct reactor materials experiments and associated analysis. The U.S. funding team consists of NRC, EPRI, and DOE. This is a single project and, at the present time, there are no planned additions to this collaboration.

Descriptions of the workscopes, listings of funded projects, and brief project status reports are provided in Sections 5, 6, and 7 and Appendices A, B, and C for the U.S./France, U.S./Republic of Korea, and U.S./OECD collaborations, respectively.

I-NERI Program Funding

For projects funded through the end of FY 2003, funding for the I-NERI program was part of the overall NERI appropriations. There is no industry cost-share, however, the I-NERI program receives approximately a 50 percent cost-share from partnering countries. FY 2001 appropriations amounted to a total of $6.8M. Appropriations for FY 2002 amounted to $9.1M, which provided for continuation of FY 2001 projects and $2M for new starts. The FY 2003 appropriations were $6.8M and have been used to fund the continuation of FY 2001 and FY 2002 projects. About $2.5M was provided for new starts. The I-NERI appropriation for FY 2004 is $4.2M, which provide for the continuation of ongoing I-NERI projects.

The following provides a complete list of program participants.

U.S. National Laboratories

Argonne
Brookhaven
Idaho
Oak Ridge
Sandia

Award Profiles

The current cumulative organizational profiles for awards for the U.S./France, U.S./Republic of Korea, U.S./OECD, and overall I-NERI collaborations (based on numbers of domestic and international organizations) are illustrated in Figures 3, 4, 5, and 6 respectively.

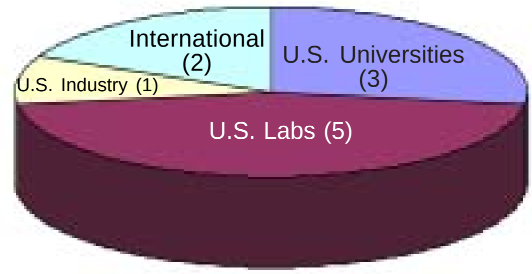


***Figure 3**. U.S./France Organizational Profile*

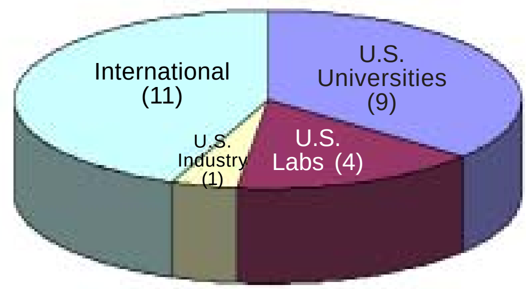


***Figure 4**. U.S./Republic of Korea Organizational Profile*

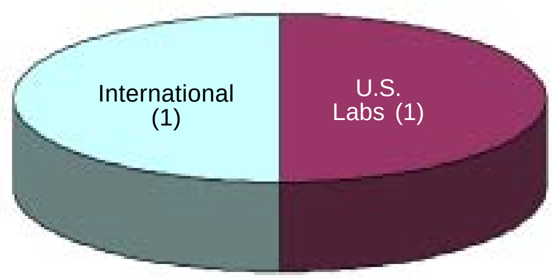


***Figure 5**. U.S./OECD Organizational Profile*

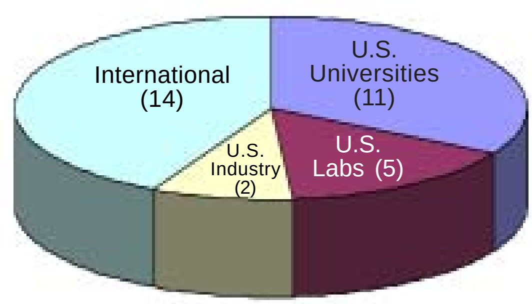


***Figure 6**. Overall I-NERI Organizational Profile*

U.S. Universities

Iowa State University
Massachusetts Institute of Technology
Ohio State University
Pennsylvania State University
Purdue University
University of California, Santa Barbara
University of Illinois, Chicago
University of Manchester
University of Maryland
University of Michigan
University of Wisconsin

U.S. Industrial Organizations

General Atomics
Westinghouse Electric

International Collaborators

Commisariat a l'Énergie Atomique
Framatome, ANP
Korea Atomic Energy Research Institute
Korean Electric Power Research Institute
Korea Advanced Institute of Science & Technology
Korean Maritime University
Chosun University
Cheju University
Seoul National University
Pusan National University
Chungnam National University
Hanyang University
Korea Hydro & Nuclear Power Co.
Organisation for Economic Cooperation and Nuclear Energy Agency

5.0 U.S./France Collaboration

The U.S./France collaboration was the first I-NERI agreement to be implemented. The competition for project awards was limited to DOE and CEA laboratories in recognition of limited budgets and the desire of the parties to facilitate timely initiation of the program.

Workscope

R&D topical areas selected in FY 2001 by BINERIC for the initial competition were as follows:

- Advanced Light Water-Cooled Reactors

- Advanced Gas-Cooled Reactors
  - Advanced Gas-Cooled Reactor Concepts
  - Fuel Development
  - High-Temperature Systems Technology
  - Mechanistic Behavior Model for Triple-Coated (TRISO) Fuel Particles

- Advanced Fuel and Materials Development
  - Nano-Composited Steels

- Radiation Damage Simulation

- Advanced Fuel Cycle Chemistry

- Advanced Energy Products.
  - Hydrogen Production

Three projects were awarded to teams of DOE and CEA federal laboratories in FY 2001 and a fourth in FY 2002. A fifth project was added to the DOE/CEA collaboration at the end of FY 2002 in the area of hydrogen production using nuclear energy.

Projects

Appendix A provides a list of projects currently funded under the DOE/CEA I-NERI collaboration through the end of FY 2003.

6.0 U.S./Republic of Korea Collaboration

The U.S./Republic of Korea collaboration was the second I-NERI agreement to be implemented. The project award selection was competitive and open to all U.S. and Republic of Korea participants.

Workscope

The R&D topical areas selected by the U.S./Republic of Korea BINERIC in FY 2001 for the initial competitive procurement were as follows:

- Advanced Instrumentation, Controls, and Diagnostics

- Advanced Light-Water Reactor Technology.

In December 2001, six projects were awarded as a result of this solicitation.

For the second solicitation under the U.S./Republic of Korea I-NERI collaboration in FY 2002, the BINERIC broadened the R&D topical areas to encompass the generic I-NERI R&D workscope more generally:

- Next-generation reactor and fuel cycle technology (including nonproliferation and safety)

- Innovative nuclear plant design, manufacturing, construction, operation, and maintenance technologies (including instrumentation, controls, and robotics)

♦ Advanced nuclear fuels and materials.

The second U.S./Republic of Korea I-NERI procurement was completed at the end of FY 2002 and resulted in the award of five new collaborative projects for FY 2003.

Projects

Appendix B provides a list of projects funded under the DOE/Republic of Korea I-NERI collaboration through the end of FY 2003.

7.0 U.S./OECD Collaboration

The U.S./OECD collaboration has an I-NERI agreement with one funded project and there is currently no plan for addition of other projects. The U.S. Nuclear Regulatory Commission and the U.S. Department of Energy jointly fund this collaboration.

The title of the U.S./OECD project is *Melt Coolability and Concrete Interaction (MCCI)*. The lead technical institution is the Argonne National Laboratory. The OECD/NEA is the active international organization contributing to the funding of the research at the Argonne National Laboratory. The OECD/NEA member countries are very interested in this project and are following it through the OECD/NEA.

Workscope

The MCCI project is primarily experimental in nature and the scope is as follows:

♦ Resolve ex-vessel debris coolability issues through a program that focuses on providing both confirmatory evidence and test data for the coolability mechanisms identified in the Melt Attack and Coolability Experiments integral effects tests.

♦ Address remaining uncertainties related to long-term, two-dimensional, molten core-concrete interaction under both wet and dry cavity conditions.

The project was awarded in March 2002 and will be funded for three years on an annual basis.

Project

A brief project summary describing the status of this research through the end of FY 2003 is provided in Appendix C.

Appendix A

U.S./France Collaboration
Project Summaries/Abstracts

International Nuclear Energy Research Initiative

Project #	Title
2001-002-F	Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/ Fast Neutron Spectrum
2001-003-F	Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels
2001-006-F	OSMOSE - An Experimental Program for Improving Neutronics Predictions of Advanced Nuclear Fuels
2001-007-F	Nano-Composited Steels for Nuclear Applications
2002-001-F	High-Efficiency Hydrogen Production from Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting

International Nuclear Energy Research Initiative

Development of Generation IV Advanced Gas-Cooled Reactors with Hardened / Fast Neutron Spectrum

Principal Investigator (U.S.): TYC Wei, Argonne National Laboratory

Principal Investigator (France): J. Rouault, DEN/DER/SERI CEA Cadarache

Collaborators: Brookhaven National Laboratory; General Atomics; Massachusetts Institute of Technology; Oak Ridge National Laboratory; Framatome – ANP (Fra-ANP), Lyon

Project Number: 2001-002-F

Project Start Date: January 2002

Project End Date: September 2005

Reporting Period: October 2002 — September 2003

Research Objective

The project objective is to design a Gas Fast Neutron Reactor (GFR) with a high level of safety and full recycling of the actinides, which also must be highly proliferation resistant and attractive in terms of economics.

Research Progress

Collaboration has been very effective, and three progress meetings have been held. Work on the definition of design goals and criteria for the GFR has been finalized and a research and development (R&D) plan focusing on the important issues for the design (fuel, high-temperature structural materials, safety, etc.) has been issued. It is now the basis for the Generation IV GFR R&D plan definition. Collaboration between the two countries has resulted in the completion and release of the topical report GFR 002, Rev. 0 “GFR R&D Plan.” This met the milestone for producing a 10-year GFR development plan, which can be used in the Generation IV program effort on the GFR.

The first year of the project ended on February 28, 2003. Work on the first two quarters in FY 2003 focused on meeting the milestones and deliverables of the project for the first year. This work completed the exploratory stage of the collaborative project, where several new and innovative concepts for fast reactor core designs and corresponding passive decay heat removal schemes were explored and integrated. Implications for the primary system and balance of plant layout were also investigated. Technical exchanges of ideas regarding the development of the required innovative fuel forms continued between CEA and ANL. In addition, interaction was started with DOE’s Advanced Fuel Cycle Initiative (AFCI)/Generation IV programs. Similar interactions began with the Generation IV program in the area of structural materials.

In summary, Task 3 (Exploratory Core/Fuel Forms/Primary Systems Concepts) was completed this year. Subtask 3.1 on core designs refined a number of the core neutronic design calculations performed during the last year and produced report GFR 004, “Gas-Cooled Fast Reactor Core Designs.” Subtask 3.2 on core-decay heat-passive safety completed calculations on passive decay heat-removal concepts for block/plate-based cores, pebble/particle-based cores, and pin-based cores. A combination of natural convection and in-core heat exchangers is being proposed. The implications for containment response were examined. In the case of the pebble-bed core, a design-specific core dump system was explored. This effort produced the following deliverable reports:

1. GFR006 “Passive Decay Heat Removal in Pin-Based Cores”
2. GFR007 “Cold Finger In-Core Decay Heat Removal for Pebble Bed GFR Designs”
3. GFR 008 “Passive Decay Heat Removal in Particle/Pebble Based Core”
4. GFR009 “Analysis and Recommendation for Decay Heat Removal from Gas Cooled Fast Reactor”
5. MIT-ANP-TR-095 “Analysis of a Convection Loop for GFR Post-LOCA Decay Heat Removal.”

In addition, a top-level document on safety approach and goals was produced entitled GFR005 “Gas-Cooled Fast Reactor Safety Approach General Recommendations.”

The investigation of the implications of the safety approach on passive decay heat removal produced the following reports:

1. GFR003, “High Temperature Vessel Structures”
2. GFR010, “Feasibility Study on the Application of a Leak Before Break Procedure in Metallic Vessels of a GFR.”

Finally, regarding the possibility of an alternative primary coolant, supercritical CO_2 as a backup to the reference primary coolant helium produced the deliverable entitled MIT-ANP-TR-090, “CO_2 Brayton Cycle Design and Optimization.”

Preparations were made toward the end of the first year for the second CEA Cadarache-France/Argonne National Laboratory (ANL)-U.S. technical workshop on the I-NERI GFR project. This workshop was held at ANL March 31-April 3, 2003, to present and discuss the results of the work during the first year. Based on this discussion, down selection of innovative core/reactor design concepts and safety approaches occurred. These selected concepts now form the basis of the concept trade studies that are being carried out in the second year of the I-NERI project. Agreement was reached between France and the U.S. project staff on the collaborative work scope to implement the trade studies in the second year.

It was agreed that the major design goals would focus on 300 to 1000 MWe cores with power densities in the range 50 to 100 MW/m^3. For sustainability and non-proliferation reasons, self-generating cores will be considered with integral homogeneous recycling of all actinides present in spent fuels.

Current important common conclusions are as follows:

- Design Goals and Criteria: an agreement was reached on the major goals. Discussions took place on the necessity of a criterion on the Pu inventory per GWe. It was agreed that this would be necessary to easily deploy the GFR. A value not exceeding 15 t/GWe was found to be a necessity at least for situations in France and the U.S. The question concerning the origin of the criteria on the containment pressure resistance (value and time required) remains to be assessed.

- Fuels and Materials: recommendations for the ongoing detailed studies were made on both the design and material choices. Detailed comparison of carbide and nitride fuels concludes that these two remain serious candidates and that only ongoing R&D could bring new elements for focusing the choice. Nevertheless, current French evaluations stressed the difficulties of ^{15}N enrichment and recovery. Carbide fuel was proposed to be the reference fuel material for the design studies. As an ambitious choice, block/plate-type dispersed fuels (carbide fuel dispersed in a SiC matrix, with SiC as in-core structural material) was selected as reference. The reference backup (concept robustness) is pintype solid solution fuel. Again, carbide fuel is the reference and, for the cladding, SiC is the ambitious solution whereas ODS steel is envisioned as the backup solution (even if it may require a revised value for the coolant temperature). Particle fuel was ranked third.

- Core Design: using criteria such as ease of deployment, reactivity effects, ability for passive DHR, and flexibility (burnup increase, volumetric power, transmutation, etc.), selecting core designs is based on common detailed neutronic characterization, thermal-hydraulic assessments, and preliminary evaluation of core’s behavior in anticipated transients without scram and follows the fuel and material reference and reference backup recommendations. The reference design must be optimized in the range of 50/50 to 70/30 (for the dispersed fuel composition) and 50 to 100 MW/m^3 for the volumetric power to be as near as possible to 100 MW/m^3 (Pu inventory criteria). For the pin-type core, refractory alloy claddings are excluded for neutronics and safety considerations. In addition to these reference designs for the second-year detailed studies, some effort will be spent on particle fuel (France) and pebble fuel (U.S.).

- DHR Approach: results concluded that the GT-MHR safety strategy doesn’t apply to the GFR, and a well-weighted mix is required between passive, semi-active, and active systems where gas plays a major role (circulators, fiddler crab, injection, natural convection, wall radiation, backup pressure). The general strategy in the definition of the reference safety case for depressurization studies (in particular, choice of the size of the breach), remains to be finalized. The use of in-core devices (heat pipes, heat exchangers, cold legs, etc.) appears to raise many technological and safety questions. In any case, their efficiency remains limited. It has been proposed not to include those types of systems in the reference and reference backup designs.

- Primary System: primary system exploratory studies will be pursued. For the second year studies, it was agreed to select the two following basic options: 1) up-core flow and top-entry refueling, and 2) the rest needing further assessment (position of control rods, vessel material, etc.). More detailed primary and balance of plant (BOP) options will be developed at the end of the contract second year.

Detailed characterization of commonly selected cores is now underway, as are exploratory studies on the choices to be performed for the primary system and BOP. Recent results demonstrate the interest to consider higher power unit (typically 2400 MWt) to recover margins with respect to the fuel design. In particular, it is identified that the use of a 50/50 cercer fuel would remain compatible with a high level of volumetric power (100 MW/m^3) and offers the possibility for an increase of burnup. Based on this consensus, work has been initiated on the second year plant trade studies.

Preliminary results from these on-going trade studies were discussed at the third semi-annual workshop/meeting held at Cadarache in October 2003.

Planned Activities

Plant trade studies will be completed and the results documented in the reports shown in the milestone/deliverable table. These documents will include reports on the core design and GFR safety approach. The trade studies of the natural convection cooling in the block/pin core options will be completed. The work on vessel/internals/shielding will also be completed. In addition, the experiments to study the deposition of TiN on carbon-coated oxide ceramic beads will be completed.

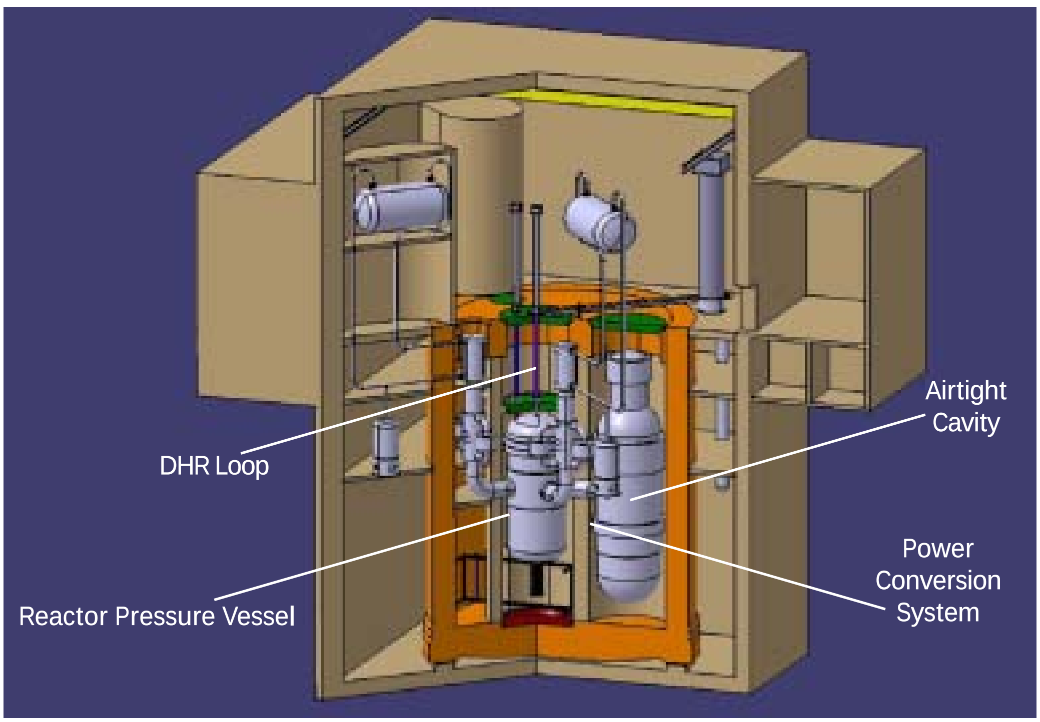


***Figure 1**. 600 MWth GFR Global Picture*

International Nuclear Energy Research Initiative

Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels

Primary Investigator (U.S.): David Petti, Idaho National Engineering and Environmental Laboratory

Primary Investigator (France): Philippe Martin, DEN/DEC/SESC CEA

Collaborators: Massachusetts Institute of Technology

Project Number: 2001-003-F

Project Start Date: September 2001

Project End Date: September 2004

Reporting Period: October 2002 — September 2003

Research Objective

The objective of this I-NERI project is to develop improved fuel behavior models for gas reactor coated-particle fuels and to develop improved coated-particle fuel designs that can be used reliably at very high burnups and potentially in fast gas-cooled reactors.

Research Progress

The development of the INEEL fuel performance code, PARFUME, continued from earlier efforts. Modeling of other potentially important failure modes such as debonding and asphericity was compiled. A paper on the statistical method was published in the *Journal of Nuclear Materials*. A benchmark test matrix was developed for use by CEA and INEEL in comparing the two fuel performance models. Preliminary calculations of the stresses in a coated particle have been performed by the CEA using the ATLAS finite element model. This model and the material properties and constitutive relationships will be incorporated into a more general software platform termed Pleiades. Pleiades will be able to analyze different fuel forms at different scales (from particle to fuel body) and to handle the statistical variability in coated particle fuel. Initial benchmarking between the French and U.S. models show good agreement.

Work has begun on developing the fission product chemistry and transport module in PARFUME. The chemistry module was developed last year under internal INEEL funding. In addition, a model for the release to birth ratio for a failed particle and uranium contamination has been developed and integrated into the code.

In the area of fission product transport, an extensive review of the literature was performed to understand the physical mechanisms for fission product transport in PyC and SiC. Mechanisms include: vapor transport via Knudsen diffusion for gaseous fission products, intercalation of alkali and alkali-earth fission products like Cs and Sr in the PyC layers, grain boundary diffusion, surface diffusion, and bulk diffusion. Diffusivities for Ag, Xe, Cs, and Sr have also been gathered from the literature. Knudsen and viscous pressure-driven diffusion calculations have been performed to examine transport in pores or cracks in a layer of the TRISO coating. In addition, scoping calculations have begun using a diffusion and trapping code called TMAP to model fission product transport from the particles. The code can model diffusion and trapping of multiple species as well as diffusion in the presence of a temperature gradient (the so-called Soret Effect.) The code also has a thermal model that has been used to determine the temperature distribution and thermal gradient in each layer of the coated particle. Sensitivity studies have been conducted to examine thermal diffusion effects, which are most important in the low-density buffer where large thermal gradients could be expected depending on the power density in the fuel particle.

Diffusion couple experiments to study Ag and Pd transport through SiC continued. Analysis and characterization of the samples continues. Silver ion implantation studies indicate that silver does not move by classic condensed phase Fickian diffusion in the SiC. Knudsen pressure-driven diffusion looks like a promising mechanism for silver transport. This finding would imply transport via nanoporosity or nanocracks in the SiC. Work on Pd behavior has begun and will continue next year.

Calculations have been performed to examine the feasibility of using TRISO-coated particles in a gas-cooled fast reactor. Damage rates as well as helium and hydrogen production in PyC and SiC were calculated using a gas-cooled fast-reactor neutron spectrum. The calculated damage rates (~ 50 dpa) are high enough that radiation damage could influence the material properties. In particular, the high radiation damage to the C layers would result in unacceptable dimensional change. At this level of radiation damage, SiC would also see significant property changes in terms of strength, swelling, and other material properties. The use of the traditional TRISO coatings is not recommended for coated particle fuels in fast spectrum reactor applications.

Planned Activities

It is planned that the I-NERI project will be completed during FY 2004. This will encompass the completion and final documentation of the following activities: In the Ag-SiC migration study, fission product-SiC interaction model and the ZrC coating study will be completed. In addition, the feasibility study of using particle fuel in a fast neutron environment will be completed, and finally, an assessment of predictive irradiation performance for advanced prototype particle fuel will be completed.

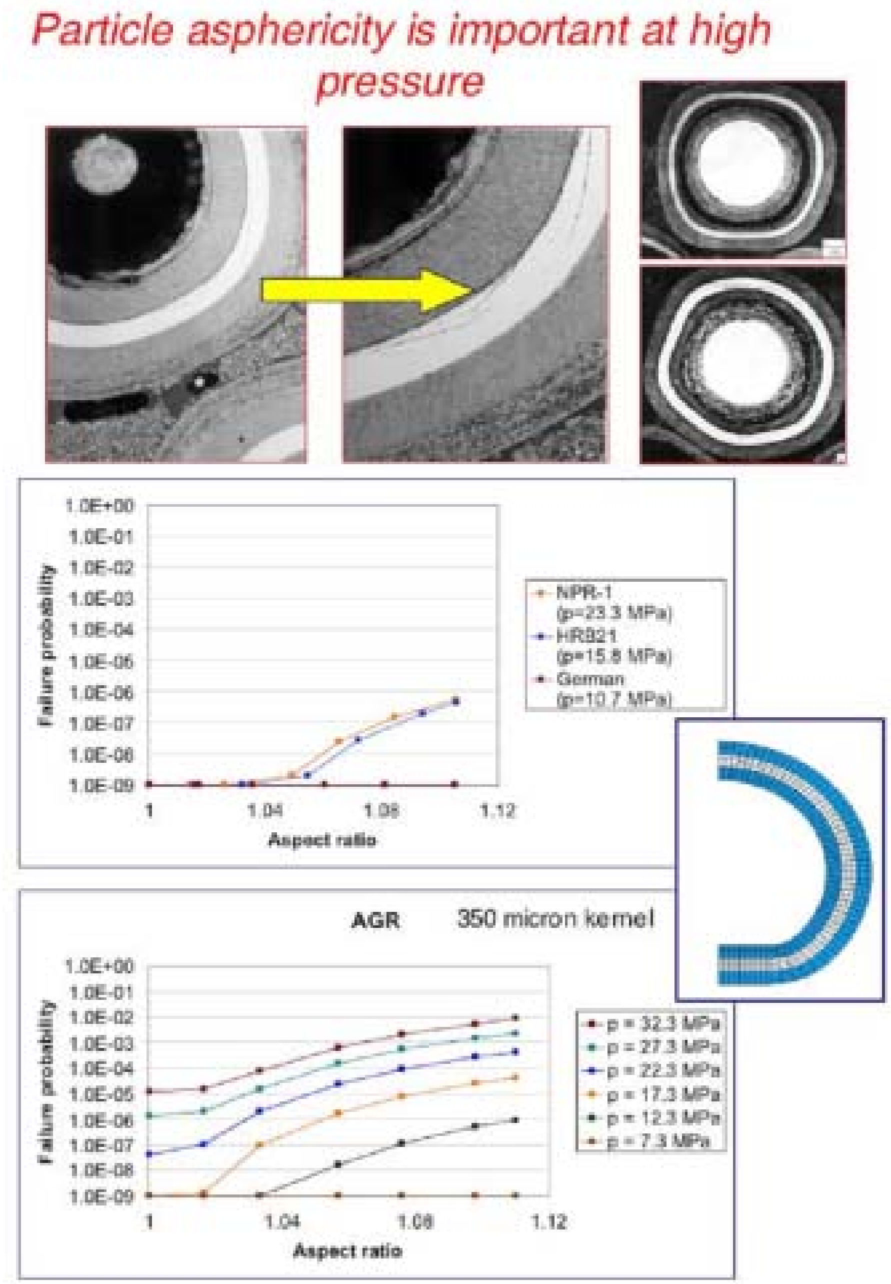


***Figure 1**. The influence of asphericity (aspect ratio) and internal gas pressure on particle failure probability.*

International Nuclear Energy Research Initiative

OSMOSE – An Experimental Program for Improving Neutronics Predictions of Advanced Nuclear Fuels

Principal Investigator (U.S.): Dr. Raymond Klann, Argonne National Laboratory

Principal Investigator (Int.): Dr. Jean-Pascal Hudelot, CEA-Cadarache

Collaborators: Prof. John Lee, University of Michigan

Project Number: 2001-006-F

Project Start Date: September 2001

Project End Date: September 2004

Reporting Period: October 2002 — September 2004

Research Objective

The objective of this collaborative program between the U.S. DOE and the French CEA is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs, and to provide the experimental data for improving the basic nuclear data files. The main outcome of the OSMOSE measurement program will be an experimental database of reactivity-worth measurements in different neutron spectra for the heavy nuclides. This database can then be used as a benchmark to verify and validate reactor analysis codes. The OSMOSE program aims at improving neutronic predictions of advanced nuclear fuels through measurements in the MINERVE facility on samples containing the following separated actinides: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, and ^{245}Cm.

Research Progress

The collaborative project is defined by five major tasks: 1) reactor modifications, 2) reactor modeling, 3) sample preparations, 4) experiments, and 5) data analysis.

Task 1 – Reactor Modifications

Reactor modifications were necessary to maintain the MINERVE facility consistent with operating requirements specified by safety authorities. In addition, the modifications were to refurbish equipment and upgrade systems to allow operations and experiments to be conducted more efficiently and with higher quality assurance of the results.

The upgrade of the control system of the reactor and the control room was completed in December 2002. The new SIREX control system demonstrates better performance than the previous system. Maintenance of the pilot rod was also completed in December 2002, improving the positioning of the rotary rod. It now provides an improved reproducibility for the oscillation measurements.

The oscillator system was upgraded and new software associated with the numerical recording and clock systems were installed. A new system for measurement of the fission rate axial profile was developed and qualified. Finally, new ionization chambers for the reactor monitoring system (high-level and low-level chambers) were installed.

In the end, all of the reactor modifications were completed on schedule, and approval to operate was granted by the French safety authorities in March 2003.

Task 2 – Reactor Modeling

The analytic effort is being performed using separate suites of reactor analysis codes in the U.S. and France. In this manner, a cross comparison can be performed on the results to identify potential errors in the cross-section evaluations in the numerical methods and assumptions used within the codes. This will allow the improvement of the codes.

The initial task included developing a Materials Specification Document for the MINERVE facility. This reference document includes the detailed description of the geometry and compositions for all regions of the reactor so that computational models can be assembled.
The initial tasks included the development of reactor models using MCNP for the R1-UO2 and R1-MOX core configurations.

The models were used to calculate the energy spectrum in the sample location, the excess reactivity of the core, the reactivity with the control rods in the critical positions, the reactivity effect of each control rod, the spectral indexes in the sample location, and the axial and radial power distributions in the experimental region. Calculation results are discussed in the data analysis section below.

Task 3 – Sample Preparations

The OSMOSE program requires the fabrication of oxide samples containing separated actinides (^{232}Th, $^{233, 234, 235, 236, 238}$U, ^{237}Np, $^{238, 239, 240, 241, 242}$Pu, $^{241, 243}$Am and $^{244, 245}$Cm). The samples consist of assembled fuel pellets containing the isotopes of interest and a double zircaloy cladding.

The OSMOSE oven was tested with success at the company where it was manufactured in May 2003. After transfer to CEA Marcoule, the oven underwent extensive testing in the mockup shop to demonstrate proper operation and functioning before installation in the hot cell. The qualification tests were completed in September 2003.

Pellet fabrication for ^{237}Np (x 2), ^{232}Th (x 2) is still planned for the end of 2003. Fabrication of the UO_{2nat} pellets was completed and included analysis of the pellets. Sintering of individual pellets was performed in a dilatometer on 23 UO_2 pellets. 20 pellets were acceptable, with a density more than 95% of theoretical density, with a mean density of 95.9%TD, and within tolerances on the dimensions and reproducibility of the dimensions. Five pellets were sintered with the dilatometer having a ratio of ^{232}Th inside UO_2 equal to 4% (corresponding to the highest ratio occurring in OSMOSE pellets). It was necessary to perform this study because of the high ratio of ^{232}Th (4%) so that the consequences on the sintering process could be examined. Examination of the pellets by electron microscopy revealed no porosity or cracks inside the $UThO_2$ (4%) pellets. This served to qualify the modified MIMAS process developed by CEA Marcoule.

Isotope stocks (^{243}Am, ^{240}Pu, ^{241}Pu, and ^{242}Pu) were shipped from the U.S. to CEA Marcoule in August 2003. The total inventory of ^{241}Pu is only 0.3 g. This means that only one sample containing ^{241}Pu can be manufactured instead of two. All the other isotopes needed for the OSMOSE program are now supplied and available in CEA Marcoule. Processing of the isotopes before use in sample preparation is necessary.

The last activity to report was the laser welding technique test at CEA Marcoule, which will be formally qualified by an external company (Welding Institute) in October 2003.

Task 4 – Experiments

The objective of the measurements is to characterize the reactor sufficiently that the neutron flux, spectrum, and power distribution are known and can be used for computational modeling and assessment of neutron cross sections. The goal of the calibration measurements is to demonstrate the oscillation technique on known and calibrated samples and to use this data to support the development of the analytic technique for measuring reactivity-worths of separated actinides and cross-section evaluations.

Experiments first consisted of measurements in the R1-UO2 configuration (PWR spectrum) that was loaded for the startup of MINERVE after modifications. The core was reloaded with the R1-MOX configuration for the HTC-MOX program and measurements continued. The HTC-MOX measurements were completed and the core is being reloaded with the R1-UO2 configuration. The measurements that were conducted included measurements of the safety parameters, reactor characterization measurements, and measurements of calibration samples to establish a calibration curve for the OSMOSE program measurements.

The reactivity-worth of the control rods was determined using the rod drop technique between the critical position of each control rod to the fully inserted position. The reactivity excess of the core was measured by the doubling time method associated with the Nordheim curve of the reactor for both configurations. The critical positions for each of the individual control rods (and banks of control rods) were also determined for both configurations.

Measurements of the axial fission rates were performed using the POLINE device with miniature ^{237}Np and ^{235}U fission chambers. The accuracy on the measurements was better than 1%, and the two measurements were consistent over the height of the fuel in the experimental region.

Extensive gamma spectroscopy measurements were also performed on series of fuel pins to determine the relative power distribution (i.e., total fission rate distribution) inside the experimental region. Additional core characterization measurements were also performed, including spectral indexes and the conversion ratio of ^{238}U.

Task 5 – Data Analysis

This task addresses the analysis and reduction of data for each series of measurements for the different core configurations. In general, data analysis includes the review and analysis of the raw data for the full range of separated samples for the OSMOSE program in each reactor configuration, the analysis of the raw data from the calibration and test measurements performed with calibration samples of differing uranium and boron compositions, and the analysis of the raw data for all spectral indexes and axial and radial distributions measurements performed to support the OSMOSE program.

Data analysis activities were not originally scheduled for 2003. However, preliminary data analysis was performed and included C/E results of the reactivity-worth of the control rods in R1UO2 and in R1MOX, the axial profile of ^{237}Np and ^{235}U fission rates, the radial power profile, spectral indexes in R1UO2 and in R1MOX, and the modified conversion ratio in R1MOX.

An example of the results of the comparisons for the spectral indexes in the R1-UO2 configuration is shown in Table 1 and Figure 1. In R1-UO2, the C/E ratios are consistent within better than 1.5% for ratios of fissile isotopes, i.e., Pu9/U5 and Pu1/Pu9. For the Pu0/Pu9 and Pu2/Pu9 ratios, the calculations overestimate experimental results by about 7% and 33%, respectively. These discrepancies can be explained by the large experimental uncertainties on the indexes and by a need to improve knowledge and accuracy of ^{240}Pu and ^{242}Pu fission cross sections in both ENDF-B6 and JEF2.2.

***Table 1**. Comparison of Experiments and Calculations of Spectral Indexes in R1-UO2 Lattice*

	ENDF-B6		JEF2.2	
Ratio	**C/E**	**s.d.(%)**	**C/E**	**s.d. (%)**
Pu9/U5	1.008	3.5	1.012	3.5
U8/U5	1.182	7.6	1.22	7.8
Pu0/Pu9	1.067	7.6	1.125	7.8
Pu1/Pu9	1.004	3.6	1.003	3.6
Pu2/Pu9	1.334	15.4	1.461	16.1
Np7/Pu9	0.938	3.7	0.953	3.7

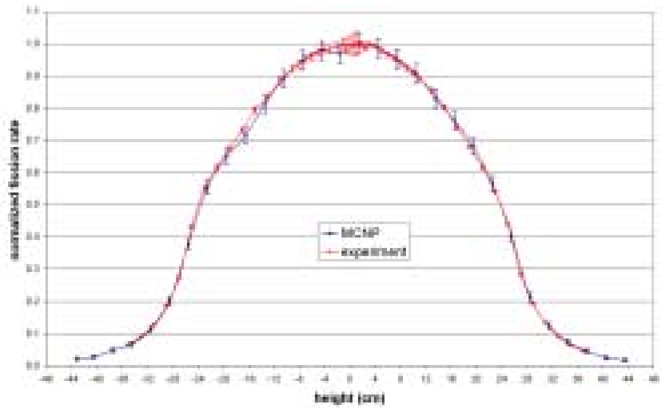


***Figure 1**. A comparison of calculations to measurements of the axial fission rate for ^{237}Np. Excellent agreement was observed over the full height of the fuel pin.*

Planned Activities

Activities scheduled in 2004 are continual effort from 2003. For the reactor modeling and pre-analysis efforts, the tasks include finishing the deterministic models for the R1-UO2 and R1-MOX configurations, providing reactivity-worth estimates for the calibration samples in the R1-UO2 and R1-MOX configurations and providing reactivity-worth estimates for the OSMOSE samples in the two configurations.

As part of the sample preparation tasks, the OSMOSE oven will be qualified. In addition, pellets will be fabricated with ^{237}Np and ^{232}Th, pellets for eight of the other samples will be fabricated, and purification and processing of the isotope stocks will be completed.

Most of the MINERVE operations planned in 2004 are to support the HTC-MOX program, VALMONT program, and for reactor operator training – leaving little time for experimental measurements. However, reactor characterization measurements and calibration measurements will be performed for the R1-UO2 configuration.

The significant effort in 2004 will be on data treatment, analysis, and interpretation of all of the measurements performed in 2003 and 2004. These measurements include all of the reactor characterization, safety parameters, and calibration measurements for the R1-UO2 and R1-MOX configurations.

International Nuclear Energy Research Initiative

Nano-Composited Steels for Nuclear Applications

Principal Investigator (U.S.): Roger Stoller, Oak Ridge National Laboratory

Principal Investigator (Int.): Ana Alamo, CEA

Collaborators: R. Robert Odette, UCSB

Project Number: 2001-007-F

Project Start Date: September 2001

Project End Date: September 2004

Reporting Period: October 2002 — September 2004

Research Objective

The primary goal of this I-NERI project is to develop a scientific knowledge base on the processing, deformation mechanisms, fracture behavior, and radiation response of existing oxide-dispersion strengthened (ODS) steels to guide future development of advanced alloys capable of meeting the Generation-IV reactor needs for higher operating temperatures.

Research Progress

Significant progress has been made on the primary objectives, which include: 1) further characterization of the microstructure and mechanical properties of reference ODS steels: 12YWT, MA-957, and PM-2000, and 2) using of this knowledge to develop an improved alloy for use in advanced nuclear reactors such as the Generation-IV designs with their demands for high operating temperatures. The reference alloys are all mechanically alloyed (MA) materials. The 12YWT is an experimental heat that was produced in a late 1990's collaboration between Kobe Steel, Nagoya University, and ORNL, while the MA-957 and PM-2000 are commercial ODS alloys produced by INCO and Plansee, respectively. The 12YWT composition is Fe-12Cr-3W-0.4Ti with 0.25 wt-% Y_2O_3. This alloy exhibited better high-temperature strength and creep properties than either conventional ferritic-martensitic steels or commercial ODS steels. The improved mechanical properties have been attributed to the very fine dispersion (radius~2 nm) of mixed-oxide (Ti+Y) clusters that were observed in the 12YWT.

Work to develop an improved ODS alloys has necessarily involved determining the salient differences between the reference materials, as well as differences in their composition and thermal-mechanical processing history that could influence final mechanical properties. Experimental investigations of processing history have included varying parameters in the mechanical alloying process, which occurs by ball milling of the alloy and oxide powders and variations in the methods and thermal conditions used to consolidate the powder into a solid product form. Three different milling techniques have been applied by the three partners, and powder consolidation has been carried out over a range of temperatures by both hot extrusion and hot isostatic pressing (HIP). An investigation of reactive milling using a model system of YNi_2 and NiO indicated that the mechanical alloy process could promote the formation of yttria particles in a metal matrix.

Microstructural analysis of the reference alloy MA-957 by atom probe (Figure 1) has demonstrated that this alloy has a nanometer-scale dispersion of the oxide clusters similar to 12YWT. This was a surprising result because previous creep testing of MA-957 indicated that its properties were inferior to 12YWT. Additional creep testing has extended the MA-957 database, as shown in Figure 2. These new results suggest that 12YWT's superiority is not as great as previously reported. Further work is underway to determine whether the differences in oxide cluster distributions are sufficient to explain the observed difference in creep strength, or if a previously undetected microstructural difference between 12YWT and MA-957 is responsible.

Extensive characterization of the PM2000 was carried out at CEA Saclay. Involving both metallurgical and mechanical properties. Both optical and electron microscopy demonstrated the presence of extensive texture with highly elongated grains. A modest but significant difference was found in tensile properties in the transverse and longitudinal directions. Greater ductility was obtained in the transverse direction, with the difference increasing at higher test temperatures.

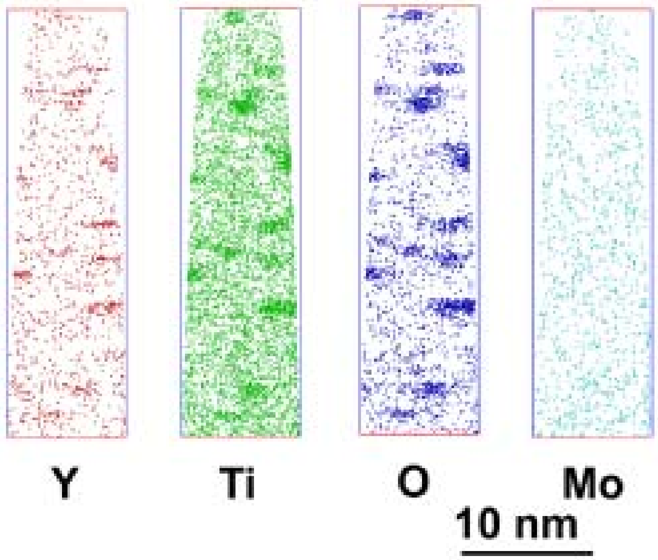


***Figure 1**. Atom probe image of oxide containing Y and Ti in MA-957.*

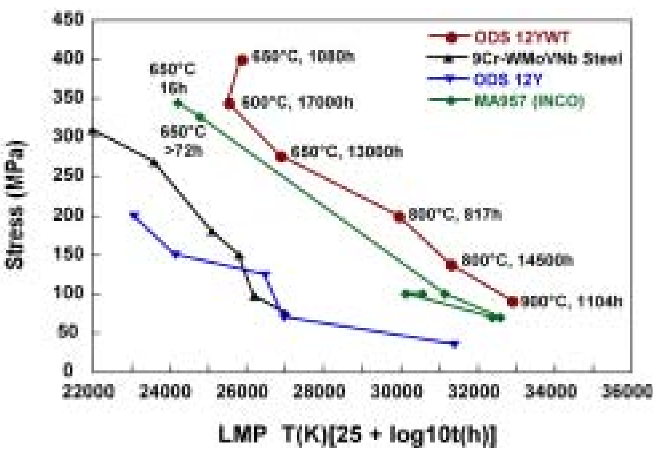


***Figure 2**. Comparison of reference ODS alloy creep data.*

Fine-grained regions of the material exhibited higher strength and lower ductility than the coarse-grained regions.

Based on work carried out in the first year, an initial developmental alloy has been prepared. A higher chromium content was chosen and a base alloy powder with a composition of Fe-14Cr-3W-0.4Ti was procured. This 14 wt% Cr alloy powder has been mechanically alloyed with 0.25 wt% Y_2O_3 powder. The base alloy powder has been designated 14YWT and has been the subject of an extensive investigation of milling and consolidation. Material has been milled by alternative techniques at ORNL and UCSB and consolidated by hot isostatic pressing (HIP) at UCSB and by extrusion at ORNL at a range of temperatures. The powders milled at UCSB and ORNL carry the designations U14YWT and O14YWT, respectively. HIPed material is designated by an "H" and extruded material by "E." Thus, a material designated OE14YWT was milled at ORNL and consolidated by extrusion, and UH14YWT was milled at UCSB and consolidated by HIPing. The as-milled powders and the as-consolidated alloys were extensively characterized. Initial mechanical property measurements have been made and compared with the reference alloys. Although the microstructure requires further optimization, the OE14YWT exhibits good strength. It is softer than the reference alloys but exhibits better ductility and work-hardening behavior. Specimens have been prepared for inclusion in a neutron irradiation experiment that will begin during the first quarter of FY 2004.

An initial evaluation of the radiation stability of the oxide clusters was obtained in a novel experiment with French collaborators at the University of Rouen (P. Pareige and E. Cadel). The experiment is illustrated in Figure 3. A pre-polished atom probe (AP) specimen of reference alloy 12YWT was irradiated with 150 keV Fe^{2+} ions to about 0.5 dpa at 300°C. Following the irradiation, a direct AP examination of the specimen was carried out. Although some fluctuations in cluster composition were noted, the clusters remained stable.

An extensive set of microstructural and mechanical property specimens were prepared from both the reference and developmental alloys for inclusion in several irradiation experiments that will begin in FY 2004. Irradiation conditions include temperatures from 300 to 700°C and doses up to 40 dpa.

Planned Activities

Further basic studies of the mechanical alloying process and powder consolidation conditions will be developed to confirm correct understanding. This will include reactive milling using the model Fe-Y and FeO system, as well as studies employing the developmental 14YWT alloy components. Some of this work will be planned and carried out during a short-term assignment of an ORNL staff member (Dr. David Hoelzer) to CEA Saclay. A significant component of the processing studies will be to minimize the residual material texture following powder consolidation.

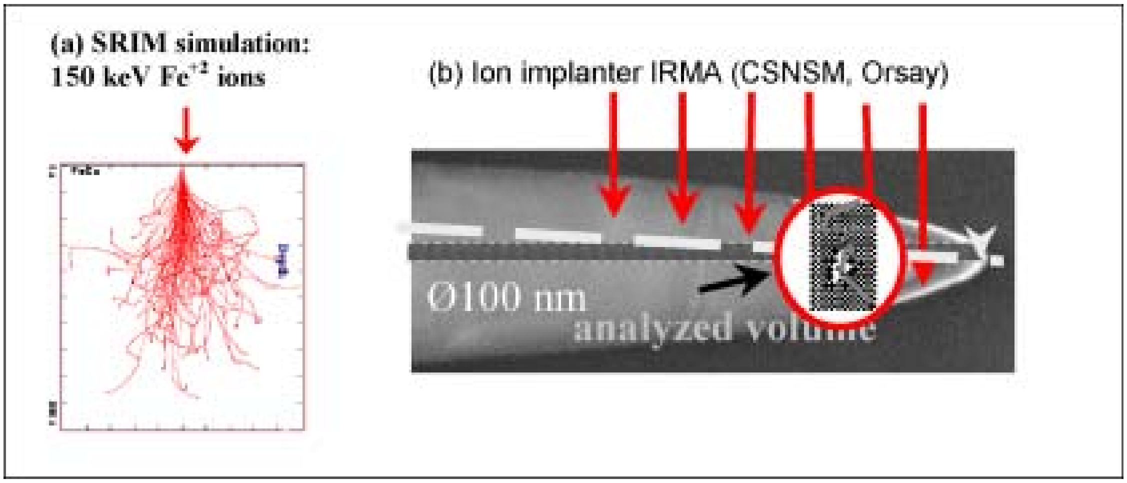


***Figure 3**. (a) Computer simulation (SRIM code) of 150 keV Fe2+ ion displacement cascade and, (b) TEM micrograph of atom probe specimen showing irradiated volume.*

Larger quantities of the OE14YWT developmental alloy variants will be prepared to permit more detailed mechanical testing, including fracture toughness measurements. The primary variables to be investigated include temperature, strain rate, and specimen orientation relative to the direction of extrusion (effects arising from material texture).

Specimens of the reference alloy MA-957 and the developmental alloy OE14YWT will be irradiated by heavy ions (14 MeV Fe ions) to ~20 dpa at 600°C to investigate the radiation stability of the nanometer-sized clusters that are responsible for their high strength. Microstructural analysis will be carried out, and mechanical property changes will be measured.

Neutron irradiation data on the reference and variants of the developmental alloys will be obtained following irradiation in the HFIR to ~10 dpa at 300°, 400°, and 500°C, and ~8 dpa at 600° and 700°C. Extensive microstructural characterization and mechanical property measurements are planned. Higher-dose neutron irradiations (up to 40 dpa) will be initiated in the HFIR, but irradiations will not be completed in FY 2004. Tensile properties will be obtained on specimens of reference alloy MA-957 that were irradiated in the OSIRIS and BOR-60 to different dose levels at 325°C, and the irradiated microstructure will be examined.

Results of initial corrosion tests will be completed on specimens of the reference alloys MA-957 and 12YWT exposed to high-temperature (750°-1000°C) helium with representative impurity levels. It is necessary to develop the ability to join the ODS materials while maintaining the required fine oxide dispersion if these materials are to see broad use in structural components. Conventional welding techniques that melt material cannot be used because the particle distributions would be coarsened. Other work at ORNL has led to the development of materials that may permit friction stir welding to be applied for joining high-strength materials such as the ODS steels. The feasibility of employing this method will be investigated.

International Nuclear Energy Research Initiative

High-Efficiency Hydrogen Production from Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting

Principal Investigator (U.S.): Paul S. Pickard, Sandia National Laboratory (SNL)

Principal Investigator (France): Stephen Goldstein, CEA

Collaborators: Gottfried Besenbruch, General Atomics (GA)

Project Number: 2002-001-F

Project Start Date: October 2002

Project End Date: September 2005

Reporting Period: October 2002 — September 2003

Research Objective

The objectives of the I-NERI project are to demonstrate the sulfur-iodine (S-I) water splitting cycle at a laboratory scale and to assess the potential of this cycle for application to nuclear hydrogen production. The S-I cycle (Figure 1) is one of the leading thermochemical cycle candidates for the production of hydrogen from nuclear energy. The project will design, construct, and test the three major component reaction sections that make up the S-I cycle. The CEA is designing and testing the prime (Bunsen) reaction section; General Atomics is developing the HI decomposition section; and SNL will develop and test the H_2SO_4 decomposition section. The results of this research will provide the technical basis for subsequent decisions on the scale up of the S-I process for nuclear hydrogen production.

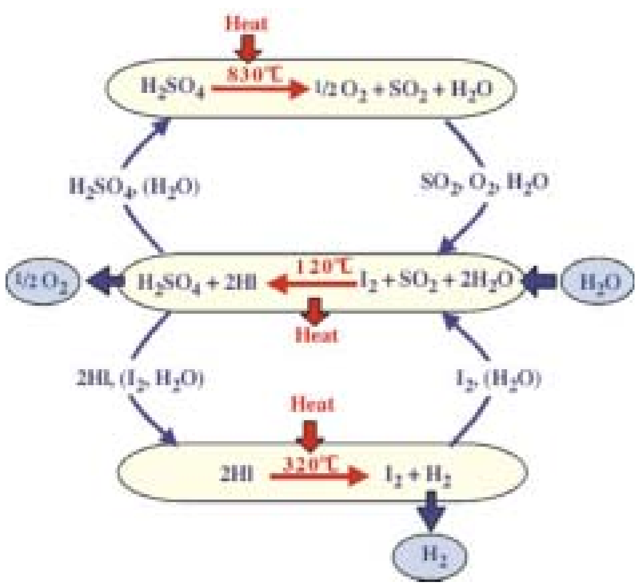


***Figure 1**. Sulphur-iodine cycle*

Research Progress

The sulfur-iodine cycle consists of three chemical reactions, which sum to the dissociation of water. During the first year of this project, work focused on preliminary designs for the component reaction sections, the development of models and analyses that define operating requirements and conditions of the component sections, and the development of experimental capabilities to measure basic thermodynamic data necessary for the S-I cycle.

Task 1 – Thermodynamic Data for $HI/I_2/H_2O$ System (CEA)

Additional data on the vapor-liquid equilibrium of the system $HI/I_2/H_2O$ are needed to properly design the HI decomposition section. Only total pressure vapor-liquid-equilibrium measurements were performed previously. Partial pressures of these components are necessary to develop an adequate understanding of this system. These measurements are difficult; therefore, the program has been divided into three steps: 1) measurement of the total pressure, up to 50 bar and 300°C using a micropressure vessel made of tantalum; 2) measurement of the partial pressures at ambient total pressure by means of optical spectroscopy (IR and UV-visible); and 3) measurement of the partial pressures up to 50 bar. Work this year focused on the design of experimental hardware and identification of diagnostics for these measurements. Based on the first experiments, we have determined that UV-visible fits the requirements for iodine measurements and that Fourier-Transform IR is a suitable method for H_2O and HI. Measurement of the partial pressures up to 50 bar is difficult because it requires control of a high-temperature and pressure mixture. Spectroscopic IR and UV-visible methods suitable for the actual process conditions (which may induce prohibitive absorption) are needed. Two Raman diffusion methods that are promising for highly concentrated media are being investigated for these experiments, which are planned for the second half of 2004. Selection of diagnostic hardware is nearly complete and the experiments are scheduled to begin in early 2004. The development of the initial $HI/I_2/H_2O$ Vapor Liquid Equilibrium (VLE) model has also been completed and it will be used to support the experimental program.

Task 2 – Bunsen (Prime) Reaction Section (CEA)

The Bunsen reaction (Figure 2), where SO_2 and I_2 are added to water to produce H_2SO_4 and HI, operates with excess water and also with excess iodine to allow separation of the H_2SO_4 and HI. The amount of water is important because it must be removed and recycled to the Bunsen reactor before the decomposition steps, which are energy-intensive operations. Therefore, approaches must be investigated that reduce the amount of recycle water. Parametric tests have been initiated to identify a better stoichiometry of this reaction. Preliminary experiments were started in March 2003. These experiments consisted of separation of the two liquid phases, under ambient conditions, formed by the mixing of final products of the reaction: H_2SO_4, H_2O, HI, and I_2, and concentration measurements (iodine in the H_2SO_4 phase and sulphur in HI_x phase). Diagnostic techniques were developed to permit quantitative determination of H^+ ions, sulphur, and iodine in each phase. These measurements were consistent with JAERI results for these conditions.

A preliminary design of the laboratory-scale Bunsen section has been completed. The design is sized to produce 100 L/hr of hydrogen and permits autonomous operation with reprocessing of the iodine as necessary. This work is essential to design the laboratory loop and to identify the improvement margins. This is a modification of the initial GA initial flowsheet in that Bunsen and boost reactors are assembled in one gas-liquid exchange column, and the SO_2 and O_2 are separated by selective condensation of SO_2. This design will be the basis for defining interfaces between the three parts of the loop and the subsequent detailed design of the Bunsen section.

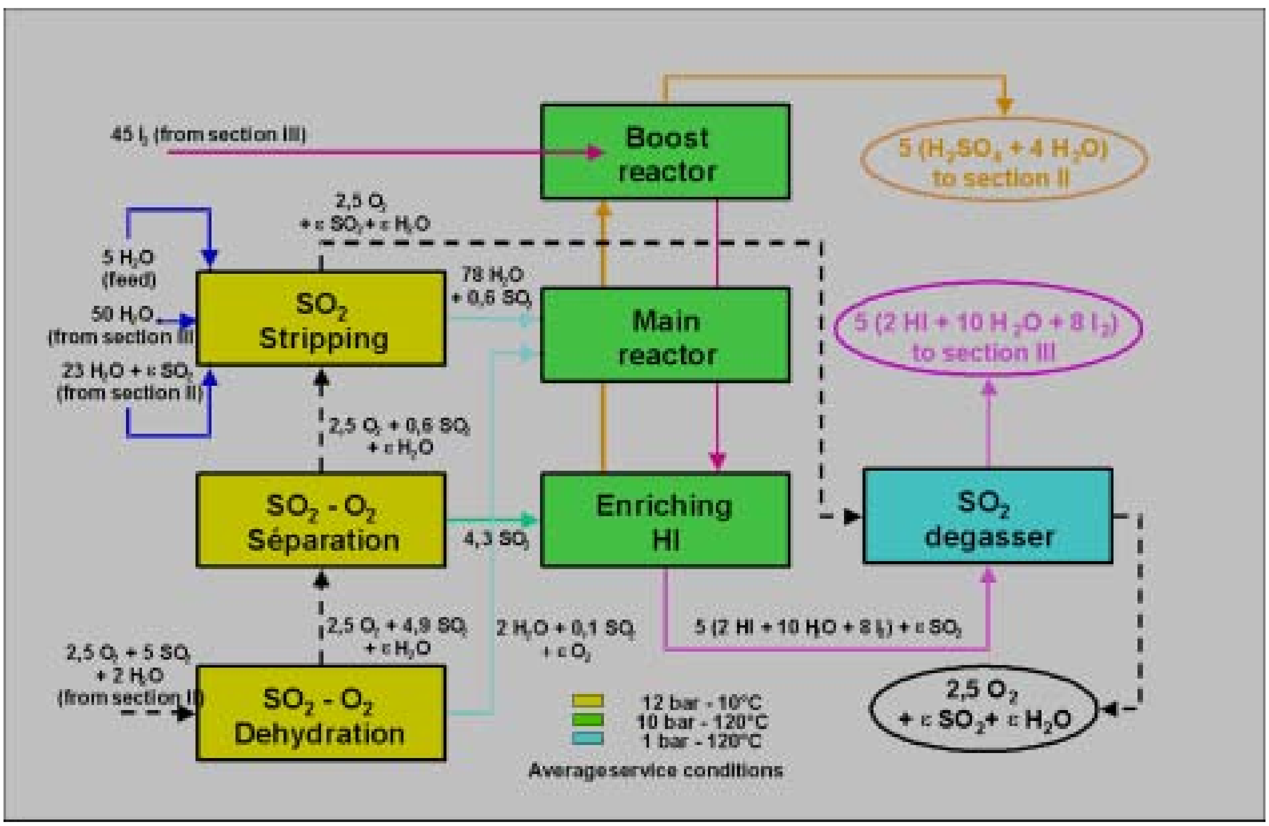


***Figure 2**. Bunsen reaction.*

Task 3 – HI Decomposition Section (GA)

The challenge of this section is how to efficiently and economically separate the HI from the HI, I_2, H_2O chemical complex, in which the HI is accompanied by a great deal of water and iodine. Initially, the focus will be on reactive distillation as the method of processing the HI_x mixture. During the first year, work focused on the analyses and design of the reactive distillation column for HI decomposition, which involves the reactive still, a hydrogen scrubber, and the iodine wash column. Electric heat input will drive the process using cooling water or air to remove heat where required. The demonstration will be performed under pressure and temperature conditions prototypic of full-scale process conditions. The three columns will be constructed from borosilicate glass (Pyrex®) and located inside a single pressure vessel. Because the focus of these early experiments is the evaluation of reactive distillation as an option for HI decomposition, quartz or Pyrex construction is the most cost-effective approach to develop the required information. The reactive still design is key to the efficiency of the reactive distillation method. A design has been developed using a borosilicate glass sieve tray column. Flow of feed into the column, and subsequent liquid products out, is controlled by magnetic valves similar to those used for reflux splitting in glass distillation columns. To minimize the amount of iodine and hydrogen iodide required to perform a test, the bottoms and side product will be recycled to the feed vessel.

Task 4 – Sulfuric Acid Section (SNL)

The purpose of the sulfuric acid boiler section is to receive the recirculated flow of aqueous, impure sulfuric acid, strip it of as much water as possible, vaporize the sulfuric acid, and drive the decomposition to form SO_2. Efficient design of this section is essential because of the significant energy consumption associated with the vaporization of sulfuric acid (and water) and the decomposition of the sulfuric acid. FY 2002 activities focused on evaluating alternative designs for the boiling and decomposition sections and identifying candidate materials for boiling and superheating sulfuric acid to define a preliminary design for the boiler and decomposer sections.

Materials for boiling sulfuric acid in the 400°–500°C range is the most challenging issue for the sulfuric acid section. Materials compatibility requirements depend on the temperature of the acid concentrations regime, and different materials will be needed in the various stages of concentration, boiling, and decomposition. Table 1 summarizes the candidate materials identified in the literature for these highly oxidizing conditions.

A dry-wall boiler design using SiC or Si_3N_4 as the boiling interface was identified as a promising approach to mitigating materials requirements in this regime. This approach allows different materials to be used for pressure boundaries and the corrosive boiling interface. To support nominal system operation (100 liters of hydrogen per hour), a sulfuric acid flowrate of approximately 250 mL/hr is required, requiring a heat input of approximately 602 kJ/hr (240 W) to vaporize the sulfuric acid and 1270 kJ/hr (350 W) to decompose the acid. Initial system testing will be carried out based on once-through operation. The SO_2 formed will be monitored by in situ optical or electrochemical methods.

Task 5 – Flowsheet Analysis (GA and CEA)

To develop consistent section designs and interface specifications for the next phase of the S-I project, a consensus flowsheet is needed for the overall process. Because the thermodynamic database is not complete in all areas, the analyses require developing a consistent

Table 1. *Structural Material Candidates for Sulfuric Acid Section*

Process Regime	Conditions(Temp., Conc.)	Candidate Materials
H_2SO_4 Concentration	300 – 450K, <50% 50-75% 450-700K 75-95%	Glass-lined steel, plastics, ceramics Hastelloy B-2, C-276 Incoloy 800H, AL610, high Si steel, Au or Pt plating
H_2SO_4 Vaporization (H_2O, SO_3)	600 – 800K, 95%	Structural: Incoloy 800H, AL610, high Si steel, SiC, Si_3N_4, Hastelloy G, C-276
H_2SO_4 Decomposition	800 – 1200K, 95%	Structural: Incoloy 800 H, HT (aluminide coatings), AL 610, ceramics, Pt, or Au coatings

approach and assumptions that can be refined as more information becomes available. This process has been initiated with a review of the full system flowsheet developed by GA (ASPEN) and independent analyses performed by CEA using PROSIM. The most important differences in these analyses are in the HI reactive distillation section. Comparative analysis of extractive distillation will also be developed before designs for this section are finalized.

Planned Activities

The objective of the second year of the S-I I-NERI project will focus on construsting the component reaction sections, completing vapor-equilibrium measurements, and refining of the flowsheet analyses to support the design and testing activities. Task 1 will focus on completing measurements of the total pressure of $HI/I_2/H_2O$ up to 50 bar and 300°C and partial pressures at ambient total pressure. Measurement of the partial pressures up to 50 bar will be initiated later in 2004. Work on the Bunsen reaction section (Task 2) will focus on completing tests on the initial reactor (B1) and development of a thermodynamic model based on those results. Tests with initial reactants (B2 reactor) will be initiated to see if the amount of water and iodine can be reduced. Design and construction of the final section configuration (B3 reactor) will be completed. Task 3 will complete testing of the glassware version of the reactive distillation column and construct the final HI section hardware. Task 4 will construct the initial version of the sulfuric acid boiler decomposer and complete testing to support the construction of the sulfuric acid section. Refinement of the S-I flowsheet will continue during this next year as new data are developed.

Appendix B

U.S./Republic of Korea Collaboration Project Summaries/Abstracts

International Nuclear Energy Research Initiative

Project #	Title
2002-008-K	Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in ALWRs
2002-010-K	The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena
2002-016-K	Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors
2002-020-K	Development of Enhanced Reactor Operation Strategy Through Improved Sensing and Control at Nuclear Power Plants
2002-021-K	Condition Monitoring Through Advanced Sensor and Computational Technology
2002-022-K	In-Vessel Retention (I & II)
2003-002-K	Passive Safety Optimization in Liquid Sodium-Cooled Reactors
2003-008-K	Developing and Evaluating Candidate Materials for Generation IV Supercritical Water Reactors
2003-013-K	Development of Safety Analysis Codes and Experimental Validation for a Very-High-Temperature Gas-Cooled Reactor
2003-020-K	Advanced Corrosion-Resistant Zirconium Alloys for High Burnup and Generation IV Applications
2003-024-K	Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte

International Nuclear Energy Research Initiative

Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in ALWRs

Principal Investigators (U.S.): M. Anderson, University of Wisconsin; M. Corradini, University of Wisconsin

Principal Investigator (International): K.Y. Bang, Korean Maritime University (KMU)

Collaborators: R. Bonazza, UW-Madison; D. Cho, Argonne National Laboratory

Project Number: 2002-008-K

Project Start Date: January 2002

Project End Date: September 2005

Reporting Period: October 2002 — September 2003

The interaction and mixing of high-temperature melt and water is the important technical issue in the safety assessment of water-cooled reactors to achieve ultimate core coolability. For specific advanced light water reactor (ALWR) designs, deliberate mixing of the core-melt and water is being considered as a mitigative measure to ensure ex-vessel core coolability. The goal of this work is to provide the fundamental understanding needed for melt-water interfacial transport phenomena, thus enabling the development of innovative safety technologies for advanced LWRs that will ensure ex-vessel core coolability. The work considers the ex-vessel coolability phenomena in two stages. The first stage is the melt quenching process and is being addressed by Argonne National Lab (ANL) and University of Wisconsin (UW) in modified test facilities. Given a quenched melt in the form of solidified debris, the second stage is to characterize the long-term debris cooling process and is being addressed by Korean Maritime University (KMU), via tests and analyses.

Research Objective

Research objectives include the following:

Task 1: Measure the cool-down behavior of the melt-water mixing zone by thermal mapping of this multiphase, multicomponent system (ANL lead).

Task 2: Measure the flow regime and interfacial area behavior of the melt-water multiphase, multicomponent mixture by the use of innovative, real-time, X-ray imaging (UW lead).

Task 3: Develop and integrate analytical models of interfacial transport phenomena in a model, including separate effects experimental studies (Korean researchers at the Korea Maritime University lead).

Task 4: Integrate the various models for long-term debris coolability into an overall approach (Korean researchers at the Korea Maritime University lead).

Task 5: Develop an approach for applying this fundamental knowledge to the development of a novel safety concept of ex-vessel core debris coolability (all participants).

Research Progress

Task 1: Transient Thermal Mapping of the Mixing Zone (ANL Lead)

The results of the scoping tests reported in the first year indicated that the volumetric cooling rate of melt might not increase in proportion to the increase in the water injection rate, particularly for larger injection rates. To investigate the effect of the water injection rate in a systematic manner, a series of tests was conducted during this year as a function of water injection rate. In addition, these tests provided information on the effect of noncondensable gas mixed with the injected water on the overall heat transfer rate. The possible effect of noncondensable gas on the melt quenching behavior is of prime interest for two reasons. First, noncondensable gases would be generated upon the contact of core melt with the concrete basement in the reactor cavity. Second, the presence of noncondensable gas in the injected water would likely eliminate flow oscillations observed in COMET tests and reduce the risk of an energetic fuel-coolant interaction, which must be avoided if deliberate mixing of the core melt and water is to be a successful mitigative measure leading to core coolability. Thus, the working hypothesis was to quantify the molten pool quench process with a co-injection of water and a gas. The experimental apparatus modified for these tests consists of a test section and associated components (e.g., water supply tank, condenser, water collection tank, and melt vessel). The experimental results of the tests are summarized in Table 1. In all tests, the melt was lead-bismuth eutectic alloy (melting point = 125°C) and the collapsed melt height above the injector was 32 cm, corresponding to a melt mass of 23 kg. The injected gas was argon.

Experiment	INERI TEST CONDITIONS					
	2	3	4.1	4.2	4.3	4.4
P_{system} [psia]	14.7	14.7	14.7	14.7	14.7	14.7
$\dot{m}_{water}$ [ml/min]	40	20	40	60	80	100
$T_{water,\ inlet}$ [°C]	60	60	60	70	70	70
$\dot{m}_{argon}$ [slpm]	8	0.0	10	10	10	10

Experiment	INERI TEST CONDITIONS				
	3	4.1	4.2	4.3	4.4
Heat Removal Rate [MW/m³]	0.4	0.78	1.16	1.57	1.89
Volumetric HTC* [kW/m³-K]	2.2	4.0	6.2	8.4	10.6

* Based on the collapsed liquid metal height.

The primary variable in the experiments was the water injection rate at a fixed gas flow rate to determine whether the gas flow impedes the quenching behavior of the water injected into the simulant melt pool. The preliminary data seem to indicate that injecting argon assisted in suppressing vapor explosions but did not noticeably affect the overall heat transfer coefficient. The local heat transfer coefficient results are to be examined in more detail in the UW 2-D experiments as the overall thermal mapping continues in the ANL tests.

Task 2: Real-Time X-ray Imaging of Multiphase Structure (UW Lead)

This task is to measure the volumetric and the associated local heat transfer phenomena (heat transfer coefficient and associated flow stability) for the quenching phase of melt coolability at a range of water mass fluxes (0.1-1 kg/m^2-s) and prototypic containment pressures (1-5 bar). To successfully quench a molten pool with water injection in a reliable fashion, the flow stability of the injected water stream into the molten pool should be preserved. ANL/UW researchers have found that injecting a noncondensable gas together with the water can stabilize the flow. The gas injected together with the water may act to stabilize the heat transfer process, reducing the vapor production rate below the vapor expansion rate. This suggests an optimization problem to solve—injection of enough gas to stabilize the flow with a minimum decrease in heat transfer coefficient. These local measurements should identify this regime. In concert with the ANL work, the UW 2-D experimental apparatus was modified to allow for co-injection of water and gas to test this hypothesis

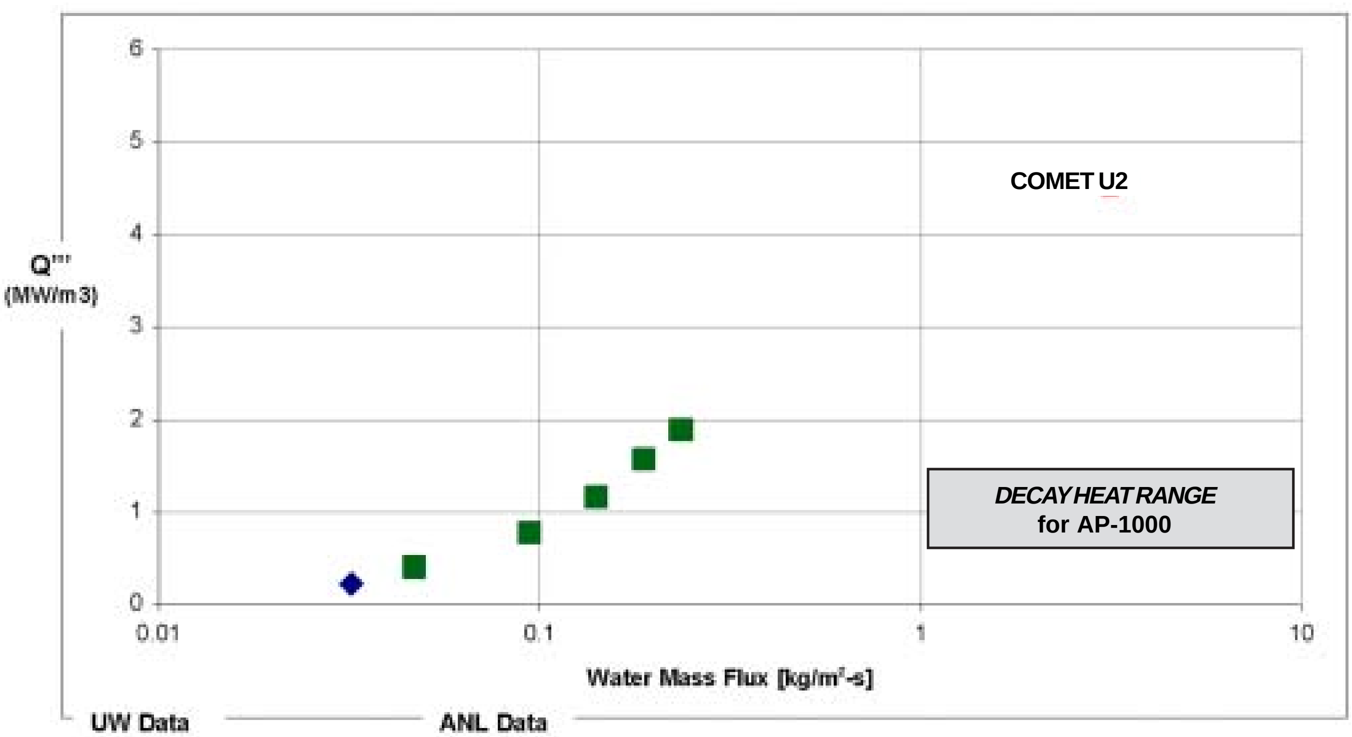


Figure 1. *Volumetric Heat Removal for Various Water Mass Fluxes.*

under a range of flow and pressure conditions beyond what may be achieved at the ANL facility, with dynamic X-ray imaging to determine the local heat transfer coefficients coincident with measuring pool volumetric heat transfer.

The initial test campaign involved water injection (up to 1 gm/s) into a 2-D test section (18 cm x 10 cm in cross-section, 70 cm high) with co-injection of argon gas through a single nozzle with a 2 mm diameter. The dynamic X-ray imaging was used in these tests to determine that the mixture was well-mixed and to measure the level swell height. The overall volumetric heat removal rate was determined and found to be unaffected by a variation in the ambient pressure (1-5 bar) as well as the gas co-injection rate (0-30 slpm). The completed set of current results are presented below and compared to the single prototypic COMET-U2 test and the expected range of decay heat power that must be removed by the quench process in a typical LWR design (AP1000). One first notes that only modest mass fluxes of water are needed to quench the melt; and second, one notes that the COMET-U2 tests may indicate a limit to quench rate as first suspected. This will be confirmed as quenching experiments are completed in the next year.

Task 3: Modeling of Interfacial Transport Phenomena: Separate Effect Tests (KMU lead)

The focus of this task is to conduct separate effect tests for key physical models. These include 1) visualization of melt-coolant mixing using transparent stimulant material, 2) film boiling on spheres in high temperature, 3) two-phase heat transfer in porous media, and 4) development of simultaneous measurement technique for temperature and flow field.

A high-temperature film boiling experiment has been carried out up to 2000 K of sphere temperature. Direct measurement of heat transfer coefficient is not possible due to the difficulty of instrumentation for high temperature, but the heat transfer coefficient for saturated film boiling can be estimated by estimating the steam bubble volume and release period in the visual images. The experimental parameters are: sphere diameters of 19 mm and 12.7 mm, water temperature of 50~100°C, and sphere temperature of 500~1700°C. The saturated flow film boiling heat transfer coefficient ranges 70~200 W/m^2K for the sphere temperature 750~1950 K and it increases slightly as the sphere temperature increases. The existing heat transfer correlations for saturated flow film boiling overestimate the present data. A new correlation has been proposed and this correlation can be extended to the temperature range beyond 2000 K.

Film boiling of spheres in porous environment has been experimentally studied and compared with the dryout heat flux data for top flooding. The film boiling heat transfer coefficients were measured for water temperature 80~100^oC, sphere temperature 500~700^oC. The porosity was 0.45. It is observed that the effect of water subcooling is small for porous medium case because the instrumented sphere actually experiences heated water below even in case of subcooled water initially. Using the measured heat transfer coefficients, the volumetric heat transfer coefficient can be calculated for saturated water, and it ranges 50~150 kW/m^3K. Also, the heat removal capacity per unit volume in this porous medium was calculated and compared with the dryout heat flux data for porous medium of 3 mm diameter spheres reported by Barleon et al. (1984). The present data are for 22.2-mm-diameter porous medium, and it is higher than the dryout heat flux compared. But the cooling rate in film boiling can be lower than the dryout heat flux if the same diameter spheres are used—this will be conducted next year.

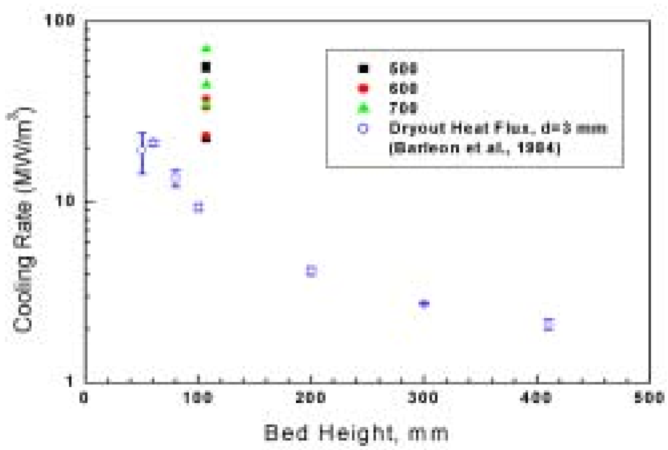


Figure 2*. Volumetric Heat Removal Rate as a Function of Debris Bed Height*

Task 4: Integration of Analytical Models (KMU lead)

The objective in the first year was to check the adaptability of commercial code CFX4.4 to the melt coolability problem. The results showed positive sign in some numerical calculations but revealed certain limits in handling three or more phases. The fatal problem is that it cannot deal the interphase transfer between two continuous media. Only the continuous-disperse pair is possible in calculating interphase transfer. The only way to make it possible for continuous-continuous pair is to assume one fluid dispersed with a certain diameter, which may cause large error to result in solution failure. The CFX5.6, updated version of 4.4, adopts a coupled algorithm and possibly saves computing time by faster convergence. In addition to this, a solution for mixture model becomes possible by user input subroutines. The model for drag and heat transfer coefficient must be set by the users. These models will be established by further study. In this fiscal year, some sample problems were set up for computational work. Although only the results for a sample computational domain that is similar to but not exactly the same as UW experimental setup are shown in this report. The geometry used for computation is a cylinder 60 cm high with a 10-cm diameter. The whole container is assumed to be the computational domain. The lead melt of 690 K in temperature is filled up to 30 cm in height. The 2-mm-diameter inlet is located at the center of the bottom face. Water injection rate is 10 g/s. Extensive analyses were not carried out at this point due to the time—consuming nature of this kind of work. Other geometry and melt-coolant pairs were used for comparison. One of these trials is the R22-water pair, which is used for visualization purpose in KMU. More computational work for accurate results will be performed next year after the research team sets up appropriate transfer models to be adapted to the CFX5.6 code. Some preliminary results are given in the planned activities section of this report.

Planned Activities

During FY 2004, the research team expects to complete the quenching experiments and associated analysis, and produce an approach to predict the molten pool quench process (Tasks 1 & 2). In addition, the long-term debris cooling experiments will be completed (Task 3) and the resulting data incorporated into an integrated modeling approach (Task 4). Finally, the whole research team will address the overall integration into a complete coolability design.

International Nuclear Energy Research Initiative

The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena

Principal Investigator (U.S.): D.P. Weber, Argonne National Laboratory

Principal Investigator (Int.): H.G. Joo, KAERI

Collaborators: Purdue University; Seoul National University

Project Number: 2002-010-K

Project Start Date: December 2001

Project End Date: September 2004

Reporting Period: January — October 2003

Research Objectives

A comprehensive, high-fidelity reactor core modeling capability is being developed for detailed analysis of current and advanced reactor designs. The work involves the coupling of advanced numerical models such as computational fluid dynamics (CFD) for thermal hydraulic calculations, whole-core discrete integral transport for neutronics calculations, and thermo-mechanical techniques for structural calculations. The product code has been designed to run on parallel, high-performance computers. This integrated simulation capability will provide a verifiable computational tool to perform intensive studies on the operational and safety characteristics of various design alternatives and to compare the results obtained with presently available tools to those from this high-fidelity capability.

In each of the phenomenological areas, including reactor physics, thermal-hydraulics, and thermo-mechanics, the objective was to develop and demonstrate the ability to accurately calculate the individual key phenomena. The integration of these high-fidelity models into a robust computational tool, along with verification and validation of the integrated capability, provides the desired tool for advanced reactor design. In each of the three key phenomenological areas, examination of numerical performance and verification/validation are being performed. For the coupled code, strategies and numerical performance have been investigated and verification of the code against integrated benchmarks is initiated.

Research Progress

The reactor physics module, being developed at KAERI, is a whole-core transport code, DeCART (Deterministic Core Analysis based on Ray Tracing), based on the method of characteristics. This code generates sub-pin level power distributions by representing local heterogeneity explicitly without homogenization, using a multi-group, cross-section library directly without group condensation and incorporating pin-wise thermal hydraulic feedback. As part of the efforts to verify this module, a Monte Carlo computational scheme with pin-by-pin thermal hydraulic feedback capability has also been developed by partners at Seoul National University. Comparisons between conventional methods on benchmark problems, fuel assemblies, and whole core calculations indicate excellent performance in terms of accuracy and computational time. An example of DeCART calculations for the VENUS-2 benchmark problem is shown in Figure 1, where predictions of k_{inf} show excellent agreement. Similarly good agreements have been observed for pin power distributions.

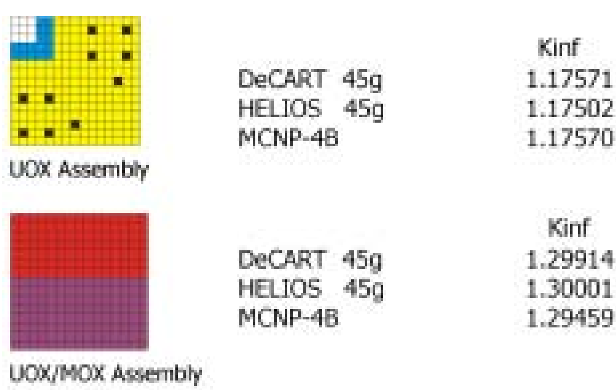


***Figure 1**. Benchmark of DeCART using VENUS2 assembly calculation.*

Thermal hydraulic analysis has been focused on the use of high-fidelity CFD capabilities available in commercial CFD software, with specific focus being applied to the STAR-CD and CFX codes. While it is recognized that these CFD codes have the theoretical capability of simulating events with fine detail in a reactor application, a demonstration of their ability to predict observed flows in rod bundle geometry was considered critical for their ultimate inclusion in the integrated code system. It is believed that such codes have a sufficient range of turbulence and heat transfer models that can be applied successfully to both current reactors and advanced reactors, such as those being considered in the Generation IV program. Nonetheless, proof of this capability is a key project objective, with initial focus on water- and gas-cooled systems.

Project efforts at ANL and KAERI in thermal-hydraulics have thus focused on experimental validation of these codes, with particular emphasis on demonstrating their ability to predict turbulent flow and associated heat transfer in rod bundle flows. Figure 2 illustrates the ability of various turbulence models to predict detailed flow characteristics including sub channel mean and turbulence flow. Although completely accurate prediction of turbulent structures in the subchannel is yet to be demonstrated, prediction of heat transfer coefficients, considered critical for these applications, was quite good for forced flow.

The third phenomenological element of the integrated code relates to modeling of thermo-mechanical response. For conventional reactor applications, coupling of the thermo-mechanical and thermal hydraulics analyses can be used to assess, for example, the effects of rod bowing on flow and heat transfer, which may affect assessments of departure from nucleate boiling. In advanced reactors, where inherent safety characteristics may depend on structural and neutronic response to thermal transients, the ability to closely couple the three phenomena may lead to clearer demonstration of inherent safety or a reduction in overly conservative safety margins. The project is using a three-dimensional finite element code, NEPTUNE, developed at Argonne, to simulate the response of reactor components to design basis and beyond the design basis loads. The element formulations can properly treat large deformations and the rate-type material models can handle large material strains.

The ultimate objective of this effort is to produce an integrated analysis capability utilizing the multi-physics models, having demonstrated the validity of the individual components. General purpose coupling schemes

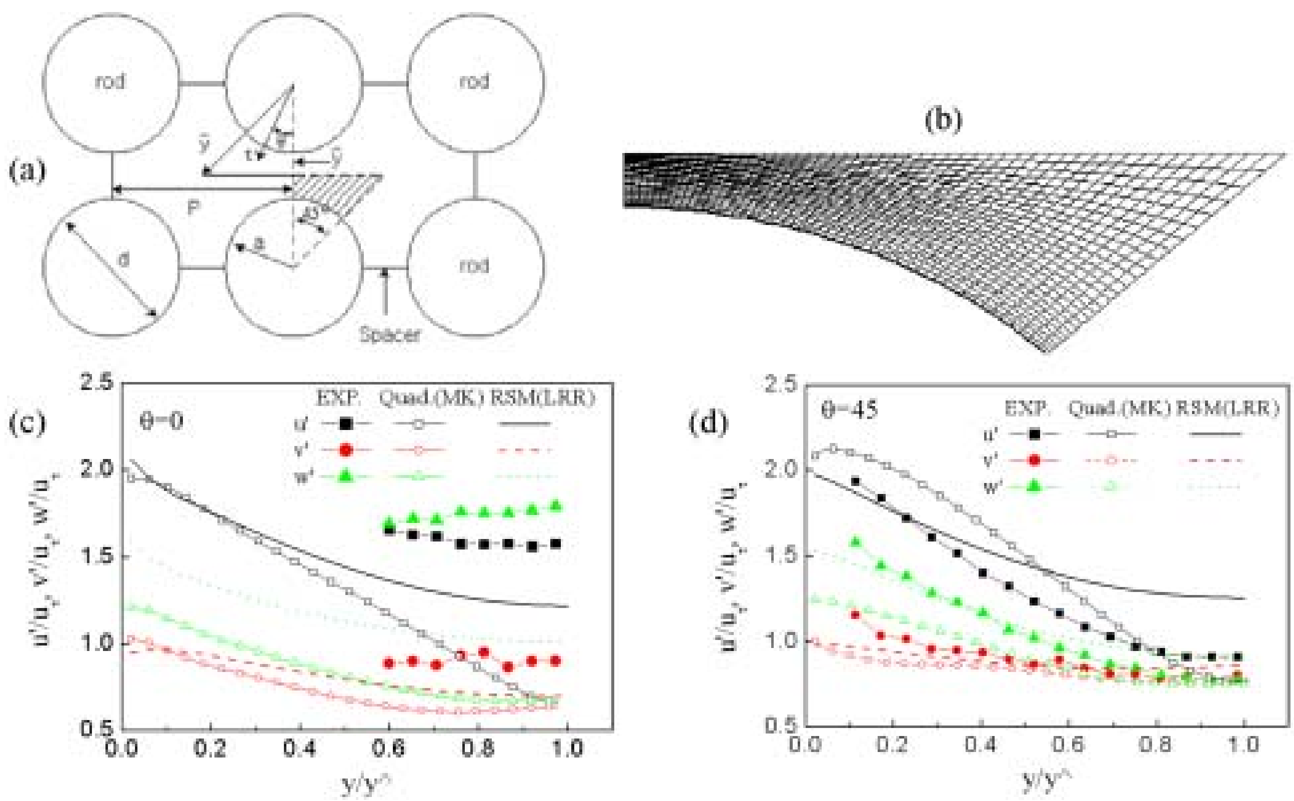


Figure 2. *(a) Schematic of the square rod bundle, (b) computational grid of an octant subchannel, (c) comparisons of turbulence intensities at the gap, (d) comparisons along the diagonal.*

have been developed as illustrated in Figure 3 and inter-compared for quality assurance. Strategies for exchange of relevant information among the modules have been investigated and demonstrated the importance of proper mapping between the vastly different neutronics and thermal hydraulics grids. From the convergence standpoint, selection of optimal timing for exchange of information among models is currently being examined. Finally, advanced coupling strategies based on matrix-free Newton-Krylov methods to accelerate the coupled field problem have been investigated at Purdue University.

Demonstration of the integrated analysis capability has been initiated for a series of sample problems. The initial test problem was a simple 3-by-3 pin system containing UO_2 and MOX pins and a central guide tube. Results illustrated expected temperature differences between the fuel types and asymmetric heating and cooling. Test results have also been performed for a "mini-core" problem, representing four quadrants of adjacent assemblies of different types. Results such as those indicated in Figure 4 show the calculation of the expected asymmetry in the power and temperature distributions. Detailed examination illustrated the important effect of properly treating the temperature distribution within the fuel pin. Computational results on the JAZZ multiprocessor Linux cluster at Argonne indicate that extrapolation of such calculation to whole-core application can be accomplished with current generation, high-performance computer systems. Preliminary estimates indicated that steady state calculation of prototypic PWR cores can be performed on a teraflop class machine in less than a day. Extrapolation of these estimates suggest that transient calculations for relevant scenarios can also be accomplished in computing times of several tens of hours on such machines. Scalability of results to date also indicates that expected availability of more powerful machines will result in proportional reduction in computing time, with the expectation that such whole-core, high-fidelity calculations will soon be possible in times measured in hours, rather than days or week.

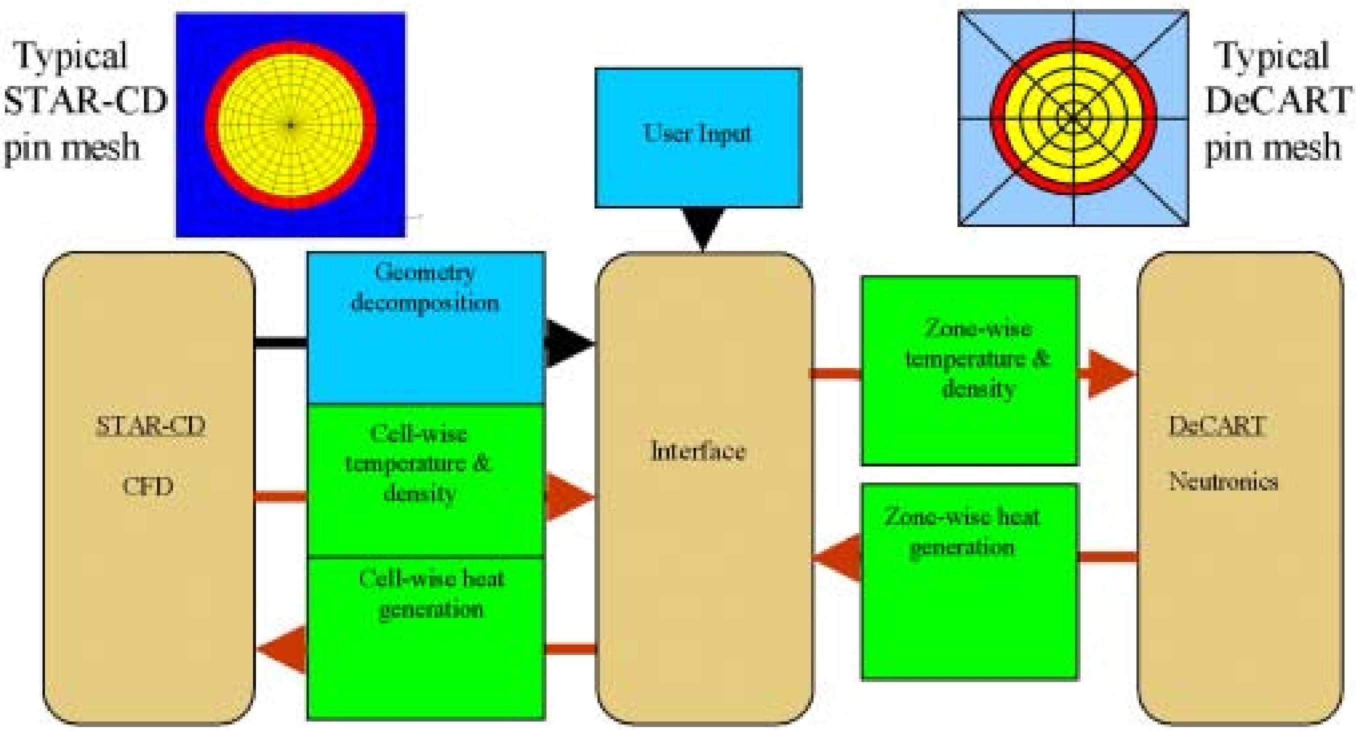


***Figure 3**. Coupling Mechanics: Interface Design*

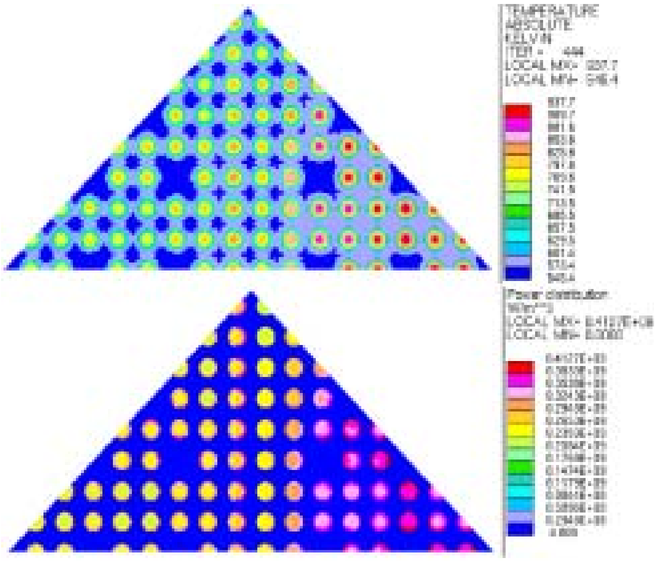


***Figure 4**. Mini-Core Model Results*

Planned Activities

Future activities in this project are focused on implementation of transient neutronics capabilities, continued validation of CFD models, integration of CFD with thermo-mechanics, and performance of integrated calculations. The integrated calculations will focus on improvement to the numerical iteration strategy for the multi-physics problem to reduce the overall computing time for steady-state and transient problems. In addition, verification of the integrated capability against standard benchmark problems will be assessed. Such analyses will not only confirm the ability of this analysis system to correctly calculate the integral measures being compared, but also demonstrate the ability to predict important detailed information at the sub-pin and sub-channel level that may be important in assessing safety characteristics or increasing performance, without reducing true safety margins.

International Nuclear Energy Research Initiative

Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors

Principal Investigator (U.S.): D.M. McEligot, Idaho National Engineering and Environmental Laboratory (INEEL)

Principal Investigator (Int.): J.Y. Yoo, Seoul National University (SNU)

Collaborators: Iowa State University; Pennsylvania State University; University of Maryland; University of Manchester; Korea Advanced Institute of Science and Technology (KAIST)

Project Number: 2002-016-K

Project Start Date: December 2001

Project End Date: September 2004

Reporting Period: January 1 — November 30, 2003

Research Objective

The ultimate goal of this collaboration of coupled fundamental computational and experimental studies is the improvement of predictive methods for Generation IV reactor systems (e.g., supercritical-pressure water reactors) and associated Advanced Fuel Cycle Initiative (AFCI) and Nuclear Hydrogen Initiative (NHI) activities. The general objective is to develop the supporting knowledge needed of advanced computational techniques for the technology development of the concepts and their passive safety systems. The resulting specific objectives are to develop and to extend direct numerical simulation (DNS), large eddy simulation (LES), and differential second moment closure (DSM) techniques to treat supercritical property variation and complex geometries, thereby providing capabilities to do the following:

- ♦ Assess predictive capabilities of current codes for supercritical water-cooled reactors (SCWRs), very high temperature gas-cooled reactors (VHTRs), etc.
- ♦ Provide bases to improve nuclear reactor thermal hydraulics (NuReTH) safety and subchannel codes
- ♦ Provide computational capabilities where current codes and correlations are inadequate
- ♦ Give predictions for Generation IV conceptual and preliminary designs for
  - full power operation (LES, DSM, and Reynolds-averaged Navier-Stokes approaches)
  - reduced power operation
  - transient safety scenarios
- ♦ Ultimately, handle detailed NuReTH flow problems for final Generation IV designs for improved performance, efficiency, reliability, enhanced safety, and reduced costs and waste.

This project will provide basic thermal fluid science knowledge to develop increased understanding for the behavior of superheated and supercritical systems at high temperatures, apply and improve modern computation and modeling methods, and incorporate enhanced safety features.

Research Progress

This basic thermal fluids research applies first principles approaches DNS and LES) coupled with experimentation (heat transfer and fluid mechanics measurements). Turbulence is one of the most important unresolved problems in engineering and science, particularly for the complex geometries and fluid property variation occurring in these advanced reactor systems and their passive safety systems. DNS, LES, and differential second moment closures (DSM or Reynolds-stress models) are advanced computational concepts in turbulence “modeling” whose development is being extended to treat complex geometries and severe property variation for designs and safety analyses of Generation IV reactor systems such as SCWRs.

Variations of fluid properties along and across heated flows are important in SCWRs, VHTRs and gas-cooled fast-spectrum reactors (GFRs), all Generation IV reactor systems concepts. Significant differences and uncertainties have been found between thermal hydraulic correlations for these conditions. Improved computational techniques and supporting measurements are needed to assist the developers of codes for reactor design and systems safety analyses to treat the property variations and their effects reliably for some operating conditions and hypothesized accident scenarios of these reactors. The geometries of the reactor cooling channels of some SCWR concepts are demonstrated in Figure 1. Most of these geometries are more complex than those that have been used to generate the empirical correlations employed in the thermal hydraulic codes. Advanced computational techniques may be applied, but measurements with realistic geometries are needed to assess the reliability and accuracy of their predictions.

Iowa State is extending LES to generic idealizations of such geometries with property variation; SNU supports these studies with DNS. KAIST is developing DSM models and will evaluate the suitability of other proposed RANS (Reynolds-averaged Navier-Stokes) models by application of the DNS, LES, and experimental results. INEEL will obtain fundamental turbulence and velocity data for generic idealizations of the complex geometries of these advanced reactor systems.
U. Maryland is developing miniaturized, multi-sensor probes to measure turbulence components in supercritical flows. SNU is developing experiments on the effects of property variation on turbulence structure in superheated and supercritical flows. Pennsylvania State and U. Manchester provide industrial insight and thermal-hydraulic data needs and review the results of the studies for application to realistic designs and their predictive safety and design codes.

DNS employs no turbulence modeling; it solves the unsteady governing equations directly. Consequently, along with measurements, it can serve as a benchmark for assessing the capabilities of LES, DSM, and general RANS techniques. It also can be applied for predictions of heat transfer at low flow rates in reduced power operations and transient safety scenarios, such as loss-of-coolant or loss-of-flow accidents, in SCWRs, GFRs, and VHTRs. Figure 2 indicates that for SCWRs it can handle sensitive situations that are difficult to treat properly with correlations or with many turbulence models. Once validated, LES and DSM techniques can be applied for predictions at higher flow rates, such as near normal full-power operating conditions, for these Generation IV reactor concepts. The flow facility developed at SNU provides a means of measuring heat transfer to supercritical fluids for assessing of the effects of their property variations. The miniaturized multi-sensor probes from U. Maryland will permit measuring the turbulence that is modeled by the codes. The INEEL experiment models the complex geometry of coolant passages in an SCWR concept to provide benchmark data.

INEEL has developed the world's largest matched-index-of-refraction flow system. By using optical techniques such as laser Doppler velocimetry (LDV), measurements can be obtained in small complex passages without disturbing the flow. The refractive indexes of the fluid and the model are matched so that there is no optical distortion. The large size provides good spatial and temporal resolution. This facility provides the means to investigate the complex flow features of Generation IV reactor geometries.

Following is a summary of research progress during this reporting period:

- Completed seventeen runs for heat transfer to supercritical fluids with the DNS code for a circular tube with conditions spanning the pseudocritical

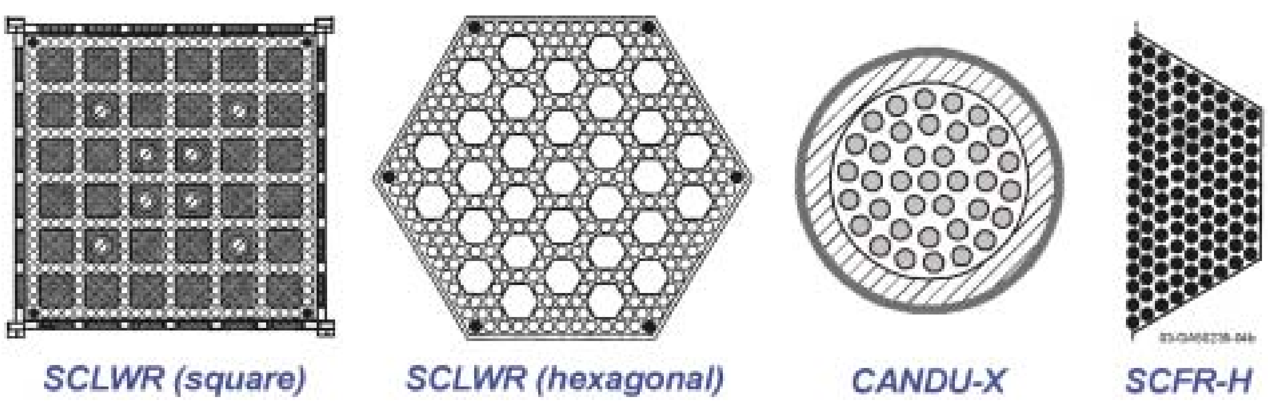


Figure 1. *Some proposed designs for fuel assemblies in some supercritical-pressure water reactor concepts.*

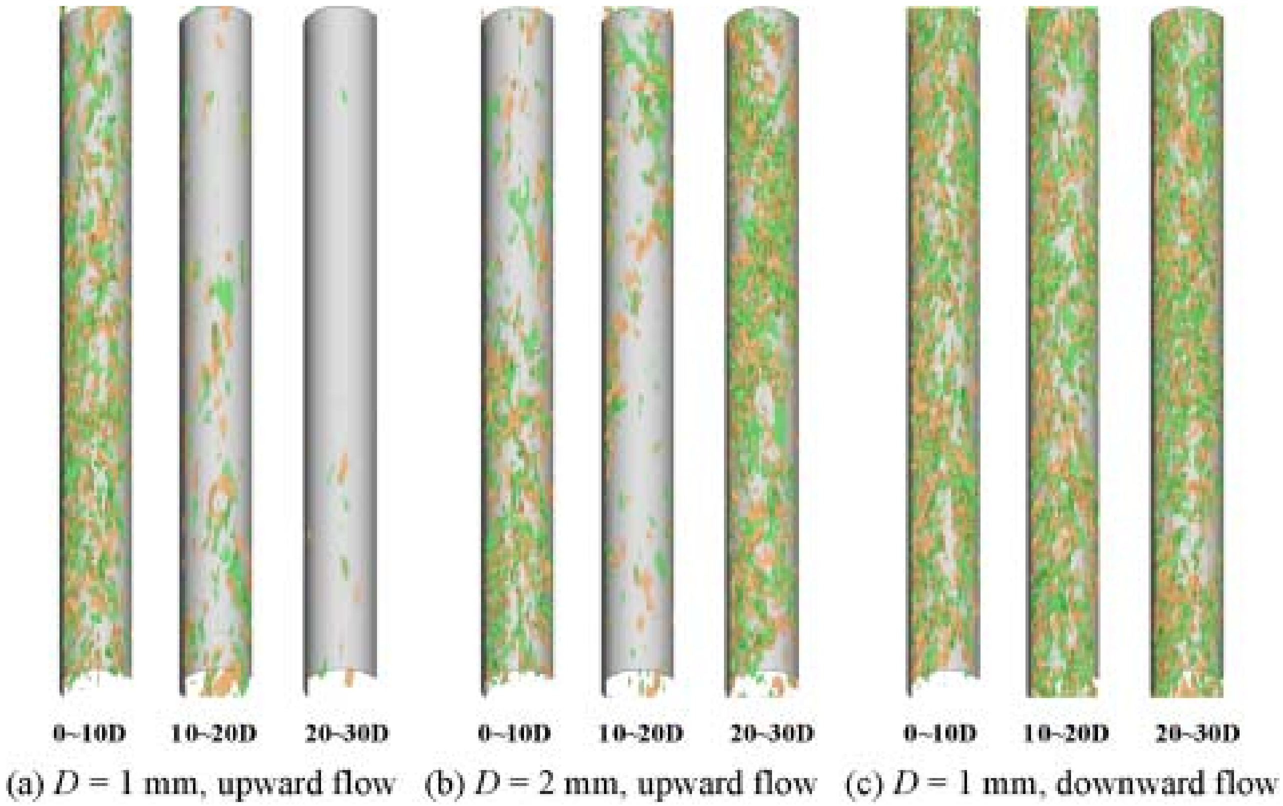


***Figure 2**. Direct numerical simulation of heat transfer to supercritical flow demonstrates sensitivity of turbulence (hence heat transfer) to fluid property variation and buoyancy influences.*

temperature; significant effects of buoyancy and property variation on the turbulence were demonstrated (Figure 2).

- Extended the quasi-developed turbulent LES code to include supercritical fluid properties and validated its performance for adiabatic flow of supercritical CO_2 by comparison to DNS and experiments.

- Applied DSM code with various turbulence models to heat transfer to superheated gas flows and to supercritical flows and compared predictions to experiments; adequacy of predictions varied with the turbulence model chosen and none was universally good.

- INEEL completed fabrication of their large-scale model for simulating flow in SCWR passages (Figure 3), installed it in their matched-index-of-refraction flow system and initiated LDV measurements.

- Successfully tested the response of their miniature multi-sensor probe at elevated temperatures and designed, fabricated, and tested a facility for its calibration in supercritical CO_2.

- Completed fabrication of facility to measure heat transfer to supercritical flows and successfully tested its operation.

- Five topical technical reports were delivered.

Since January 2002, the project partners have had thirteen archival papers published or in press, thirty-four conference presentations, and nine invited presentations relating to this collaborative RoK/U.S. I-NERI project. They also had ninety-six publications and presentations on other topics.

Planned Activities

Planned activities during the third phase are as follows:

- Investigate geometric effects on supercritical flow and heat transfer by simulating flows in complex geometries with DNS code.

- Extend LES code to handle developing flows and complex geometries.

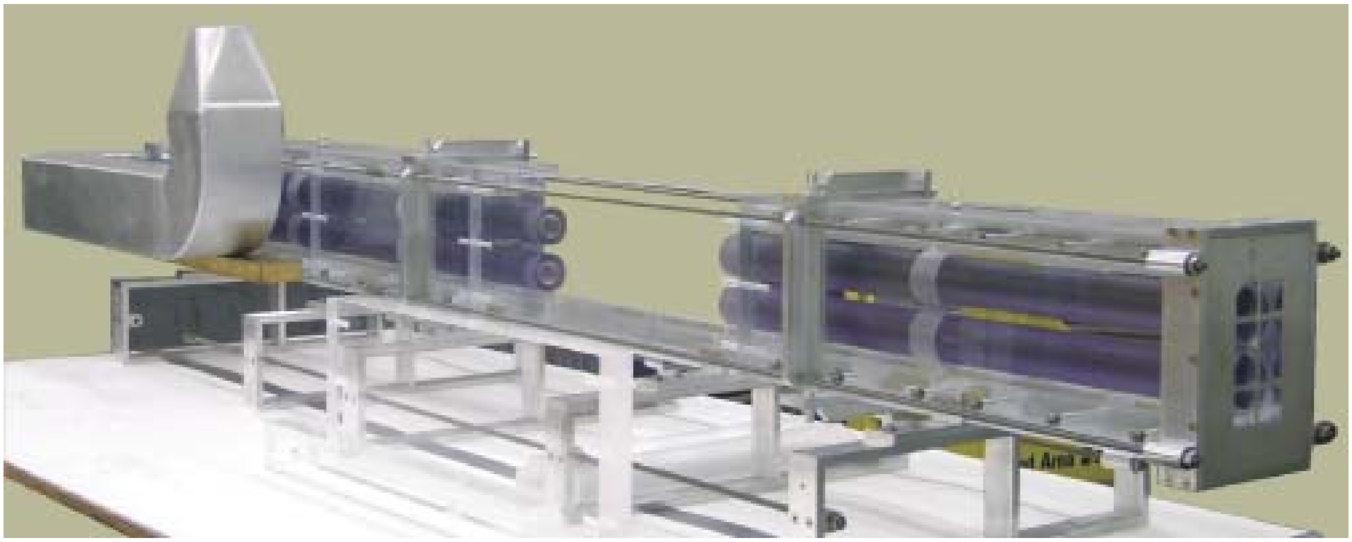


***Figure 3**. Large-scale model to be used in INEEL Matched-Index-of-Refraction flow system for simulation of flow features of conceptual SCWR coolant passages.*

- Complete assessment of eight combinations of explicit algebraic and heat flux models and extend code to three dimensions for complex geometries.
- INEEL will conduct LDV measurements of the velocities and turbulence with their model simulating features of a complex Generation IV SCWR coolant channel with grid spacers.
- Test and calibrate a miniature velocity-temperature probe with supercritical CO_2 for use in the SNU facility.
- Conduct experiments with heated circular and non-circular test sections in their supercritical CO_2 loop.
- Publish related technical papers.

International Nuclear Energy Research Initiative

Development of Enhanced Reactor Operation Strategy Through Improved Sensing and Control at Nuclear Power Plants

Principal Investigator (U.S.): David E. Holcomb, Oak Ridge National Laboratory

Principal Investigator (Korea): Man Gyun Na, Chosun University

Collaborators: Ohio State University; Korea Atomic Energy Reseach Institute; Cheju National University

Project Number: 2002-020-K

Project Start Date: December 2001

Project End Date: September 2004

Reporting Period: October 2002 — September 2003

Research Objectives

The overall project objective is to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants. A more precise knowledge of the reactor system state (e.g., primary coolant temperature, core flux map, primary and feedwater flowrates) can facilitate operation closer to design margins, improve thermal efficiency, and extend fuel burnup. As a result, advanced control models and methods are needed to realize the benefits offered by improved sensing capability.

The project consists of three major tasks. The objectives of the first task are to evaluate the basis for current reactor operation strategies including assessing the state-of-the art for primary system measurement, investigating the effects of measurement limitations on the operational performance of existing nuclear power plants; and identifing potential operational/safety improvements resulting from improved measurement and control. The objective of the second task is to develop and demonstrate three advanced sensors: a solid-state in-core flux monitor (SSFM) applicable to high temperature reactors, a first-principles and thus drift-free temperature measurement system (Johnson noise-based), and the component technologies required to make magnetic flowmeters function on the primary side of pressurized water reactors. The objective of the third task is to take advantage of the benefits of improved sensors by devising advanced reactor operational strategies that optimize core performance and permit reduced operational margins.

Research Progress

Task 1.1 (Assess the State-of-the-Art for Primary System Measurements) and Task 1.2 (Investigate the Effects of Measurements Limitations on Operational Performance of Existing NPPs), except subtask 1.1.2, were completed in Phase 1 of this project. Subtask 1.1.2 (Comparative Analysis of the SSFM, the Johnson Noise Thermometer, and the Magnetic Flowmeter with Measurement Systems of Comparable Physical Variables that are Currently Used in Nuclear Power Plants) was completed by Ohio State University. Task 1.3 (Identification of Potential Operational/Safety Improvements) was continued starting from Phase 1 through the first quarter of Phase 2 by Chosun University. For the most part, this task was completed in Phase 1. The effects on operational margin (DNBR, LPD) of measurement errors and the effects on operational margin of advanced algorithms were studied in Phase 1. Also, possible improvements of core protection algorithms were presented by enhancing existing core protection algorithms and by applying new techniques through improved sensing. In the first quarter of Phase 2, the measurement uncertainty of the feedwater flowrate sensors of venturi meters were reviewed, of which the measurement uncertainty has a large effect on the operational margin. The effect of using in-core neutron sensors was analyzed in Task 3.1, where the in-core sensors are applied to the estimation of the DNBR and LPD.

The underlying rationale for the SSFM project (subtask 2.1) is that it is not currently possible to measure the neutron flux within the core of high-temperature reactors. Both the nuclear hydrogen initiative and the Generation IV program directly depend on high-temperature reactor technology. The high reactor temperature, combined with high flux, prevents using traditional instrumentation directly in-core. The approach envisioned to predict the in-core flux is to measure the flux outside the core, where both the temperature and flux are lower, and use reactor physics models to predict the actual in-core flux profile. However, graphite moderated, high-temperature reactors exhibit high local flux peaking factors near the fuel-moderator boundary. Moreover, much of the safety case for the Generation IV-VHTR rests on the integrity of the fuel coating. The long-term, local temperature and flux are the primary challenges to fuel coatings. Model predictions of the detailed spatial peaking of the neutron flux in the fuel-moderator boundary regions are sufficiently uncertain to prevent high confidence in the coating integrity thereby significantly limiting the core design and reactor operations.

***Figure 1**. Johnson Noise Thermometry Amplifier Boards*

The SSFM has made significant technical progress over the past year. Conceptually, the SSFM is based on employing a polycrystalline aluminum nitride compact as a flux-sensitive resistor. The sensor is intended specifically to operate to at least 1000°C and to operate a power range neutron flux sensor directly within core. ORNL has designed, fabricated, and delivered (to OSU and KAERI) two generations of prototype sensors. The design refinements have allowed using standard semiconductor fabrication technology as the basis for device assembly enabling repeatable, inexpensive devices. KAERI's characterization results are highly positive; the SSFM devices function as intended. While much of the reactor testing results remain very preliminary (and high temperature in-core testing has yet to be performed), the devices respond to neutrons and gammas as predicted.

The central reasoning for the Johnson noise thermometry (JNT) project is that all available versions of temperature measurement technology drift under the harsh environment of a nuclear power plant. Knowledge of the plant thermal condition is a primary performance and safety requirement for current reactors as well as all of the proposed future reactors. While the device being developed and demonstrated in this project is specifically intended to provide a "first-principles" measurement of temperature within the primary coolant loop of PWRs, Johnson noise is a general technology equally applicable to direct in-core temperature measurement. The technology being developed under the current program can be applied to the reactors of the Generation IV and nuclear hydrogen initiatives.

Variation in materials or changes in material characteristics over time do not affect the fundamental physics of how Johnson noise is produced. Because of the fundamental nature of JNT, calibration is not needed. The NRC mandates periodic calibration of primary coolant system temperature measurement instruments. Without an assured measurement uncertainty, safety margins would be required to be increased, leading to reduced thermal efficiency. In high-temperature reactor designs, standard sensors change calibration very rapidly, making their use impractical. Calibration activities are time consuming, costly, and increase exposure risk to technicians. The results of work on JNT development under subtask 2.2 are demonstrating the feasibility of JNT to measure coolant temperatures with the response rate of RTD sensors but with the added bonus of eliminating the need for periodic calibration. This saves significant time and effort and eliminates potential errors introduced by the act of calibrating.

ORNL's work on the JNT development project has progressed during fiscal year 2003 to the point of fabricating two complete JNT instruments. This represents a significant technical achievement because these units are by far the most advanced Johnson noise measurement systems ever built. Attempts to measure temperatures at nuclear power plants using JNT have now been going on for more than 30 years and the technology is more than 50 years old. While several of the attempts to employ JNT at nuclear plants have been significant accomplishments, it is not yet used at any nuclear plant and not currently commercially available. Simply put, JNT, while conceptually highly appealing, has proven very difficult to implement in a sufficiently robust and cost-effective manner. ORNL's fabrication and replication of a mechanically robust JNT instrument using standard electronics and instrumentation technology within the first two years of this project is a dramatic technical advance and appears likely to lead toward general application of this technology in nuclear reactor thermometry throughout Generation IV plants. Further, KAERI has supported JNT deployment in a plant environment by transferring (work not yet completed) the high-speed digital signal processing of the ORNL units performed at the control room from a personal computer-based system to a dedicated digital signal processing system employing technologies that could be licensed for nuclear plant control system use. Ohio State has created a testing environment for the ORNL JNT instruments to allow demonstrating their performance and viability under simulated nuclear power plant operational conditions.

Primary loop flow measurements are used to determine the core heat rate in PWRs and, as such, are a basic safety indication. These measurements are conventionally made using flowmeters based on differential pressure. Differential pressure-based flowmeters have significant, fundamental accuracy limitations and have failure modes difficult to diagnose while in service. Magnetic flowmeters offer a potential solution to these limitations. Magnetic flowmeters are highly accurate, respond linearly, and are obstructionless (no fouling; consume no pumping power). Also, the transmitter for magnetic flowmeters can be located remotely (up to hundreds of feet) from the point of measurement, thus reducing environmental exposure. The major limitation to the immediate application of magnetic flowmeters to nuclear power plants is the radiation sensitivity of the nonconductive inner pipe liner. Ceramic pipe liners are available for pipe diameters up to 30 cm. However, for larger pipes, only radiation-sensitive materials such as Teflon or rubber are available. Ceramic pipe liners are not available for larger-diameter pipes due to manufacturing and material limitations. The magnetic flowmeter liner technology demonstration is now underway by ORNL. The initial effort is evaluation of the environmental survival requirements, acceptable materials, and available fabrication technologies. The demanding conditions of a PWR hot leg have all but eliminated most materials and all but one fabrication path. It appears feasible to cement a hard-fired alumina or zirconia liner inside primary piping and meet all the material requirements. As a liner of this size will be challenging to fabricate, preliminary testing of material properties is being done with coupon samples that are more readily available and convenient for laboratory-scale testing. Laboratory radiation testing of the electrical and physical properties of the materials used in these liners will be completed by Ohio State University. KAERI is investigating their Integrated Test Facility (a thermo-hydraulic test loop) and the flowmeter testing laboratory at Flowtech Co. in Korea as possible sites for prototype testing.

Under Task 3, a 3-D reactor core kinetics model integrated with a thermo-hydraulic model of a reactor core has been developed and implemented (subtask 3.2.1). This reactor core model was used to design the protection and monitoring algorithms that focus on departure from nucleate boiling Ratio (DNBR) and local power density (LPD) (Task 3.1) and also to design advanced control algorithms (Task 3.2). Two kinds of methodologies were applied to develop the protection and monitoring algorithms: model-based and data-based methods. The model-based method is directly achieved by the 3-D reactor core model. The data-based method uses fuzzy neural networks where the inputs are measured signals including in-core neutron detector signals, etc. These methods were known to have a larger margin than existing methods. In Phase 2, advanced control algorithms were designed and applied to simple reactor models. In the latter half, we reviewed the application possibility of the advanced control algorithms to the 3-D reactor core model that describes the Yonggwang Unit 3 nuclear power plant (Korea Standard Nuclear Power Plant, [KSNP]). Also, the model predictive control algorithm has been applied to the 3-D reactor model to control the power level and distribution. Figure 2 shows the performance of the model predictive controller applied to the reactor core of a KSNP that is modeled 3-dimensionally by a reactor analysis and design code.

Planned Activities

The main focus of the sensor tasks for the coming year is to validate the performance of the developed sensors under more realistic conditions and to refine the designs based on the testing results. The solid-state flux monitor now exists in a second-generation prototype form. Each reactor test, however, consumes a sensor

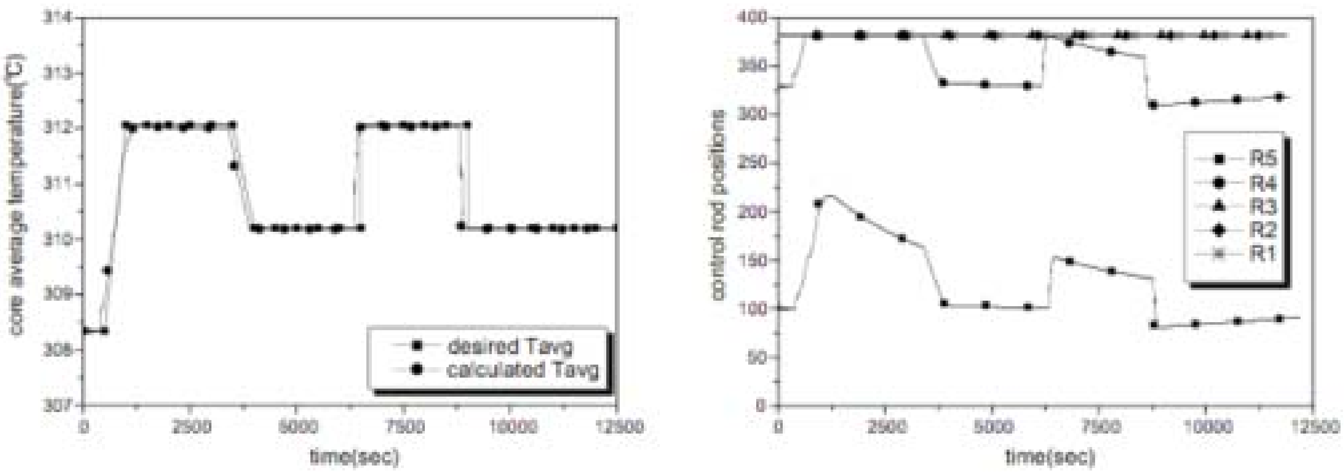


***Figure 2**. (a) Average coolant temperature and (b) regulating control rod bank positions.*

(become too activated to easily reuse). Also, to observe the mechanical condition of the sensors during testing, they are being used in a bare form and this has meant that several devices have been broken in handling. Developing an acceptable packaging system for the sensors as well as an inexpensive, rugged, repeatable wiring attachment technique is an immediate project task. Additionally, the high-temperature reactor testing remains to be performed and, in general, significantly more device testing is required to gain sufficient experience with the device characteristics.

The JNT instruments are now substantially complete. The next step is to submit these instruments to environmental testing and performance characterization at Ohio State. Following testing, a JNT instrument will be supplied to KAERI for further testing and in situ demonstrations. This year, KAERI will also complete development of the digital signal processing hardware that will be coupled to the instrument to allow the in situ testing with a realistic end-to-end temperature measurement channel.

The magnetic flowmeter fabrication testing and refinement is just now fully underway. Coupons have now been fabricated and testing setups developed. Over the course of the next year it remains the project team's intent to demonstrate whether magnetic flowmeter technology can be used as a primary flowmeter under a PWR hot-leg environment.

Task 3 will focus on developing load-following controllers and combining the core monitoring system and the control system coupled with the core design computer code. Using the core state variables and the in-core detector signals, the current reactor safety parameters are checked through the core monitoring program and the next operation core variables such as power level, control rod positions and soluble boron concentration, etc., will be determined through the core control program. The control program of a power plant determines the safety parameters based on the plant measured data. According to the determined core operation conditions, the reactor will be operated to ensure safety.

International Nuclear Energy Research Initiative

Condition Monitoring Through Advanced Sensor and Computational Technology

Principal Investigator (U.S.): Vincent K. Luk, Sandia National Laboratories (SNL)

Principal Investigator (Int.): Jung Taek Kim, Korea Atomic Energy Research Institute (KAERI)

Collaborators: Seoul National University (SNU); Pusan National University (PNU); Chungnam National University (CNU); Pennsylvania State University (PSU)

Project Number: 2002-021K

Project Start Date: December 2001

Project End Date: September 2004

Reporting Period: January — December 2003

Research Objective

The overall goal of this joint research project is to develop and demonstrate advanced sensors and computational technology for continuous monitoring of the condition of components, structures, and systems in advanced and next-generation nuclear power plants (NPPs). This project includes investigating and adapting several advanced sensor technologies from Korean and U.S. National Laboratory research communities, some of which were developed and applied in non-nuclear industries. The project team plans to investigate and develop sophisticated signal processing, noise reduction, and pattern recognition techniques and algorithms, as well as evaluate encryption and data authentication techniques for the wireless transmission of sensor data.

Research Progress

In this reporting period, this research project has actively engaged in conducting two condition monitoring test series. The first test series focuses on conducting condition monitoring of a selected check valve as an active component using a modified test loop at KAERI. The test series involves the check valve in the normal as well as degraded configurations simulating various identified failure modes. Ultrasonic device, acoustic emission, accelerometer, and ultrasonic flowmeter are used in this test loop to obtain sensor signals of the check valve at various flow conditions. KAERI staff are responsible for designing and installing the test loop and conducting the test series. PNU staff are in charge of analyzing the testing data. Sandia and CNU analysts are working on sensor signal processing and also perform finite element analyses. A schematic diagram of the check valve test loop is shown in Figure 1.

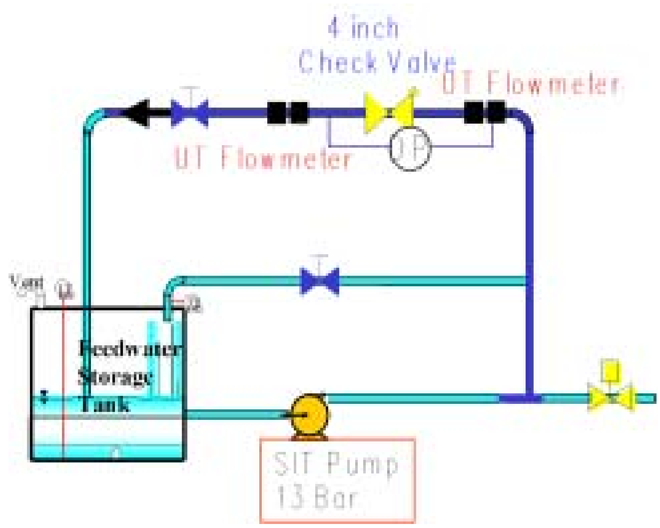


***Figure 1**. Schematic diagram of the check valve test loop.*

The actual installation of the check valve and the placement of sensors are shown in Figures 2 and 3, respectively. Sensor signal processing analyses were performed with all acquired data. Preliminary spectral analyses of acoustic emission data indicate that distinctly different vibration patterns of the check valve were obtained in its normal configuration as well as in the degraded states with disc wear and foreign objects, as depicted in Figure 4.

The second test series investigates and develops a condition monitoring system for a secondary piping elbow in a non-safety-related environment as a passive component. This test series addresses the degradation of piping systems subjected to corrosion/erosion attacks in a hostile environment of high temperature and pressure and undesirable water chemistry. The piping elbow test loop has been installed in a laboratory at SNU. This test series involves investigating the

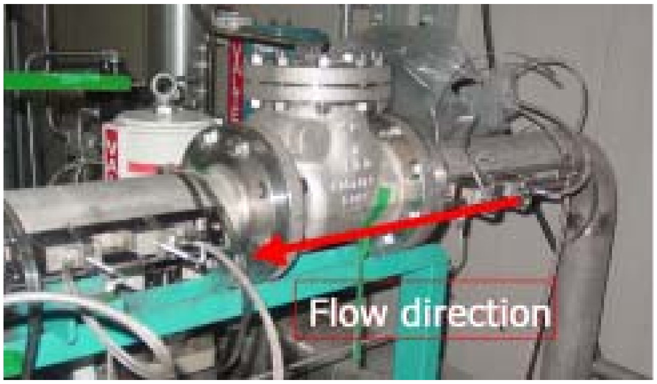


***Figure 2**. Check valve installation.*

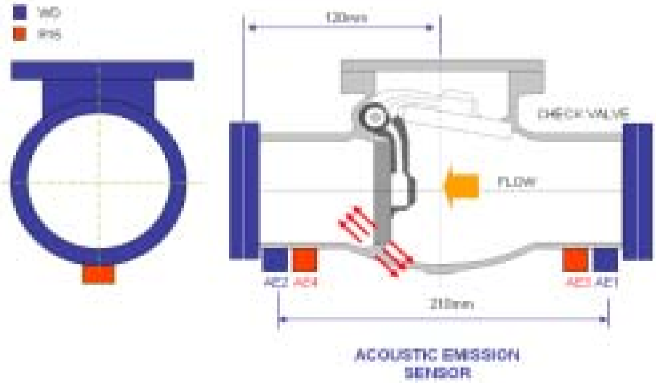
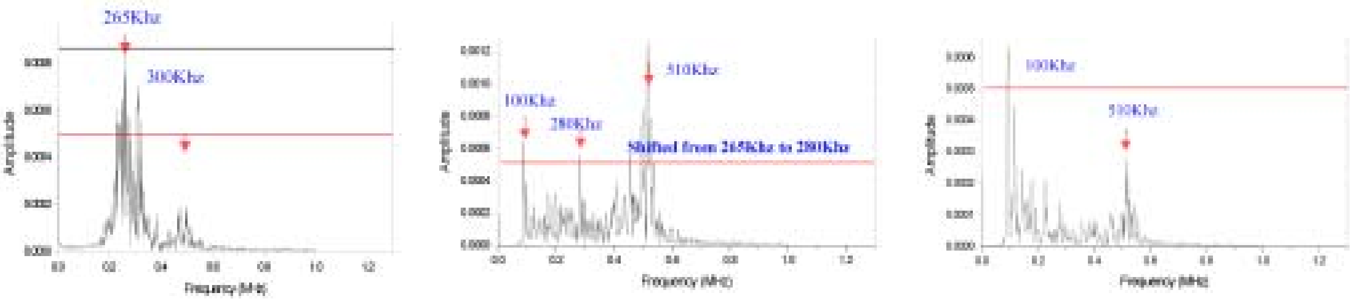


***Figure 3**. Placement of sensors.*



(a) Normal configuration

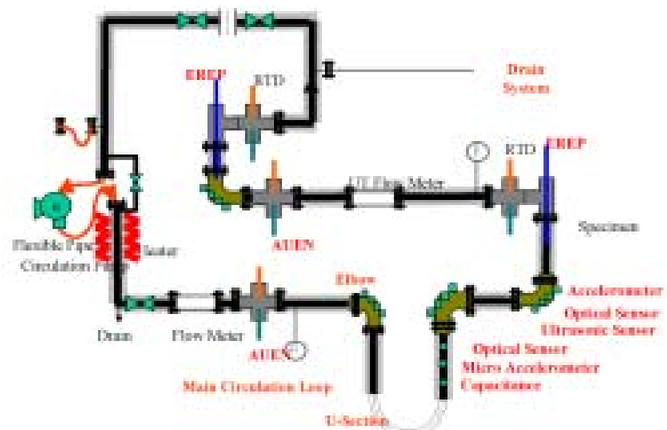
(b) Degraded state with disc wear

(c) Degraded state with foreign object

***Figure 4**. Preliminary spectral analysis results of acoustic emission signals.*

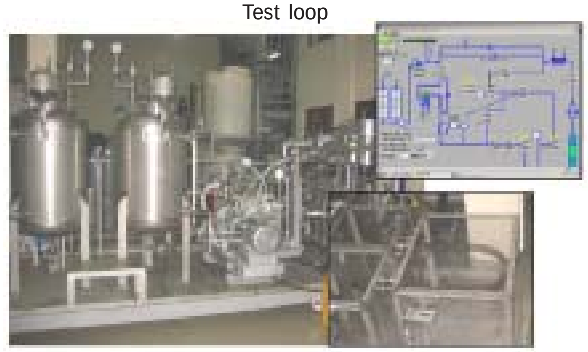
degradation behavior of the piping elbow in an accelerated corrosion/erosion environment, monitored by AUEN sensor (gold-coated electrode with metal-ceramic brazing seal). Multiple sensors will be used in this test loop, including an advanced optical fiber displacement sensor, a micro accelerometer, and a capacitance displacement sensor. The SNU staff are responsible for designing the test loop and the chemical sensors. KAERI staff are in charge of installing the test loop. Both KAERI and SNU staff share responsibility for conducting the test series. Sandia staff are in charge of designing the sensors and performing finite element and hydrodynamic analyses. Sandia and CNU staff work on sensor signal processing. A schematic diagram of the piping elbow test loop and its actual installation are shown in Figure 5.

Condition monitoring tests were conducted with the piping elbow test loop to investigate the sensitivity of the optical fiber displacement sensor, capacitance displacement sensor, and micro accelerometer. Preliminary results of this sensitivity study are shown in Figure 6, indicating that all sensors detected the same frequency peaks, and much clearer and sharper signals were obtained with the optical fiber displacement sensor. Computational fluid dynamics (CFD) analyses of the piping elbow were also performed using the FLUENT code to investigate its flow characteristics. The analysis results in terms of regions of high shear stresses and kinetic energy are shown in Figure 7.



(a) Schematic diagram

Test loop



(b) Actual installation

***Figure 5**. Piping elbow test loop.*

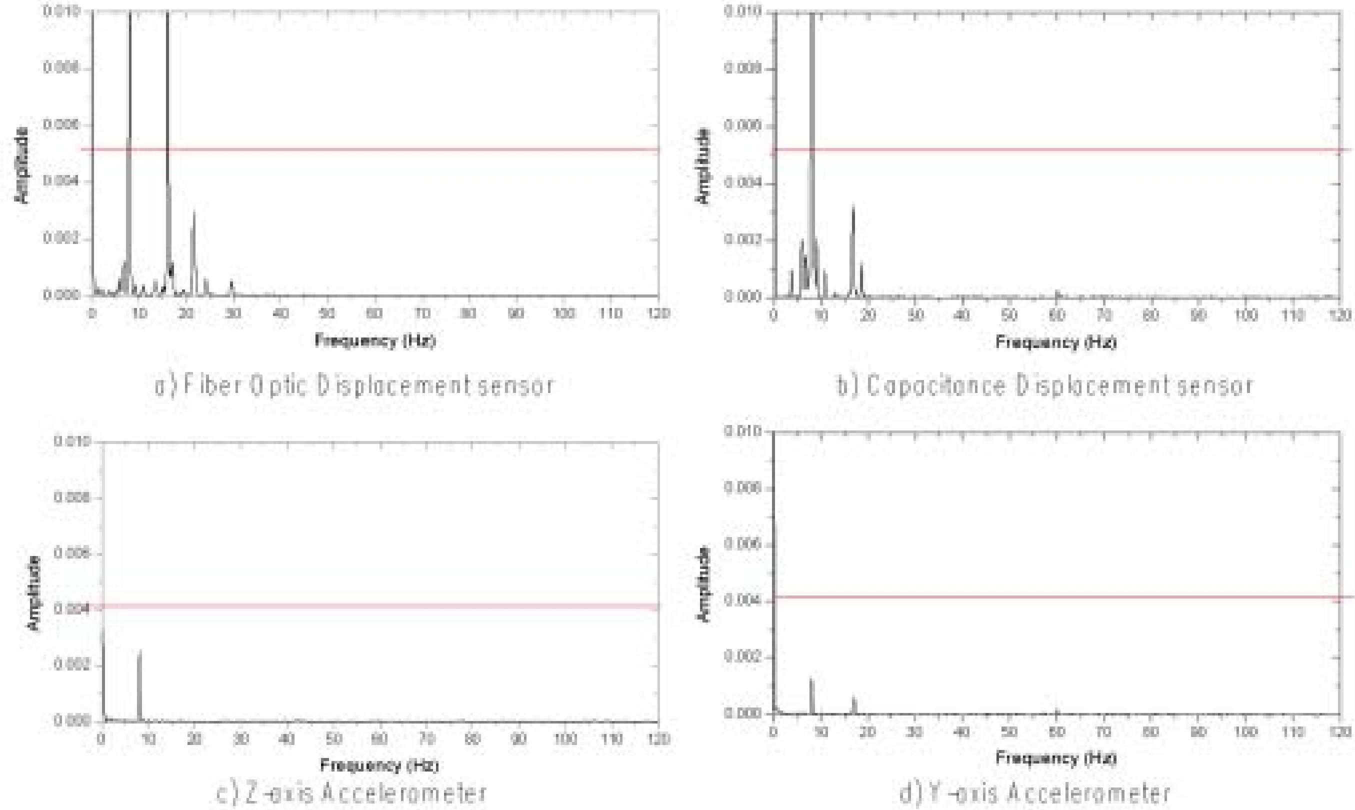


***Figure 6**. Sensor sensitivity results in piping elbow test loop.*

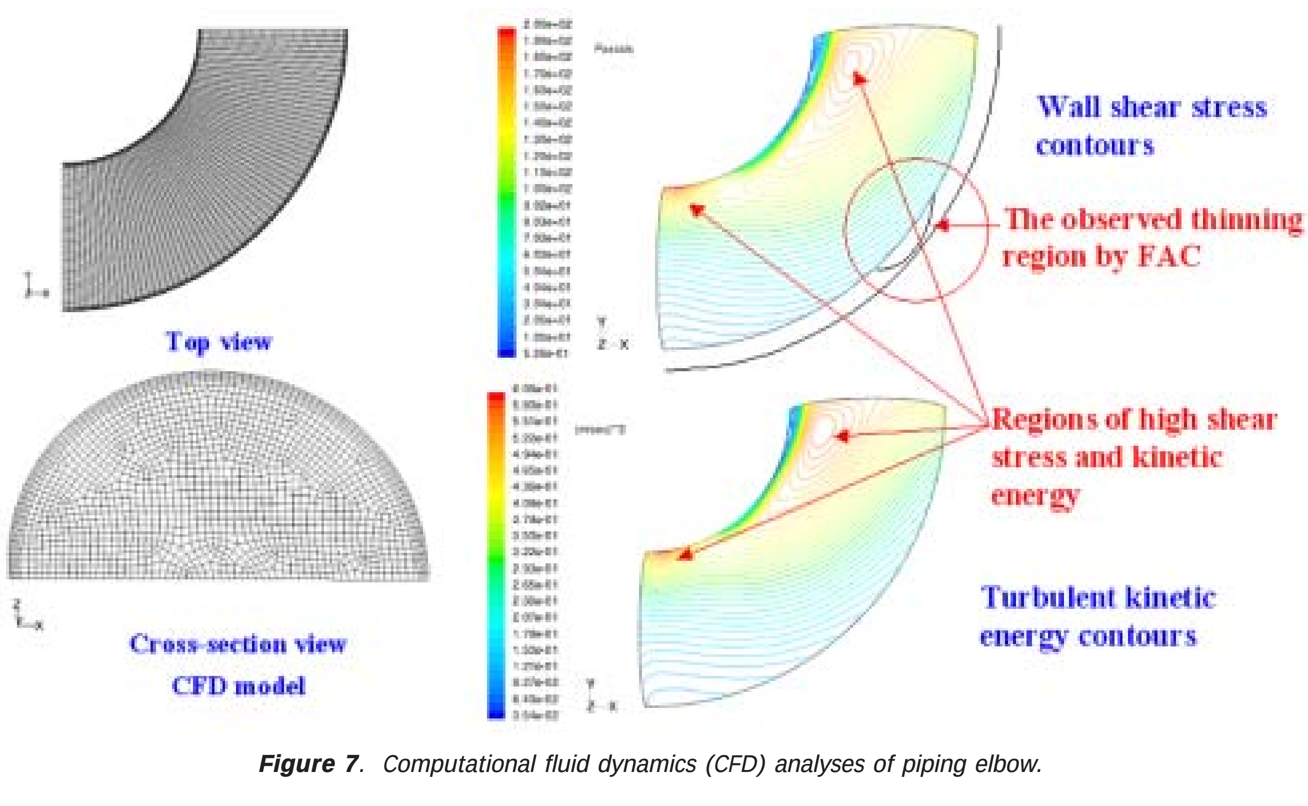


***Figure 7**. Computational fluid dynamics (CFD) analyses of piping elbow.*

In this research project, the raw sensor data obtained from the two condition monitoring test series will be analyzed and evaluated with sophisticated data processing technologies; first, by sanitizing data to filter background noise, and second, by establishing pattern recognition to correlate component response patterns to different selected degradation modes. Specific data processing techniques will be used for different sensors. In addition, finite element and hydrodynamic models of check valve and piping elbow will be developed to investigate their fluid-structure interaction response to enhance understanding of sensor data. The analysis models, when properly validated with sensor data, can be used to help select optimal sensor locations and response detection locations. In the long run, they may be used as effective predictive tools for condition monitoring of the check valve and piping elbow.

Deployment of advanced condition monitoring systems offers the prospect of improved performance, simplified design, enhanced safety, and reduced overall cost of advanced and next-generation NPPs. For advanced and next-generation NPP designs, there are opportunities to develop and implement real-time and continuous monitoring systems by integrating advanced sensor and computational technology into design and operational concepts. Through the collaborative efforts of an international team of scientists and engineers from Korea and the U.S., this research project focuses on advancing the application of the latest in sensor and computational technology for improved instrumentation, control and diagnosis of NPP components, structures, and systems.

Planned Activities

In the third year of this project, we plan to complete the two condition monitoring test series of check valve and piping elbow. A comprehensive sensor signal processing analysis will be completed for the two sets of test data. Additional finite element and hydrodynamic analyses will be performed to further understand the vibration response of check valve and piping elbow in various stages of degradation. Technical reports will be prepared to document all findings.

International Nuclear Energy Research Initiative

In-Vessel Retention Technology Development and Use for Advanced PWR Designs in the USA and Korea

Principal Investigator (U.S.): T.G. Theofanous, University of California, Santa Barbara (UCSB)

Principal Investigator (Int.): S.J. Oh, Korea Hydro & Nuclear Power Co. (KHNP)

Collaborators: J.H. Scobel (Westinghouse, U.S.)

Project Number: 2002-022-K (I)

Project Start Date: January 2002

Project End Date: September 2004

Reporting Period: January — October 2003

Research Objective

In-vessel retention (IVR) of molten core debris by means of external reactor vessel flooding is a cornerstone of severe accident management (SAM) for Westinghouse's AP600 (advanced passive light water reactor) design. The case for its effectiveness has been thoroughly documented, reviewed as a part of the licensing certification, and accepted by the U.S. Nuclear Regulatory Commission (NRC). A successful IVR would terminate a severe accident passively, with the core in a stable, coolable configuration (within the lower head), thus avoiding the largely uncertain accident evolution with the molten debris on the containment floor. This passive plant design has been upgraded by Westinghouse to the AP1000, a 1000 MWe plant very similar to the AP600. The severe accident management approach is very similar too, including IVR as the cornerstone feature, and initial evaluations indicated that this would be feasible at the higher power as well. A similar strategy is adopted in Korea for the APR1400 plant.

The overall goal of this project is to provide experimental data and develop the necessary basic understanding so as to allow the robust extension of the AP600 IVR strategy for severe accident management to higher power reactors, and in particular, to the AP1000 advanced passive design.

The project is organized in terms of two base-technology tasks and two implementation tasks. The purpose of the base-technology tasks, carried out by UCSB, is to improve the thermal margins; that is, the difference between the thermal loading (by the melt natural convection) on the inside and the limits of coolability (due to boiling crisis) on the outside. The purpose of the two implementation tasks is to apply the results of the base-technology tasks, together with timing, as deduced from core degradation/meltdown considerations, to assess IVR performance for the AP1000 (Westinghouse) and the APR1400 (KHNP).

UCSB is in charge of base-technology tasks and the overall coordination of the project, while Westinghouse and KHNP are in charge of implementation tasks for the AP1000 and APR1400, respectively.

Research Progress

On the base-technology tasks, UCSB has worked principally with improving coolability (critical heat flux) in the first two quarters and refining simulation capability of thermal loading (natural convection heat transfer) during the third quarter.

On the coolability task (for AP1000 geometric features), testing in the modified full-scale ULPU-2400 Configuration V facility (Figure 1) and in the BETA facility produced a number of important results that were used by Westinghouse to support their IVR case in the AP1000 certification by the NRC. These included:

- Demonstrated that coatings normally used (and left on) for protecting the reactor pressure vessel (RPV) can degrade burnout performance and recommended that it be removed. This was accepted by Westinghouse who defines the base material for burnout as reactor pressure vessel-grade steel.

- Defined a thermal insulation design that enhances (via streamlining the flow) coolability (critical heat flux). The streamlined geometry such as employed in ULPU Configuration V is shown to significantly increase the coolability margins previously defined in ULPU (~1.5 MW/m^2) in connection with the AP600 certification. In the upper region, a critical heat flux (CHF) of 1.8 to 2.0 MW/m^2 (an increase by 20 to

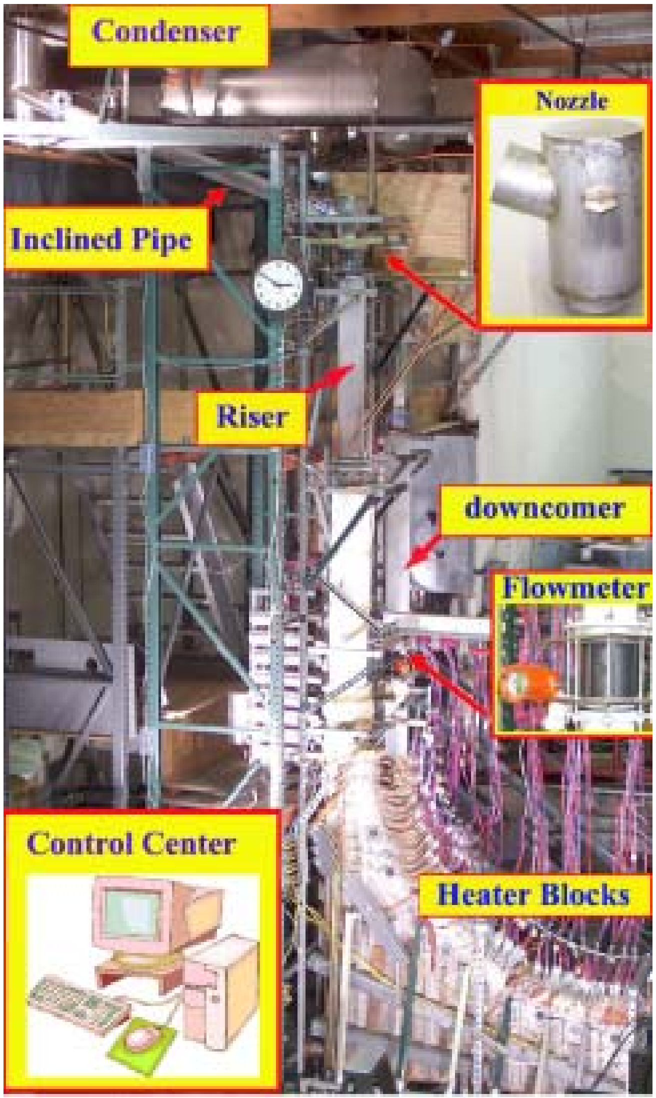


***Figure 1**. AP1000-Related ULPU 2400 Configuration V Facility.*

30%) is an appropriate estimate of the AP1000 performance (including plant-specific inlet and exit geometries) (Figure 2).

- Obtained important basic understanding (from our BETA experiments) on the relative role of hydrodynamics and heater surface nanostructure (including the effects of water chemistry), which connects the ULPU (copper material) to reactor (RPV-grade steel) and bolsters the reliability of the empirical results from ULPU. In particular, we found that a totally clean (deionized) coolant is able to promote such a degree of "cleansing" of the heater surface as to have a rather significant deleterious effect on the CHF. At the other extreme, the presence of TSP (tri-sodium-phosphate), a typical dissolved substance

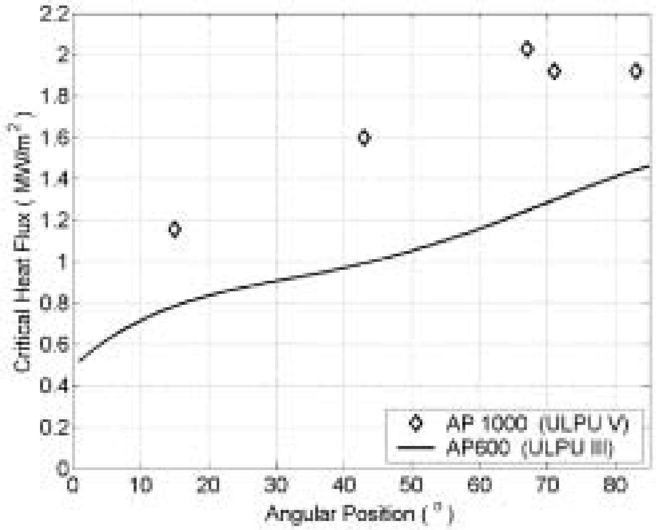


***Figure 2**. Limits of coolability in AP1000-related ULPU-V.*

in reactor cavity water (an alkaline solution), has an outstanding beneficial effect on CHF. The presence of boric acid, also a normal ingredient here, diminished this enhancement somewhat, but still the CHF is considerably higher than that with pure water. We use the term "aging" to describe the aggregate of these not yet fully understood molecular-scale phenomena. An aged copper surface such as that employed in ULPU exhibits a coolability performance similar to the bare external surface of the RPV steel.

- Under representative AP1000 exit geometry at the RPV nozzle gallery, the natural circulation flow is dominantly subcooled and modulated by periodic flashing and sweepout events and associated pressure loss phenomena at the exit. This defines the boiling/condensation loads necessary for the structural design (by Westinghouse) of the reflecting thermal insulation.

The main result on this coolability work, *Report CRSS-03/06*, was produced on schedule and delivered to the NRC in Westinghouse's response to the Draft SER on AP1000.

At the basic understanding level, the discovery of the enhancement of resistance to burnout via coolant chemistry is far-reaching. We call this phenomenon "Molecular Wicking." While not observed previously, these trends are consistent with views on basic origin of burnout, as developed from our co-current NASA-funded work carried out in BETA (with nanofilm heaters, under controls much stricter than is possible with a large-scale facility such as ULPU).

On the thermal loading task, we performed computational fluid dynamics (CFD) simulations of the natural convection startup process in a fluid layer upon a sudden application of a thermal boundary condition. Numerical results revealed a regular pattern of instability that evolves into natural convection loops. Most interestingly, natural convection experiments BETA-NC were simulated, which were performed as part of the present project. In these experiments, natural convection develops in response to the application of a constant heat-flux boundary condition on the fluid-layer's bottom surface. Temperatures of the bottom surface are measured by a high-speed, high-resolution infrared camera. Results of the 2-D and 3-D numerical simulation were compared with first-of-a-kind thermal patterns of transient natural convection (Figure 3). These comparisons allowed for evaluation of the capability of existing numerical schemes to capture such pattern formation in natural convection. Using this numerical simulation capability, we are examining models used in Westinghouse's assessment of IVR in AP1000 to calculate heat transfer in metal-rich, spherical-segment, fluid layer stratified in the bottom of the pressure vessel lower head *(findings from this effort will be provided in the final report)*. Two technical reports, one on BETA-NC experiments and the other on CFD simulation and analysis of natural convection heat transfer will be distributed to the project partners.

On the implementation task, Westinghouse put together a coherent document on the assessment of in-vessel retention for the AP1000 design that synthesized information submitted to NRC during the certification application review. This includes core degradation, melt relocation to the reactor vessel lower head and formation of melt pool in it, assessment of thermal loading, and thermal margin for IVR in AP1000. Most notably, the analysis presents bounding cases for challenges to the vessel integrity from two potential failure modes postulated from phenomena associated with lower plenum material interactions.

The first potential failure mode is from the heat fluxes generated by a bottom metal layer with a significant fraction of the fission products partitioned into the metal. The lower-bound critical heat flux at the bottom point of the reactor vessel lower head for the AP1000 geometry is greater than 800 kW/m^2. The upper bound of the heat flux from the bottom metal layer to the vessel wall is predicted to be less than 500 kW/m^2. Therefore, the heat flux from a bottom metal pool is not expected to exceed the critical heat flux at the bottom of the reactor vessel lower head by a margin of over 30%.

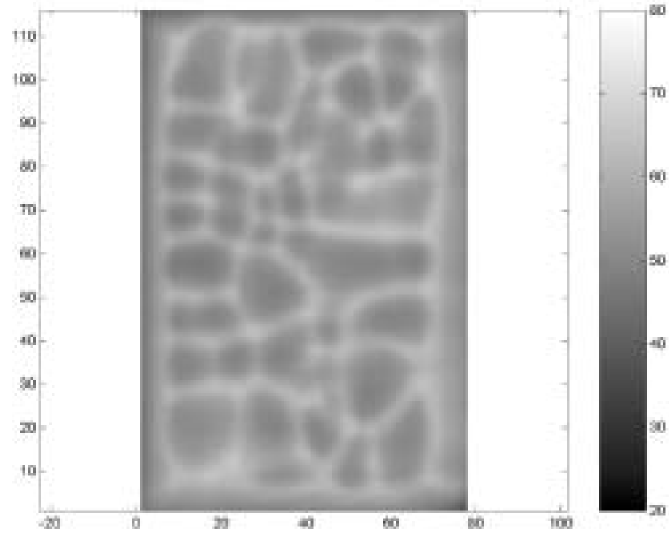


***Figure 3**. Infrared thermometry of nano-film heater under natural convection.*

The second potential failure mode is from a thinned metal layer on top of the oxidic pool, producing high heat fluxes via the focusing effect. For a bounding case of top metal layer reduced in thickness due to material interactions, the peak heat flux to the RPV wall is 1,720 kW/m^3. The lower-bound critical heat flux in this region (angular position ~ 90^o) is 1890 kW/m^3. The q/q_{CHF} is 0.91, demonstrating that margin to failure is maintained for this bounding case as well. The assumptions in this analysis are conservative with respect to increasing the heat loading to the vessel.

The main effort by KHNP this year is the construction of a natural circulation loop based on the APR1400 design and conducting the tests. Both air-water and steam-water tests have been completed. Oscillatory flow behavior was observed on some steam-water test runs. Construction of the regime map depicting the oscillatory flow is being completed. Researchers at KAIST performed forced flow CHF tests and developed CHF correlations. To carry out an integrated IVR performance evaluation for APR1400, four representative scenarios for APR1400 were examined using MAAP. Evaluations using the SCDAP code are in progress. Literature papers were reviewed for the recent results of phase separation model. The integrated IVR performance evaluation for APR1400 was completed. Except for LLOCA limiting case, sufficient margins were found using ULPU-III CHF data. The KHNP study suggests that with a proper SAM action (in-vessel water injection) vessel melt attack can be arrested under IVR even for LLOCA case.

Planned Activities

All project tasks were be completed by December 2003. A final report is being prepared jointly by the collaborating organizations and will be issued according to the schedule (April 15, 2004).

Publications of Project Results

1. T.N. Dinh, J.P. Tu, T. Salmassi, T.G, Theofanous, "Limits of Coolability in AP1000-Related ULPU-2400 Configuration V Facility," *Proceedings of the 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics*, NURETH-10, Seoul, Korea, October 5-9, 2003

2. J.P., Tu, T.N. Dinh, T.G. Theofanous, "Enhancing Resistance to Burnout via Coolant Chemistry," *Proceedings 10th Intern. Topical Meeting on Nuclear Reactor Thermal Hydraulics*, NURETH-10, Korea, October 5-9, 2003.

International Nuclear Energy Research Initiative

In-Vessel Retention Strategies for High-Power Reactors

Principal Investigator (U.S.): J.L. Rempe, Idaho National Engineering and Environmental Laboratory (INEEL)

Principal Investigator (Int.): K.Y. Suh, Seoul National University (SNU)

Collaborators: F.B. Cheung, Pennsylvania State University (PSU); S.B. Kim, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2002-022-K (II)

Project Start Date: January 2002

Project End Date: September 2005

Reporting Period: January 2004 — October 2003

In-vessel retention (IVR) of core melt is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs). If cooling were inadequate during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened at the Three Mile Island Unit 2 (TMI-2). If it is possible to ensure that the lower head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. For example, the enhanced safety of the Westinghouse Advanced 600 MWe PWR (AP600), which relied upon External Reactor Vessel Cooling (ERVC) for IVR, resulted in the U.S. Nuclear Regulatory Commission (NRC) approving the design without requiring certain conventional features common to existing LWRs. However, it is not clear that currently proposed ERVC without additional enhancements could provide sufficient heat removal for higher-power reactors (up to 1500 MWe). Hence, this three-year, U.S. Korean I-NERI project was initiated in which INEEL, SNU, PSU, and KAERI will explore options such as enhanced ERVC performance and internal core catchers that have the potential to ensure that IVR is feasible for high-power reactors.

Research Objective

The ultimate objective of this project is to develop specific recommendations to improve the safety margin for IVR in high-power reactors. The systematic approach applied to develop these recommendations combines state-of-the-art analytical tools and key U.S. and Korean experimental facilities. Recommendations focus on modifications to enhance ERVC (vessel coatings to enhance heat removal and an improved vessel/insulation configuration to facilitate steam venting) and modifications to enhance in-vessel debris coolability (improved in-vessel core catcher configuration and materials). Collaborators use improved analytical tools and experimental data to evaluate options that could increase the margin associated with these modifications. This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). This program is initially focusing on the Korean Advanced Power Reactor -1400 MWe (APR1400) design. However, margins offered by each modification will be evaluated such that results can easily be applied to a wide range of existing advanced reactor designs and next-generation reactor (Generation IV) designs.

Research Progress

As indicated in Figure 1, this three-year project includes four tasks. In Task 1, which was completed during the first year, SCDAP/RELAP5-3D© calculations were conducted to define representative bounding late phase melt conditions. Characteristic parameters from those bounding conditions (thermal loads, pressure, relocated mass, etc.) are used to design an optimized core catcher (in Task 2) and ERVC enhancements (in Task 3). Task 2 and 3 activities were initiated in the first year and will be completed during the third year of this project. In Task 4, collaborators will assess the improved margin obtained with Task 2 and 3 design modifications. Margins will be presented such that the impact of these modifications can easily be applied to other reactor designs. Because Task 4 requires results from Tasks 2 and 3, it will not be initiated until the third year of the project. As indicated in Figure 2, key facilities and capabilities of each collaborator are used to complete each task.

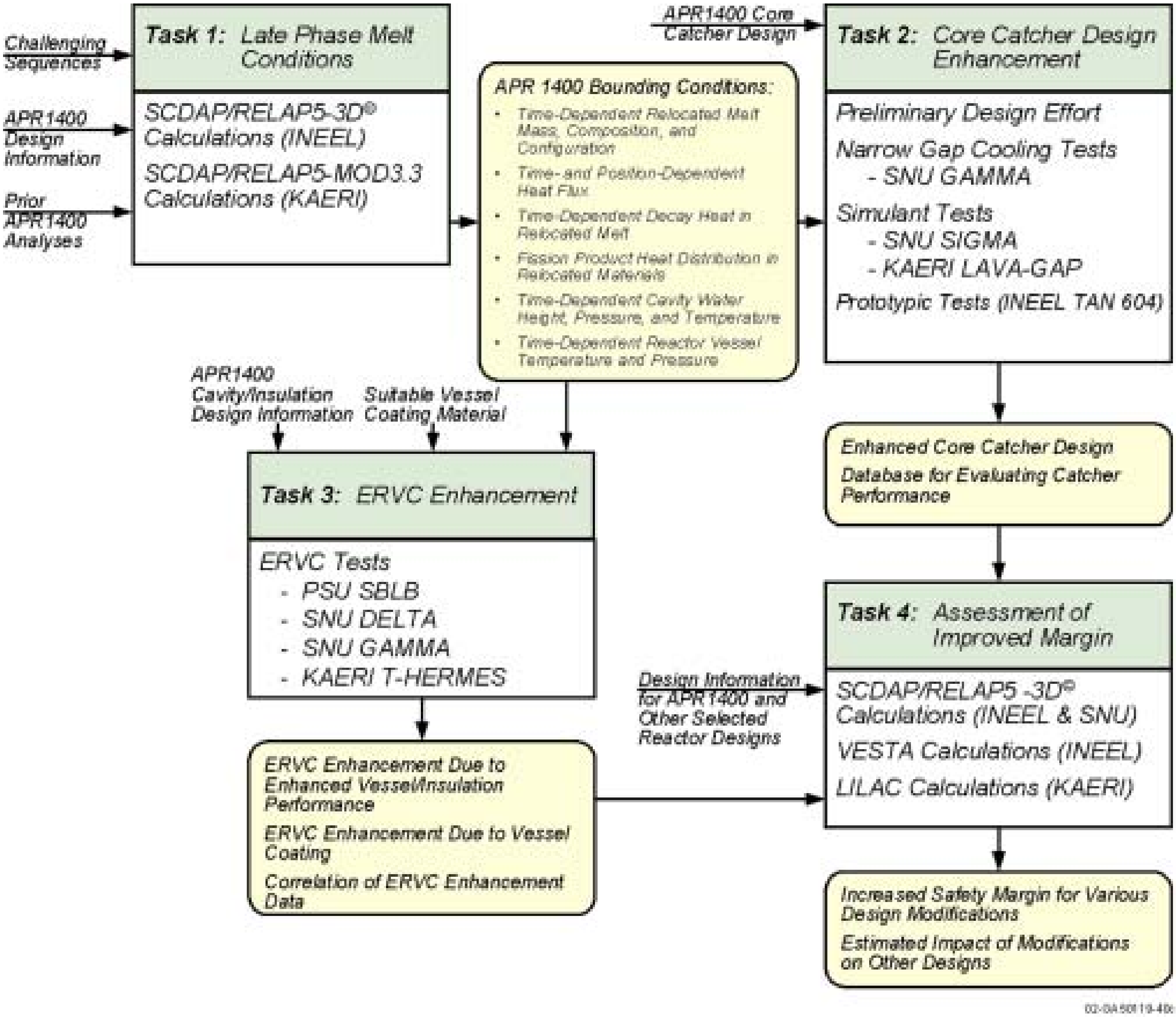


***Figure 1**. Project organization illustrating key tasks and participating organizations.*

During the second year of this project, work continued on both Tasks 2 and 3. Key tasks completed during this reporting period are highlighted below.

Continued Task 2 efforts to quantify the increased margin offered by an in-vessel core catcher. These efforts help quantify the reduced heat loads to the reactor vessel if the proposed in-vessel core catcher (see conceptual design in Figure 3) is inserted into the reactor vessel. During this reporting period, several efforts were completed to optimize the proposed core catcher design and provide insights about its performance.

- KAERI completed two in-vessel core catcher (IVCC) tests, LAVA-GAP-2 and LAVA-GAP-3. The LAVA-GAP tests, which use Al_2O_3 to simulate materials relocating from the core, provide insights about the impact of an in-vessel core catcher and candidate coatings on vessel thermal response. Post-test examinations and test instrumentation show that the presence of an insulator coating significantly reduces the thermal loads and attack from relocating materials.

- SNU conducted critical heat flux (CHF) experiments for visualization in the one-dimensional gap cooling apparatus, GAMMA 1D, and for gap cooling phenomena in the two-dimensional gap cooling apparatus, GAMMA 2D.

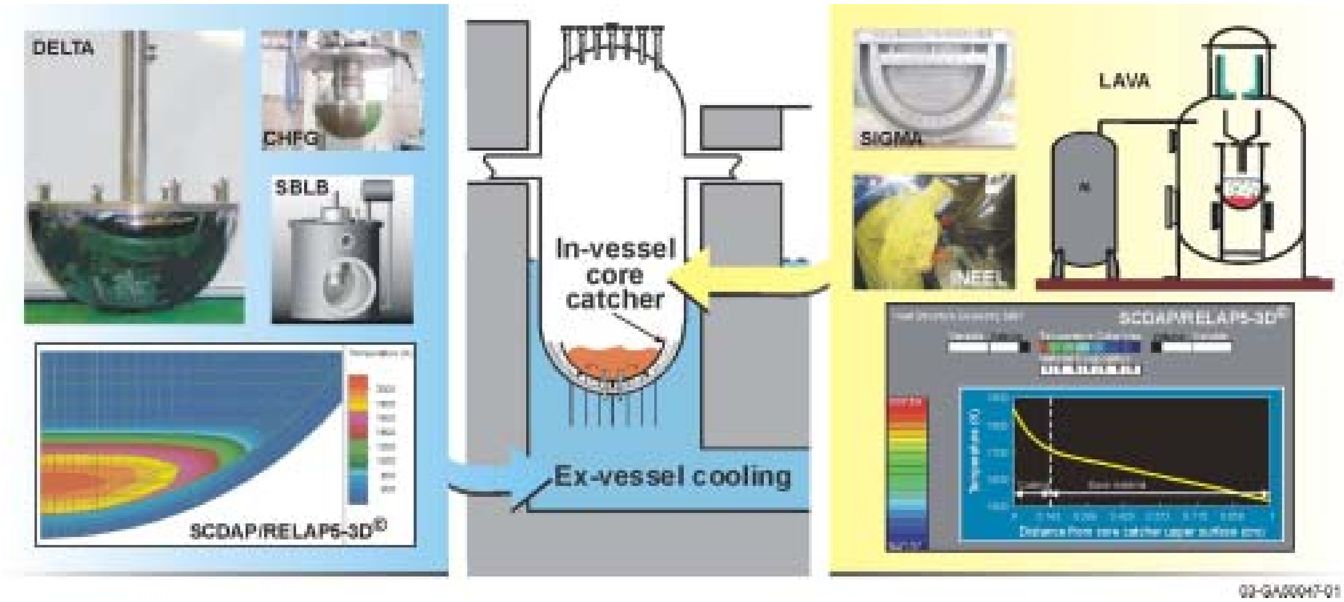


***Figure 2**. Key U.S. and Korean experimental facilities and state-of-the-art analytical tools are applied to investigate options that could enhance external reactor vessel cooling and internal core catcher performance.*

- SNU performed analyses/experiments evaluating possible gap sizes for an IVCC using the GAMMA 1D and GAMMA 2D test sections. Post-test examinations from these tests have been documented and transmitted to project collaborators.

- SNU completed post-test analyses for the SIGMA-2D tests. These tests, which provide insights about the heat load from relocated core materials, use modified cable-type heaters to simulate uniform heat generation. Results suggest that the proposed method for simulating uniform heat generation in the SIGMA-2D tests is feasible for 3D spherical geometry tests.

- SNU completed the experimental design of the SIGMA 3D facility for simulating 3D geometrical effects. Fabrication of the experimental loop is underway.

- INEEL furthered efforts to select core catcher materials by successfully completing high-temperature materials interaction tests of samples with plasma-sprayed coatings of ZrO_2 and MgO. Interim results suggest that the core catcher should consist of a MgO coating (e.g., greater than 70 wt% MgO) over a stainless steel base material. If parameters could be defined to obtain a ZrO_2 coating that does not experience any materials interactions (as was observed in one of the materials interactions tests), this coating would also be viable.

- INEEL completed the design of a test assembly for evaluating the performance of a core catcher when it is subjected to materials expected to relocate during a severe accident. The performance of the high-temperature resistance heater designed for this test assembly was successfully demonstrated.

Continued Task 3 efforts to quantify the increased margin offered with enhanced ERVC. These efforts help quantify the increased heat removal possible with enhanced coatings on the vessel outer surface and an improved vessel/insulation configuration.

- Transient quenching and steady state boiling experiments completed in PSU's SBLB facility demonstrated that microporous aluminum coatings could significantly enhance the CHF for boiling on downward-facing curved surfaces (varying from 40 to more than 100% higher). Similar results were obtained for vessels coated with copper microporous coatings. However, microporous copper coatings were found to be much less durable and tended to degrade after several boiling cycles.

- Results from PSU's analytical effort to evaluate the impact of an improved vessel/insulation design proposed during Year 1 found that a vapor spike will occur at a sudden expansion at the intersection of the conical section and the upper cylindrical section. Hence, PSU has proposed a modified design that includes a smooth transition section between these sections. Flow calculations showed that the improved design with modifications could reduce appreciably the vapor spike at the intersection when compared to the case without modifications. Fabrication of the improved vessel/insulation design with modifications

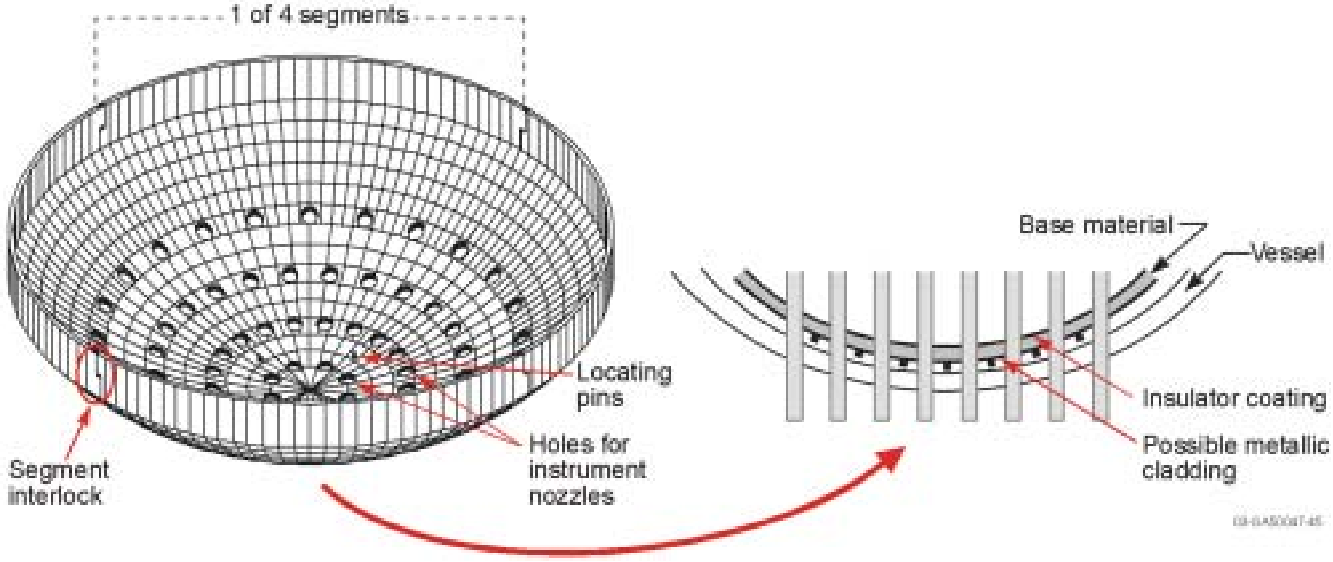


***Figure 3**. In-vessel core catcher conceptual design.*

has been completed so that its impact can be evaluated in the SBLB.

- SNU completed DELTA-3D quenching experiments for measuring heat transfer coefficients and visualizing quenching using 120 and 294 mm diameter test sections.
- SNU completed quenching experiments to visualize film boiling on a downward-facing inclined plate.
- SNU fabricated a steady-state experimental loop for measuring angular boiling curves and quantifying angular instability wavelength.
- SNU completed installation of the GAMMA 3D facility.
- KAERI completed the design of the HERMES-HALF experiment, which is a half-height, half-sector model for evaluating two-phase, natural circulation phenomena through the gap between the vessel and insulation and for providing recommendations for the APR1400 insulation. Construction and initial testing of this facility are underway. In addition, a calculational effort has been initiated to predict and provide insights about HERMES-HALF test data and project flowrates that will occur between the APR1400 vessel and insulation.

Completed programmatic requirements on or ahead of schedule. In addition to meeting programmatic requirements, collaborators participated in several program review meetings and exchanged several draft reports and technical publications. These interactions are essential to the success of this collaborative project.

- Prepared for and participated in two program review meetings: a July 8-10, 2003, meeting in Idaho Falls, and an October 5, 2003, meeting in Seoul, Korea.
- Met agreed-upon CY 2003 milestones.
- Reviewed and provided comments on draft project reports issued by collaborators. Electronic copies of reports issued during CY 2003 are provided with the complete annual report.
- Completed several peer-reviewed papers for technical conferences and archival journals. Electronic copies of these papers are provided with the complete annual report.

Planned Activities

During the third and final year of this project, recommendations regarding the design of an in-vessel core catcher and data to quantify its impact on IVR will be finalized (Task 2). Activities for enhancing ERVC through vessel coatings and an improved reactor vessel insulation geometry will also be completed (Task 3). In addition, the increased margin offered by proposed IVR enhancements will be quantified for the Task 1 bounding conditions (Task 4).

International Nuclear Energy Research Initiative

Passive Safety Optimization in Liquid Sodium-Cooled Reactors

Principal Investigator (U.S.): James E. Cahalan, Argonne National Laboratory

Principal Investigator (Korea): Dohee Hahn, Korea Atomic Energy Research Institute

Project Number: 2003-002-K

Project Start Date: January 2003

Project End Date: September 2006

Reporting Period: January — December 2003

Research Objective

This project identifies and evaluates safety advances with potential cost reductions offered by innovative design features in sodium-cooled, metallic-fueled fast reactors. Safety margin enhancements provided by specific design features are quantified with a combination of advanced computational model development (Task 1), analyses of innovative reactor (Task 2), balance-of-plant (Task 3) design features, and specification of laboratory experiments for concept and model validation (Task 4). Each of the tasks is specifically aimed at identifying and accurately quantifying the safety and operational performance benefits of innovative design features for simplifying reactor and plant designs and reducing costs. Figure 1 provides an overview of the activities in each task. Each of the four tasks in this project is a collaboration between Argonne National Laboratory (ANL) and the Korea Atomic Energy Research Institute (KAERI), with activities shared by the two organizations.

Research Progress

Task 1 provides for development, implementation, and testing of a detailed three-dimensional fuel subassembly thermal-hydraulic model. Detailed modeling is necessary for the accurate quantification of fuel, cladding, coolant, and structure temperatures in passive safety transient analysis. It enables evaluation and selection of reactor design features that promote safe and reliable operation and eliminate core damage in accidents.
The new model will be specifically designed for computational efficiency and easy integration into the U.S. SAS4A/SASSYS-1 computer code and the Korean SSC-K computer code. This integration feature is necessary for coupling to reactor kinetics, reactivity feedback, and structural mechanics models that simulate reactor design performance. During the first year of the project, ANL and KAERI researchers developed the model formulation and began implementation. ANL investigators have focused on the formulation and numerical solution technique for the conservation equations. KAERI researchers have begun detailed investigations that will yield data for correlation of constitutive relationships. Interfacing requirements for integration with SAS4A/SASSYS-1 and SSC-K have been determined, and writing of the computer code for the model has progressed to 50% completion.

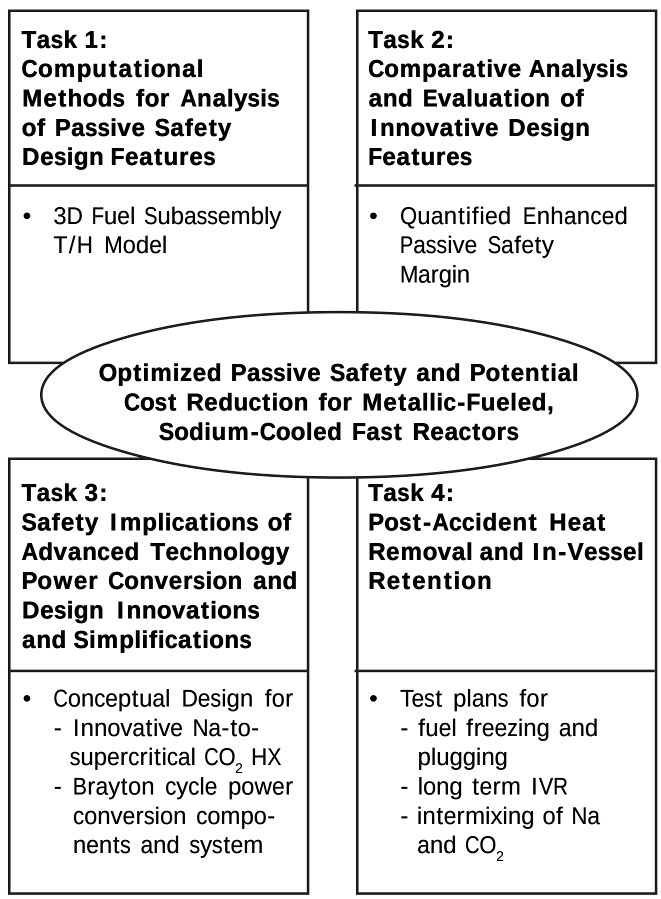


***Figure 1**. R&D task structure.*

In Task 2, the advanced modeling capability developed in Task 1 and state-of-the-art modeling capabilities are used to perform integrated safety assessments of passive safety design features that promote safe and reliable operation and eliminate core damage in accidents. The conceptual design of a prototypic metallic-fueled, sodium-cooled reactor plant serves as the framework for computational investigations that quantify the relative safety margins provided by specific advanced design features such as innovative enhancements of negative reactivity feedback based on fuel, coolant, and structural temperature changes, and the resulting mechanical and kinetics effects. Core restraint design to enhance negative temperature reactivity feedback is a leading candidate for improving passive safety performance. Additionally, metallic fuel and control rod drive thermal expansion mechanisms are also candidates for innovation and improvements of inherent safety performance. During the first year of the project, a prototype reactor design was selected and documented. ANL and KAERI investigators began assembly of models of this design and its safety features for use in integrated safety analyses using the SAS4A/SASSYS-1 and SSC-K codes.

Task 3 provides for investigations of the safety performance characteristics of coupling a supercritical CO_2 Brayton power cycle with a liquid sodium-cooled reactor. The activities include studies of innovative design concepts for sodium-to-supercritical CO_2 heat exchangers, particularly concepts that eliminate the intermediate heat transport system (IHTS) and thus reduce construction and operating costs. The evaluations include 1) determination of supercritical CO_2 Brayton cycle conditions and power cycle efficiencies; 2) investigation and optimization of coupled reactor/balance-of-plant systems, components, and control strategies; and 3) assessment of plant systems performance in accidents initiated by equipment failures in the power conversion system. Quantitative assessments of equipment failure consequences and the impacts on plant safety performance guide selection of the supercritical CO_2 system configuration. Specifically, the consequences of heat exchanger boundary failure resulting in intermixing of CO_2 and sodium determine the feasibility of IHTS elimination. During the first year of the project, innovative heat exchanger designs have been analyzed to assess their performance potential, and supercritical CO_2 Brayton cycle efficiencies have been calculated for power systems coupled to a sodium-cooled reactor.

Task 4 of the project provides for planning of three experiments designed to provide technical data to support design for limitation of severe accident consequences, and elimination of off-site impacts. The three experiments will evaluate 1) the potential for freezing and plugging of molten metallic fuel in above- and below-core structures, 2) the ability of steel structures to contain molten fuel/steel melts for long periods of time, and 3) the intermixing of high-pressure CO_2 and sodium as a result of boundary failure in a sodium-to-supercritical CO_2 heat exchanger. During the first year of the project, a test plan was prepared for determination of the solidus/liquidus and mobilization temperatures of metallic fuel and mixtures of metallic fuel and steel constituents.

All project activities are documented in a project definition report that specifies activities, organization responsibilities, and schedules. The project definition report is updated annually.

Planned Activities

In the upcoming year, the Task 1 model coding will be completed for the single computer processor version of the model. This version will be capable of simulating one or more fuel subassemblies in full geometric detail. ANL researchers will implement the model in SAS4A/SASSYS-1; KAERI researchers will begin integration of the model with the SSC-K code. Initial proof testing will be performed, and verification of the new model will begin at both organizations using instrumented subassembly test data from the EBR-II and FFTF reactors as well as data from the ORNL THORS laboratory tests. Work will begin on the multiple processor version of the model, capable of running a whole-core simulation with detailed subchannel geometry in all subassemblies. KAERI will continue development of constitutive models, including an advanced thermal conduction model and assessment of turbulent mixing parameters. Work will begin on the initial version of an input preprocessor that will reduce the amount of effort required to produce the large amount of input data required for whole-core simulations. Documentation will include a model formulation report, which will describe phenomenological equations and numerical solution techniques, and a code architecture report, which will specify the code structure, programming language, and data management techniques.

Task 2 activities in 2004 will complete the SAS4A/SASSYS-1 and SSC-K baseline input data construction. Transient scenarios to be analyzed for evaluation of safety design features will be identified, and the criteria for merit evaluation will be specified. Transient analyses with state-of-the-art methods and models will be completed, and the analysis results will be documented in project progress reports and in papers presented at international technical meetings.

During 2004, activities in Task 3 will proceed with continued evaluation of design concepts for supercritical CO_2 gas turbine Brayton cycle power conversion. Components and systems will be investigated for overpressure protection of the reactor primary coolant system due to failure of the CO_2 pressure boundary in the heat exchanger. Development of a plant analyzer computer code will be completed. The plant analyzer will have the capability to model and quantify the consequences of accidents involving supercritical CO_2 gas turbine Brayton cycle power conversion system conceptual designs. A set of accident sequences will be identified that characterizes coupled reactor/balance-of-plant safety performance, and initial analyses of the accident sequences will be completed. Needs for system or equipment modifications, revised control strategies, additional equipment, or new systems to ensure or improve safety will be identified. The effectiveness of revised design features will be quantified with additional systems analyses. Results from completed investigations and analyses will be documented in project reports.

For Task 4 in the upcoming year, a design concept will be developed for tests investigating the relocation and freezing behavior of molten metallic fuel in coolant channels, including melt interaction with steel structure. Available information relevant to design (e.g., existing facilities such as the CAMEL loop at ANL) will be gathered. Melt flows driven by gravity as well as pressure will be considered. In collaboration, ANL and KAERI have jointly identified three specific technical issues for this activity, namely, 1) the role of the mushy zone formed during freezing of metallic alloy, 2) possible effects of residual sodium in coolant channels, and 3) the characteristics of chemical/metallurgical interactions of melt constituents with steel. These issues will be addressed in the considered design concept.

International Nuclear Energy Research Initiative

Developing and Evaluating Candidate Materials for Generation IV Supercritical Water Reactors

Principal Investigator (U.S.): J.I. Cole, Argonne National Laboratory

Principal Investigator (Republic of Korea): J. Jang, Korea Atomic Energy Research Institute (KAERI)

Collaborators: Korea Advanced Institute of Science and Technology (KAIST), University of Michigan, Idaho National Engineering and Environmental Laboratory (INEEL), University of Wisconsin

Project Number: 2003-008-K

Project Start Date: January 2003

Project End Date: September 2006

Reporting Period: January — October 2003

Research Objective

The Republic of Korea and the United States have a shared interest in the development of advanced reactor systems that employ supercritical water as a coolant. The supercritical water reactor (SCWR) is one of the six concepts selected by the Generation IV Roadmap process and holds great promise in terms of increased thermal efficiency and reduced plant costs due to the elimination of many nuclear steam supply system (NSSS) components (e.g., recirculation pumps, pressurizers, steam generators, steam separators, and dryers) and a corresponding reduction in containment volume. However, little information is available on the ability of engineering alloys to withstand the aggressive environment present in the SCWR, which operates at significantly higher pressures and temperatures than light water reactors (LWRs). For example, the Generation IV reference SCWR design outlet temperature is 500°C, and the reference coolant pressure is 25 MPa.

The goal of this project is to establish candidate materials for SCWR cladding and core internals and to start evaluating mechanical properties, radiation stability, corrosion and stress corrosion cracking resistance, and weldability for a selected number of these candidate materials. As part of qualification testing, this effort will focus on understanding the mechanisms affecting property degradation and developing alloys with improved resistance to such degradation.

The project is structured into three phases. Phase I of the project involves performing a literature survey to identify the most promising candidate alloys for qualification tests. In Phase II of the project, each candidate alloy will be evaluated in terms of high-temperature tensile and creep strength, corrosion and stress corrosion cracking susceptibility, radiation stability, and weldability. Finally, the third phase of the project will provide material recommendations and propose a reactor irradiation plan to guide future in-reactor materials test programs. Although this project will not be able to perform a complete qualification of materials for SCWR applications, it will generate data to support recommending candidate materials for further evaluation.

Research Progress

Literature Survey

The literature survey was completed, and candidate materials have been selected. Table 1 lists candidate materials selected for testing along with highlights of the major advantages and disadvantages of these alloys. Ferritic-martensitic steels were chosen due to their extensive use in supercritical fossil plant (SCFP) internals. In addition, one of the early generation ferritic alloys (HT-9) was tested extensively in the U.S. breeder reactor program and shown to have extremely high swelling resistance. However, the creep strength of HT-9 is too low for application under SCWR conditions and thus it was not selected as a candidate alloy. Alloy T91 has a nominal composition Fe-9Cr-MoVNb and has seen wide use in non-nuclear applications. Alloy T122 (HCM12A) was developed for high creep resistance and has a nominal composition of 12Cr-MoVNbW, but has not yet seen extensive use in commercial SCFP. Both of these alloys are being considered as SCWR fuel cladding material; however, there are free or no data on their radiation behavior.

Results from the survey indicate that conventional 304 and 316 austenitic stainless steels may be highly

***Table 1**. Candidate alloy list.*

Alloy Class	Alloy	Advantages	Limitations
Ferritic-Martensitic			
	T91-Fe-9Cr-MoVNb	Low swelling	Corrosion, high-temperature creep strength, low-temperature radiation embrittlement
	T122 (HCM12A)-Fe-12Cr-MoVNbW	Low swelling	Corrosion, high-temperature creep strength, low-temperature radiation embrittlement, neutronics
Austenitic Stainless Steels			
	Alloy D-9-Ti–Modified Fe-15Cr-15Ni 2.2Mo	Corrosion resistance, high-temperature creep strength	Low thermal conductivity, susceptibility to irradiation assisted stress corrosion cracking
High Ni alloys			
	Alloy 690Ni-30Cr-10Fe	High-temperature creep and corrosion resistance	Irradiation-induced grain boundary embrittlement
	Alloy 800H Fe-32Ni-20Cr-TiAl	High-temperature creep and corrosion resistance	Irradiation-induced grain boundary embrittlement
Oxide-Dispersion Strengthened (ODS) Steels			
	MA957 Fe-14Cr-Mo +0.25Y_20_3	High-temperature creep strength, swelling resistance	Corrosion, low-temperature radiation embrittlement

susceptible to both swelling and stress corrosion cracking in the SCWR environment. More advanced low-swelling alloys were developed for the breeder reactor program. These alloys were optimized through the addition of stabilizing elements such as Ti and Nb in combination with thermomechanical treatments. The alloy D-9 listed in Table 1 is one of the low-swelling austenitics which may be suitable for SCWR core internals.

Superalloys have been developed for both thermal creep resistance and high corrosion resistance. Two of these alloys were selected as candidate alloys. Alloy 690 is a Ni-base superalloy developed for chemical processing and high-temperature aerospace applications. Alloy 800H is a high Ni and high Cr Fe-based alloy developed for chemical and petrochemical processing and is code qualified for nuclear applications. The major concern with Ni base alloys is that earlier studies indicate that they may be susceptible to grain boundary embrittlement during irradiation.

One of the major limitations of the ferritic-martensitic alloys is their low creep strength at temperatures above ~600°C. This problem has been addressed by the addition of dispersions of hard oxide particles to the material through mechanical alloying. The resulting oxide dispersion strengthened (ODS) material exhibits significantly higher creep strength than conventional ferritic alloys. The ODS alloys may be ideally suited for cladding applications, although the alloys are still in the early stages of development.

High-Temperature Tensile Behavior of Alloy T122 (HCM12A) and ODS Alloy PM2000

Two commercially available alloys, one ferritic-martensitic and one ODS alloy, were obtained for initial scoping studies. The ferritic-martensitic alloys was tested at room temperature and at temperatures ranging from 600° to 1000°C at a relatively fast strain rate (1 x 10^{-3} s^{-1}) to investigate the behavior of the material during short-duration, off-normal or transient temperature excursions. These tests were performed at the INEEL. Results revealed a significant drop in alloy tensile strength starting at temperatures around 700°C and an even more substantial drop above 800°C during this short-term test. The ODS alloy PCM2000 is an Fe-20Cr alloy with 0.5 wt% $Y^{2}O^{3}$ dispersions. Tensile and creep rupture tests were performed at KAERI. Results indicate improved high-temperature tensile and creep strength over conventional ferritic-martensitic steels, with the yield strength of alloy T122 (HCM12A) being less than 100 MPa at 700°C while the ODS alloy PM2000 had a yield strength close to 200 MPa at 700°C.

Supercritical Water Testing Loops

Both static and stress corrosion cracking (SCC) loops for testing candidate materials in supercritical water are being constructed. A multisample supercritical water stress corrosion cracking (SCW-SCC) facility is being constructed at the University of Michigan. The system will allow the conduct of static and SCC tests in supercritical water and operate at temperatures up to 600°C at a pressure of 25 MPa. Design of the test system is complete, and construction is nearly complete.

KAERI has completed construction of two test vessels for conducting static corrosion tests in supercritical water. Initial testing of the ferritic-martensitic alloys T91, T122 (HCM12A), T92, and MA956 have been completed. The alloys were exposed to 627°C supercritical water at a pressure of 25 MPa for 200 hours. Tests results revealed that the rate of corrosion for alloy T91 was substantially higher than for alloy T122 (HCM12A). Scanning electron micrographs of alloys T91 and T122 following the test are presented in Figure 1, while the chart in Figure 2 illustrates how the corrosion resistance of the higher Cr alloys T122 and MA956 is substantially better than the lower Cr alloys.

Planned Activities

Work continues on acquiring candidate materials and fabricating test samples for both high-temperature tensile and creep studies and corrosion and stress corrosion cracking studies. High-temperature creep strength evaluation of alloy T122 (HCM12A) will begin at INEEL during the final quarter of this year, and other candidate materials will be tested to fill gaps in data that will be critical for qualification in an SCWR during the 2^{nd} year of the project. The completion of the multisample supercritical water test loop is planned by the end of the calendar year, and testing will begin early in Year 2 of the project. An irradiation of the ferritic-martensitic candidate alloys with protons is planned by the end of this year. Proton irradiations of the rest of the candidate alloys will occur during the second year of the project.

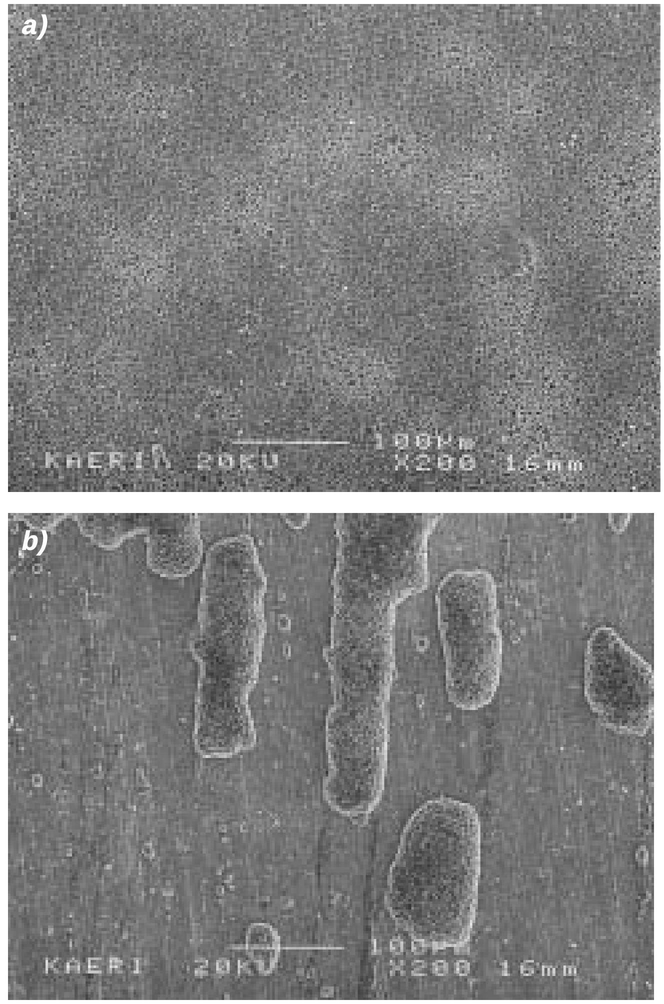


Figure 1. *Scanning electron micrographs of the surface of a) alloy T91(Fe-9Cr-MoVNb) and b) T122 (HCM12A) (Fe-12Cr-MoVNbW) following exposure to supercritical water at a temperature of 627°C and a pressure of 25 MPa for 200 hours.*

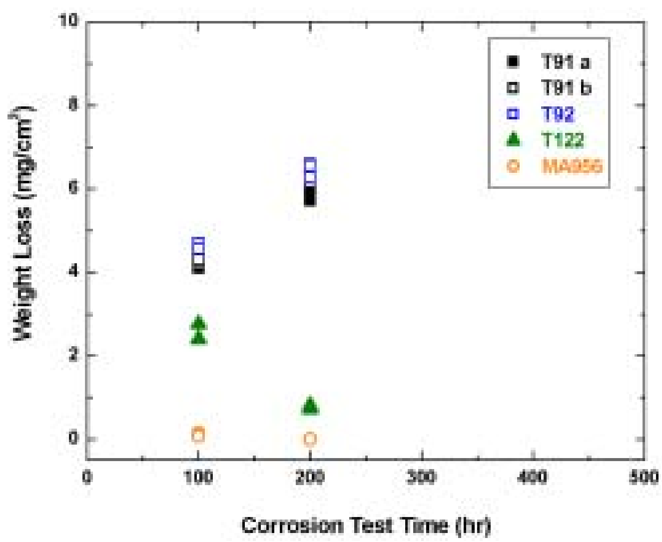


Figure 2*. Evaluation of corrosion behavior of various candidate alloys. Better corrosion resistance is indicated by lower weight loss.*

The microstructure of these alloys will then be characterized at Argonne National Laboratory, and samples will be tested in the University of Michigan SCC loop. Candidate alloy SCC samples will also undergo surface modification treatments at the University of Wisconsin followed by SCC testing at Michigan. KAIST will continue their evaluation and development of ODS alloys, while KAERI will continue with corrosion and stress corrosion cracking studies on candidate alloys. Studies on the joining of candidate materials will start at Argonne beginning in the second half of next year and be completed during the third year of the project. To optimize the relevance of project findings, the project will continue coordination with the Supercritical Water Reactor Product Manager for the Generation IV program. Knowledge of the candidate materials operational envelope will be critical to those involved with SCWR system design.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): Dr. Chang Oh, Idaho National Engineering and Environmental Laboratory

Principal Investigator (Int.): Professor Hee Cheon No, Korea Advanced Institute of Science and Technology

Collaborators: Prof. Goon Cherl Park, Seoul National University (SNU); Profs. John Lee, William Martin, and James Holloway, University of Michigan (UM)

Project Number: 2003-013-K

Project Start Date: January 2003

Project End Date: September 2006

Reporting Period: January — September 2003

Research Objective

The very high-temperature gas-cooled reactors (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-temperatures to support process heat application, such as desalination and cogeneration, the VHTGR's higher temperatures are suitable for applications such as thermochemical hydrogen production, as shown in Figure 1. However, the high-temperature operation can be detrimental to safety following a loss-of-coolant accident (LOCA) initiated by pipe breaks caused by seismic or other events. Following the loss of coolant through the break and coolant depressurization, air from the containment will enter the core by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structures and fuel. The oxidation will release heat and accelerate the heat-up of the reactor core.

Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. The Idaho National Engineering and Environmental Laboratory (INEEL) has investigated this event for the past three years for the HTGR. However, the computer codes used, and in fact the world's computer codes, have not been sufficiently developed and validated to predict this event. New code development, improvement of the existing codes, and experimental validation are imperative to narrow the uncertainty in the predictions of this type of accident.

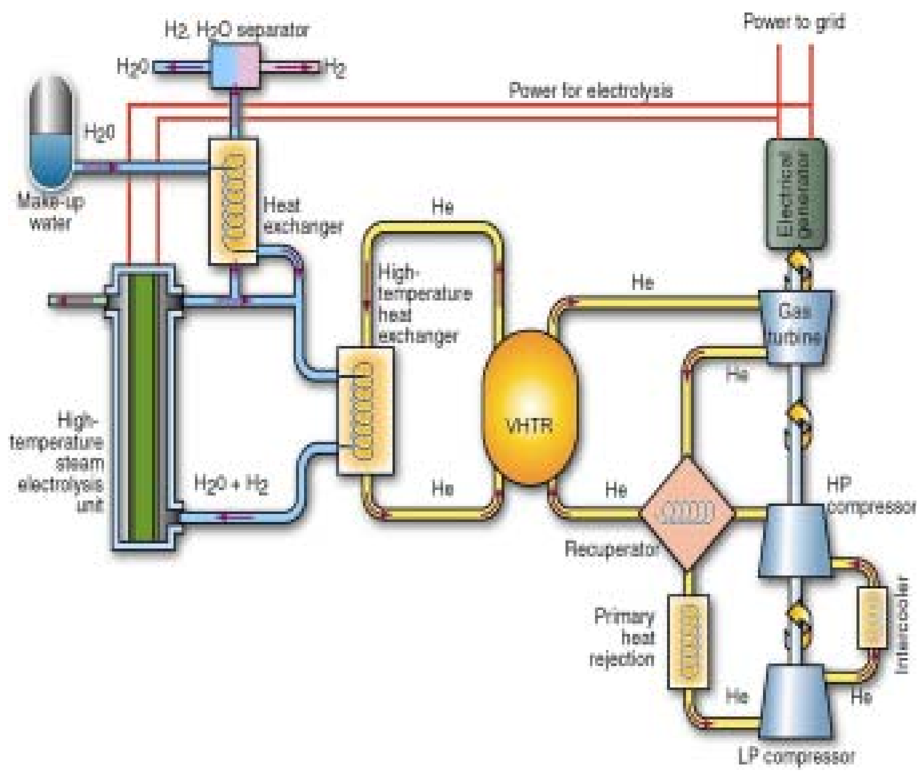


__Figure 1__. The conceptual design of VHTGR.

The objectives of this Korean/United States collaboration are to develop advanced computational methods for VHTGR safety analysis codes and to validate these computer codes.

Research Progress

The collaborators for this research project are the INEEL, the Korea Advanced Institute of Science and Technology (KAIST), Seoul National University (SNU), and the University of Michigan (UM). This project consists of six tasks for developing, improving, and validating computer codes for analysis of the VHGTR: 1) develop a computational fluid dynamics code for benchmarking, 2) perform a reactor cavity cooling system (RCCS) experiment, 3) perform an air ingress experiment, 4) improve the system analysis codes RELAP5/ATHENA and MELCOR, 5) develop an advanced neutronic model, and 6) verify and validate the computer codes. The primary activities and key accomplishments for each task are summarized below.

Task 1 — CFD thermal hydraulic benchmark code development (KAIST). The 1-D analysis tool was benchmarked using Japanese data. The present 1-D module is based on the Implicit Continuous Eulerian (ICE) technique. To solve local multidimensional effects on air ingress in a VHTGR, a test version of the 3-D analysis program was made separately from the 1-D module and was benchmarked using FLUENT6 and Ogawa's graphite oxidation experiment in a circular tube. The axi-symmetric results from the 3-D analysis module are physically well verified for local concentration and temperature profiles. After successful verification and validation of the 1-D and 3-D modules, the program structure has been improved to link both modules together, allowing sufficient flexibility to model a VHTGR system. In addition, the current version has a porous media model and 3-D heat conduction models, including a porous solid. The integrity of the new linked version has been tested using basic sample problems. Additional simulations for cases having multidimensional effects are under way to demonstrate the 3-D module's predictive capability.

Task 2 — RCCS experiment (SNU). Preliminary calculations with CFS were performed to verify the propriety of the new design of the SNU-RCCS. The SNU-RCCS has advantages of good coolability and a simpler structure than existing air-cooled and water-cooled designs. The calculations investigated the compressing power of the active cooling system (ACS) required to prevent boiling in the water tank during normal operation. The calculations also investigated the fluid temperature during accidents initiated by a partial loss of the ACS. Radiation is the most important heat transfer process in the RCCS. A simple experimental facility for the measurement of emissivity was established to measure the heat transfer rate by radiation. The emissivity of a sample is measured using an infrared thermometer.

Task 3 — Air ingress experiment (KAIST). A facility for the air-ingress experiment was manufactured. The following test parameters were determined for the experiment through a literature survey: 1) temperature, 2) gas concentration, 3) gas velocity, 4) geometry, and 5) moisture. The geometries and materials of graphite specimens were determined. The experimental strategy was established, and the air ingress experiment is progressing along eight steps: pretest 1, pretest 2, kinetic test, chemical reaction modeling, mass transfer

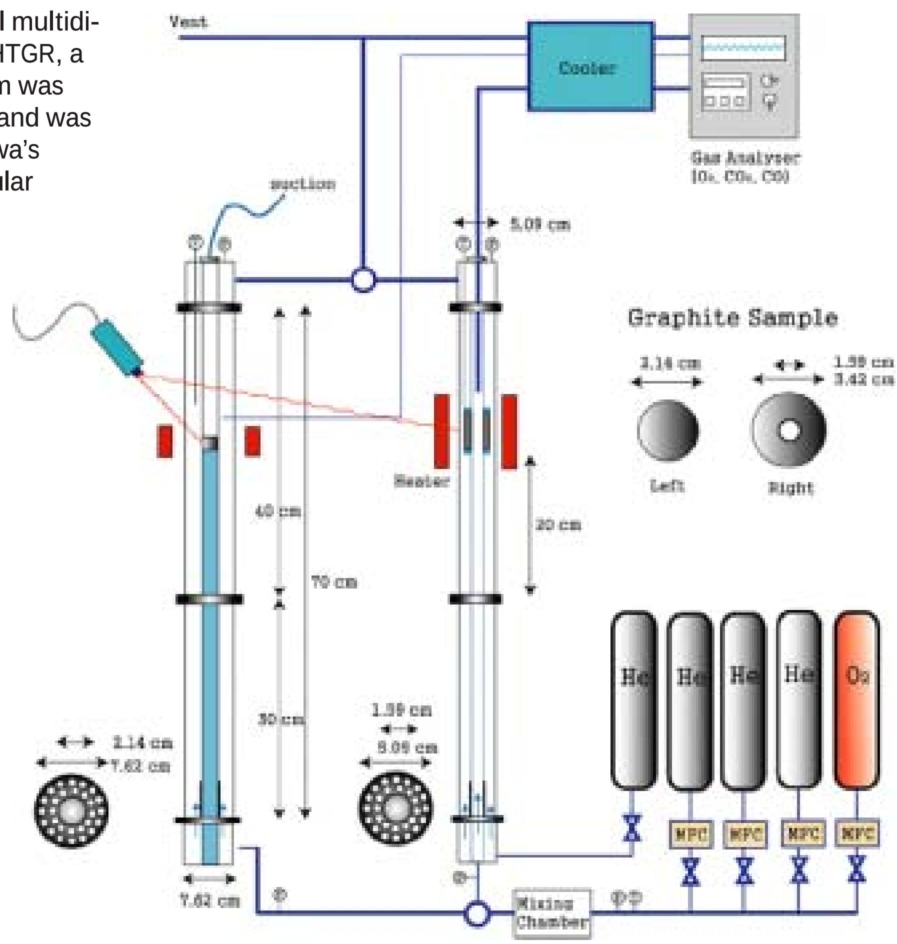


***Figure 2**. Schematic diagram of the air-ingress test facility.*

test, mass transfer modeling, validation test, and model upgrade. The experimental strategy also includes targeted results, experimental procedure, test conditions, and a test matrix for each step. The pretest 1 and 2 steps are currently being performed with the test facility. FLUENT6 is being used to perform a pretest calculation of the experiment. The detailed experimental strategy and significant results obtained until now are described in the technical section.

Task 4 — Improvement of system codes (INEEL). Staff implemented a preliminary molecular diffusion model into RELAP5/ATHENA. The model was tested through comparison with a theoretical solution from the literature. Initial assessments of the model against Japanese data were begun. Three additional noncondensable gases (O_2, CO_2, and CO) were implemented into the code that were needed to model graphite oxidation and molecular diffusion in the VHTGR. A phase equilibrium model was completed for mixtures of CO_2 and CO at various pressures and temperatures. This model was implemented into the MELCOR and will allow a more accurate calculation of fuel temperatures during hypothetical air ingress accidents.

Task 5 — Neutronic modeling (UM). With the objective to develop a state-of-the-art neutronics model for determining power distributions and decay heat deposition rates in a VHTGR core, effort was initiated during the reporting period to build MCNP Monte Carlo models both for full-core and assembly-level representations of the core. The full-core model comprises a homogenized cylindrical configuration with four radial and five axial regions. A hexagonal block-overlay version of the homogenized core is also under study toward the goal of eventually developing a hexagonal model of the whole core.

Task 6 — Verification and validation (INEEL and KAIST). As outlined in the proposal, this task will begin in FY 2004.

This project will benefit the development of several gas-cooled reactor concepts, including the HTGR, VHTGR, and fast gas reactor (FGR), by validating and verifying the computer codes that will be used in the safety evaluations. In addition, great benefits are expected from the development of the thermal hydraulic model for air ingress and its effect on graphite oxidation along with the development of state-of-the-art methodology for VHTGR neutronic analysis, which will provide accurate power distributions and decay heat deposition rates. The product from this research will ultimately benefit the hydrogen production initiative.

Other Activities

KAIST, INEEL, and SNU discussed the project progress and publications in February. Two separate technical papers were presented at the NURETH-10 conference that was held in Seoul, Korea.

Planned Activities

We will continue the tasks as outlined in the proposal.

Task 1 — We will perform various simulations to demonstrate the predictive capability of the new linked version, particularly for the cases which multidimensional effects are significant. The program structure will also be further improved for the user's convenience.

Task 2 — The experiment facility will be scaled based on design information from the VHTGR. Based on the scaling and the evaluation of properties of the SNU-RCCS, the design and manufacture of a RCCS experiment facility will be implemented. A series of experiments will be performed for the various ranges of temperature, roughness, oxidation, and fill gas compositions. The distribution of temperatures along the reactor vessel will be measured allowing reliable estimates of emissivity, which will provide verification data for cavity analysis.

Task 3 — Kinetic and mass transfer tests will be performed, as will graphite oxidation modeling. Two measurement techniques for reaction rate—gas detection method and mass detection method—will be compared and their uncertainty analyses performed.

Task 4 — The models that have been implemented into RELAP5/ATHENA and MELCOR will be benchmarked using data from Japan, Germany, and the air ingress experiments conducted by KAIST.

Task 5 — The first priority will be to verify and improve preliminary MCNP calculations obtained for a cylindrical model of the VHTGR core, where fuel blocks are homogenized into a single region. This will be followed by full implementation of a hexagonal block-overlay model for individual fuel blocks.

Task 6 — Validation and verification of RELAP5/ATHENA and MELCOR will start in the third quarter of FY 2004. This task includes validation of the RCCS model using data from SNU and comparison of calculated results with the CFD code being developed at KAIST.

International Nuclear Energy Research Initiative

Advanced Corrosion-Resistant Zirconium Alloys for High Burnup and Generation IV Applications

Principal Investigator (U.S.): Arthur Motta, Pennsylvania State University

Principal Investigator (ROK): Yeong Hwan Jeong, KAERI

Collaborators: R.J. Comstock, Westinghouse; J.S. Busby, University of Michigan; Y.S. Kim, Hanyang University

Project Number: 2003-020-K

Project Start Date: January 2003

Project End Date: September 2006

Reporting Period: January — September 2003

Research Objective

The objective of this collaboration among four institutions in the U.S. and in Korea is to demonstrate a technical basis for improving the corrosion resistance of zirconium-based alloys in more extreme operating environments (such as those in severe fuel duty cycles [high burnup, boiling, aggressive chemistry]) and to investigate the feasibility (from the point of view of corrosion rate) of using advanced zirconium-based alloys in a supercritical water environment. This technical basis is to be obtained by comparing the corrosion kinetics and examining the fine structure of oxide layers formed in model alloys. These model alloys are designed to isolate specific features of the microstructure thought to affect the formation of the protective oxide layer so that their effect on the corrosion rate can be studied individually. The key aspect of the program is to rationalize the differences in corrosion kinetics of the alloys through the differences in the structure and evolution of the protective oxide formed in each. To find these structural differences in the oxides, advanced characterization techniques (including submicron-beam synchrotron radiation [diffraction and fluorescence], cross-sectional transmission electron microscopy [TEM], transmitted light optical microscopy, electrochemical impedance spectroscopy [EIS] and nano-indentation) are used to characterize both the metal and the oxide so that these differences in oxide structure can be related to the original microstructure of the alloy.

The product of this research will be 1) a model of the rate of advancement of the oxide layer and the oxide transition process in the different alloys and 2) an assessment of whether Zr alloys have a realistic chance of surviving in the supercritical water environment. This model will allow a more mechanistically based development of Zr alloys for extreme environments, thus enhancing the ability to predict behavior and design alloys for higher burnup in light water reactors (LWRs). If the data obtained point to the possibility of using Zr alloys in the super-critical water reactor (SCWR), it opens the possibility that a great neutron economy benefit can be accrued relative to stainless steel cladding. Further, the development of the first-of-a-kind experimental techniques for advanced characterization of oxides and related analysis methods could be applied to other corrosion problems. Finally, involving graduate students in the nuclear materials area will contribute to forming the new generation of scientists and engineers in both countries.

Research Progress

The study consists of five tasks: 1) fabrication of model alloys, 2) autoclave testing, 3) testing in supercritical water, 4) characterization of alloys and oxide layers, and 5) data analysis and modeling.

Task 1, led by Westinghouse and KAERI, consists of selecting and fabricating a set of zirconium alloy corrosion samples designed to highlight the role of specific features in the alloy's microstructure (e.g., solute content or precipitate type) on the corrosion process. Twenty-six model Zr alloys were prepared during this period in the U.S. and Korea, in three groups: precipitate containing alloys, solid solution alloys, and standards. Some of the higher alloying element content alloys will be the first to be investigated in the supercritical water environment; it is thought they will behave better than standard Zr alloys in these conditions.

Tasks 2 and 3 are concerned with corrosion testing. Task 2 consists of performing lower-temperature corrosion testing in three autoclave environments (360°C pure water at saturation pressure, 360°C water containing 70 ppm Li [also at saturation pressure], and 500°C steam at 1500 psi), measuring corrosion weight gains to determine corrosion kinetics, and archiving samples at planned intervals for characterization. This is to be done at both Westinghouse and KAERI. Exchange of alloy samples between Republic of Korea and the U.S. will permit corrosion testing of a limited number of coupons that were fabricated in the other country to check the reliability of the data by generating corrosion results in the same environment but in different autoclave facilities. Task 3 consists of testing in supercritical water to be performed at University of Michigan and KAERI. The objective is to investigate the behavior of specially designed Zr alloys in supercritical water at high-temperature, measure weight gain to characterize corrosion kinetics, and archive samples for study.

Task 4 led by Penn State and KAERI, and in combination with Task 5, is the crux of the project, since this is where the differences in the oxide microstructure, which are believed to be responsible for the different corrosion kinetics in the various alloys, will be studied. It is necessary to characterize the microstructures of the as-fabricated alloys and the protective oxide. The microstructure characterization of the as-fabricated model has the goal of controlling the phases present, volume fraction precipitated, and precipitate size and solid solution to relate the microstructure of the oxide to that of the underlying metal. The characterization of the oxide is the main goal of this project, as we are trying to derive an understanding of the alloy-to-alloy differences in the corrosion process based on understanding the structure of the oxide.

Figure 1 shows the characterization of the metal alloy using synchrotron radiation at APS and of oxide using TEM. The advantage of using synchrotron radiation is that the very small volume fractions of precipitates (on the order of 0.1%) can be detected and quantified. This is shown on the left of Figure 1, where at the top (a) we see a powder diffraction profile of two variants of a model ZrFeCr alloy with high (0.4Fe, 0.2Cr) and low (0.2Fe, 0.1Cr) alloying element content created for this project. These alloys were prepared using two process temperatures (580°C and 720°C) to produce samples of two volume fractions (high and low alloying element content) and two precipitate sizes (high and low process temperature). When plotted in a linear scale, the diffraction pattern obtained shows no second phase peaks, but a blowup of the dashed region in (a) shown in (b) for the case where these two alloys were processed at 580°C and in (c) the case where the two alloys were processed at 720°C shows the precipitate peaks clearly.

The comparison of the graphs shows that the volume fraction increases from low to high alloying element content samples, while the precipitate size is constant for each temperature (35-38 nm at 580°C and 110 nm at 720°C calculated from line broadening) while the precipitate volume fractions are about the same for the two temperatures for a given alloying content. The crystal structure changes from cubic at 580°C to hexagonal at 720°C. Such detailed information was obtained about the microstructure of all the alloys fabricated and will help us separate what factors in the microstructure influence corrosion.

The right side of Figure 1 shows a cross-sectional TEM characterization of an oxide formed in Zr-Nb alloy, illustrating that oxide microstructure can be determined as a function of distance from the oxide-metal interface. For example, away from the oxide-metal interface, columnar grains (outlined in dashed lines) are present that are not present near the oxide-metal interface. Such TEM characterization is complemented by the analysis performed with synchrotron radiation submicron beam available at the Advanced Photon Source at Argonne National Laboratory for which we have obtained beam time for two years. Figure 2 shows an example of one such examination, where the submicron beam (0.25 µm) is scanned from the oxide-metal to the oxide water interface, obtaining the crystal structure, texture, and grain size of the oxide as a function of the distance from the oxide-metal interface. One example is the plot on the upper right corner, which shows the intensities of the tetragonal and monoclinic phase of ZrO_2 as a function of position, indicating a periodic pattern, with alternating monoclinic and tetragonal phases, also seen in transmitted light optical microscopy.[(a)]

Task 5 consists of the analysis and understanding of the corrosion kinetics data obtained in Tasks 2 and 3 and of the characterization data obtained in Task 4 to identify parameters important for oxide growth. Differences in corrosion behavior will be modeled and thus derive conclusions about advanced alloys.

(a) Structure of zirconium alloy oxides formed in pure water studied with synchrotron radiation and optical microscopy: relation to corrosion rate, Aylin Yilmazbayhan, Arthur T. Motta, Robert J. Comstock, George P. Sabol, Barry Lai and Zhonghou Cai, *Journal of Nuclear Materials*, 2003, in press.

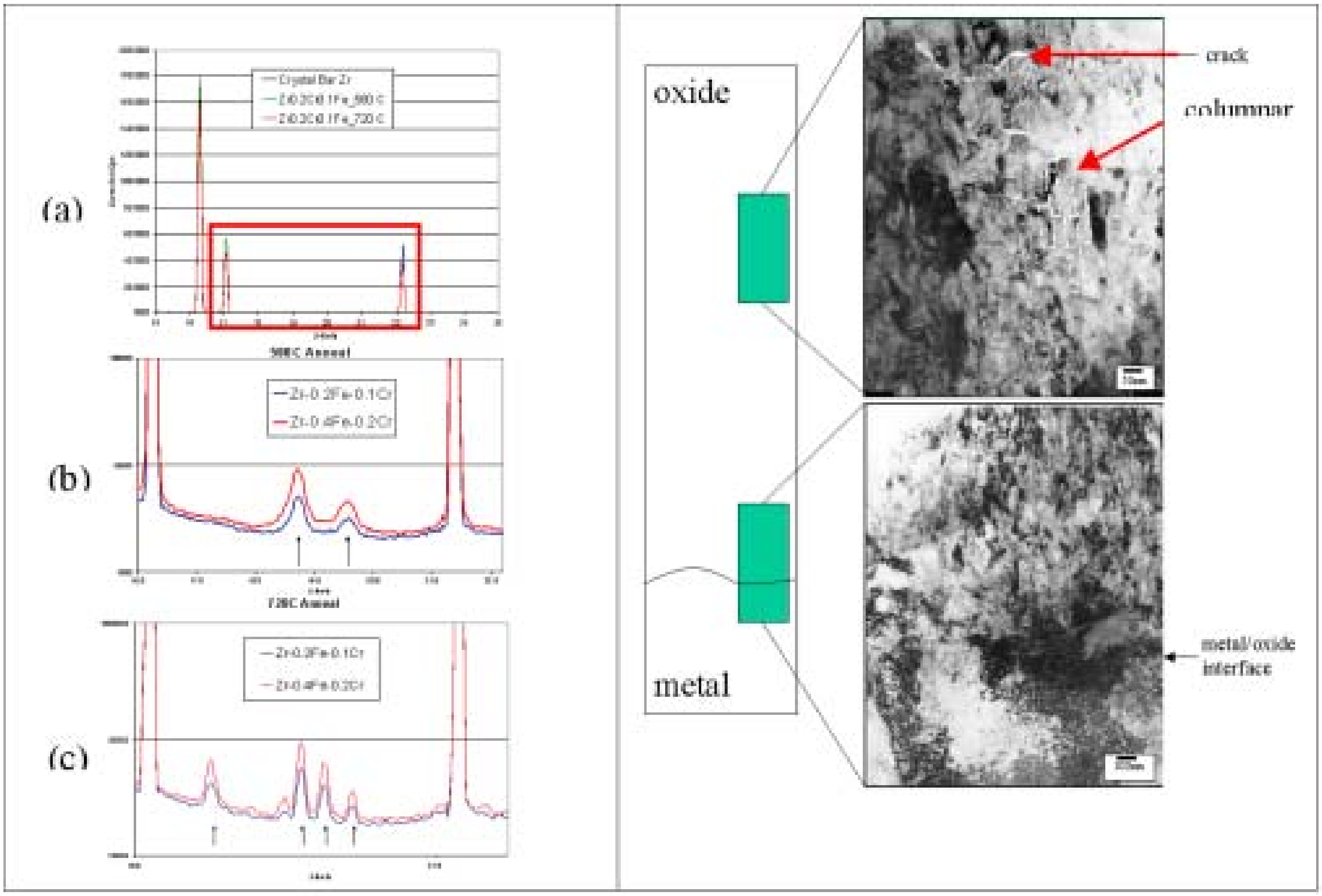
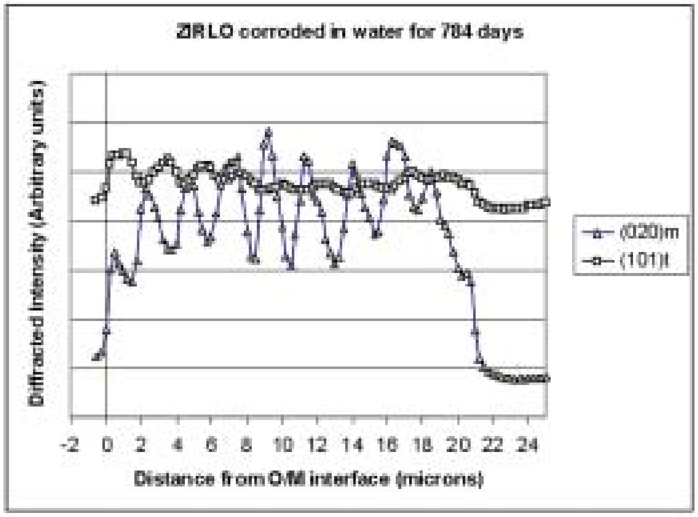


Figure 1. *(left) Synchrotron radiation diffraction patterns from as-fabricated ZrFeCr alloys of the compositions indicated shown in a linear scale (a), on a log scale for material processed at 580°C (b), and on a log scale for the material processed at 720°C (c). (right) Cross-sectional TEM micrograph of Zr-Nb alloy showing the regions near (bottom) and away from the oxide-metal interface (top), the last one showing columnar oxide grains and lateral cracks.*

Planned Activities

For the next year, it is planned to conduct extensive corrosion testing in the four environments envisaged. The 360°C water testing (slowest corrosion rate) and 360°C Li the pre-transition testing will be completed in the next year, while the higher-temperature (500°C in steam and supercritical water) will have more extensive corrosion. These oxides will be characterized using synchrotron radiation micro beam and TEM to identify phase evolution, grain size, texture, sub-oxide formation, and any elemental segregation. It is planned to characterize mechanical properties of the oxide using nano-indentation and the oxide electrical properties using electrochemical impedance spectroscopy (EIS). Then the observed microstructure will be related to the microstructure of the underlying metal and to the corrosion kinetics to start deriving a model of oxide growth.



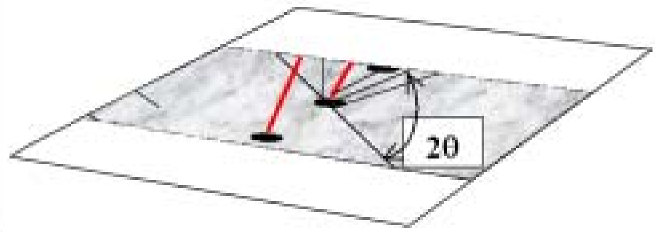


Figure 2. *Schematic drawing showing the geometry of data acquisition at the synchrotron beamline. The smallest dimension of the beam is about 0.2 micron. Both fluorescence and diffraction data are acquired simultaneously. The figure on the top left shows peak intensities for the (101) tetragonal oxide peak, and the (020) monoclinic oxide peak for an exposure of a zirconium alloy in 360°C water for 784 days, as a function of distance from the oxide-metal interface, indicating a periodicity in the corrosion process related to the oxide transition.*

International Nuclear Energy Research Initiative

Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte

Principal Investigator (U.S): James J. Laidler, Argonne National Laboratory (ANL)

Principal Investigator (ROK): Seong Won Park, Korea Atomic Energy Research Institute (KAERI)

Collaborators: University of Illinois at Chicago (UIC)

Project Number: 2003-024-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

Reporting Period: January to October 2003

Research Objective

The objective of this ANL and KAERI-led collaborative project is to develop advanced structural materials for use in a new technology for the electrolytic reduction of spent nuclear fuel in a molten salt electrolyte. The proposed effort will include the selection and testing of commercial alloys and ceramics as well as the engineering and testing of customized materials systems. The principal objectives of this project are (1) to assess and select candidate materials for service in the electrolyte reduction process, and (2) to develop new candidate material systems (e.g., functional barrier coatings) for service in the electrochemical reduction process vessel.

Research Progress

Research progress for 2003 began with a joint coordinating meeting held at ANL in Argonne, IL, on March 27, 2003, with ANL and (KAERI) personnel in attendance. The objective of the meeting was to complete an initial data exchange and discuss the experimental approach and areas of technical expertise relevant to this project. Subsequent to this meeting, we developed an initial list of high temperature alloys based on a combination of high temperature strength and corrosion resistance properties. This list, along with the schematic of our testing apparatus, was sent to Dr. Seong Won Park at KAERI on June 27, 2003. The alloy compositions are shown in Table 1A and 1B.

Previous immersion corrosion studies in hot molten salts, conducted at both ANL-East and ANL-West, indicated that preferential dissolution of container material constituents, adjacent sample constituents, or compounds from any material in contact with the molten salts, may contribute to chemical reactions related to corrosion mechanisms. For this reason, we have designed our testing apparatus to contain three independent vessels, each with a gas sparge tube and sample hanger, to eliminate any spurious sources of corrosion. The completed testing apparatus is shown in Figure 1.

The expected conditions in the electrochemical process vessel during operations are approximately 650C, 3% lithium oxide in lithium chloride, and 10% oxygen. However, conditions may vary and the temperature, concentration of lithium oxide, and concentration of oxygen may be greater than or less than these values. We expect to utilize parameters in our testing methods which will reflect these fluid conditions.

Corrosion testing began on July 15, 2003, in an argon atmosphere of a glovebox with the initiation of the 3 day testing period at 675ºC in 6w% Li_2O in LiCl with 10% oxygen flowing at 2 mL/min in the molten salt. Our sample coupons were immersed to half their length in the molten salt mixture to assess the corrosion effects of not only submersion, but also exposure of the alloys to the salt/oxygen vapors. Severe corrosion was evident after the 3 day test period for Haynes 556, Inconel 601 and 690 alloys, therefore, they were not subjected to further testing. Metallographic images of the microstructure of Inconel 690 are shown in Figure 2. Micrographs (a) and (c) were taken at 200X and (b) and (d) were taken at 500X. Micrograph (a) is an image of a sample area suspended above the level of the molten salt and contacted only by molten salt vapor and escaping oxygen gas. The corrosion scale is thin and adherent, and represents a single corrosion front with uniform penetration to ~100-120m. The corrosion layer appears porous. Micrographs (b), (c), and (d) are images of sample areas immersed into the molten salt. The corrosion scale present has areas of breakage and separation from the specimen and may be a consequence of the influence of the thickness of the surface scale, motion of the molten salt, and thermal expansion

Table 1a.

Alloy	Ni	Cr	Co	Mo	Fe	Al	Mg	W	C	O	Mn	Ti	Si
Haynes HR-160	36.8	28.3	30.8	< 0.05	< 0.1	0.09		< 0.1	0.05		0.45	0.53	2.67
Haynes 556	20.87	21.62	18.55	2.83	33.6	0.13		2.37	0		0.94		0.34
Haynes 230	58.56	22.45	0.17	1.42	1.54	0.34	0.008	14.23	0.1		0.52	< 0.01	0.38
Inconel 600	75.09	14.66	< 0.059		9.69	< 0.172			0.02		0.28	< 0.31	0.04
Inconel 601	60.72	22.16			14.83	1.26			0.03		0.27		0.37
Inconel 690	59.08	30.03	0.037		9.79	0.32			0.026		0.13	0.3	0.25
IN MA-754	77.8	20.21			0.27	0.32			0.05	0.32		0.44	
Nickel 200 (N200)	99.64				0.05				0.08		0.19		0.03

Table 1b.

Alloy	Zr	Nb	B	Cu	Ta	La	V	N	S	P	Y2O3
Haynes HR-160		< 0.05		< 0.01					<0.002	0.002	
Haynes 556	< 0.01	< 0.05	< 0.002		(+Cb)*0.75	0.034		0.16	<0.002	<0.008	
Haynes 230		< 0.05	0.002	0.05	< 0.1	0.017	< 0.05		<0.002	0.005	
Inconel 600		0.02		0.2	< 0.01				<0.001	<0.009	
Inconel 601				0.36					0.001		
Inconel 690				0.03					<0.001	0.009	
IN MA-754									0.002		0.59
Nickel 200 (N200)				0.01					0.001		
					* Nb						

a) Side View *(b) Top View*

Figure 1. *The Corrosion Test Apparatus for Molten Salt/ Oxygen Vapor Exposures: (a) Side view of assembled test apparatus on glovebox floor before insertion into the heated 10-in furnace well in a controlled atmosphere glovebox and (b) Top view showing the radial position of the three independent salt bath beakers with sample hangers (dark tubes) and gas sparge tubes (white tubes).*

coefficient mismatch between the scale product and the metal substrate. In this instance, the corrosion has penetrated below the surface of the corrosion scale. Resistance to oxidizing hot corrosion is a function of one or more alloy additions which oxidize selectively relative to the base metal. Our results to date indicate that while Ni and Cr appear essential for corrosion resistance by the formation of NiO and Cr_2O_3, Fe content above ~ 1w% has a deleterious effect on alloy performance. Alloying elements present which appear to enhance corrosion resistance are Co, Si, W, and Y_2O_3.

The subcontract with UIC is being awarded for sample preparation and metallography, scanning electron microscopy, and other post-test analyses of the corrosion samples for microstructure evaluation. In addition to this work, UIC has performed heat treatment of the as-received samples for Ni-200, Haynes 230, and Inconel 690 at 675ºC for 5 days in air to determine the microstructural changes related to heat alone. The results will be compared to the effects from both heat and exposure to molten salts, and molten salt vapor/ oxygen gas. It is expected that all as-received samples will undergo heat treatment.

Our test plan calls for 9 day testing to be performed at 3 different temperatures (625ºC, 675ºC, 725ºC) to determine the activation energy relevant to corrosion reaction kinetics. The 9 day testing at 625ºC has been completed for the remaining 5 alloys. The alloys are Haynes HR 160 and 230, Inconel 600 and MA754, and Nickel 200. Microstructural examination is underway.

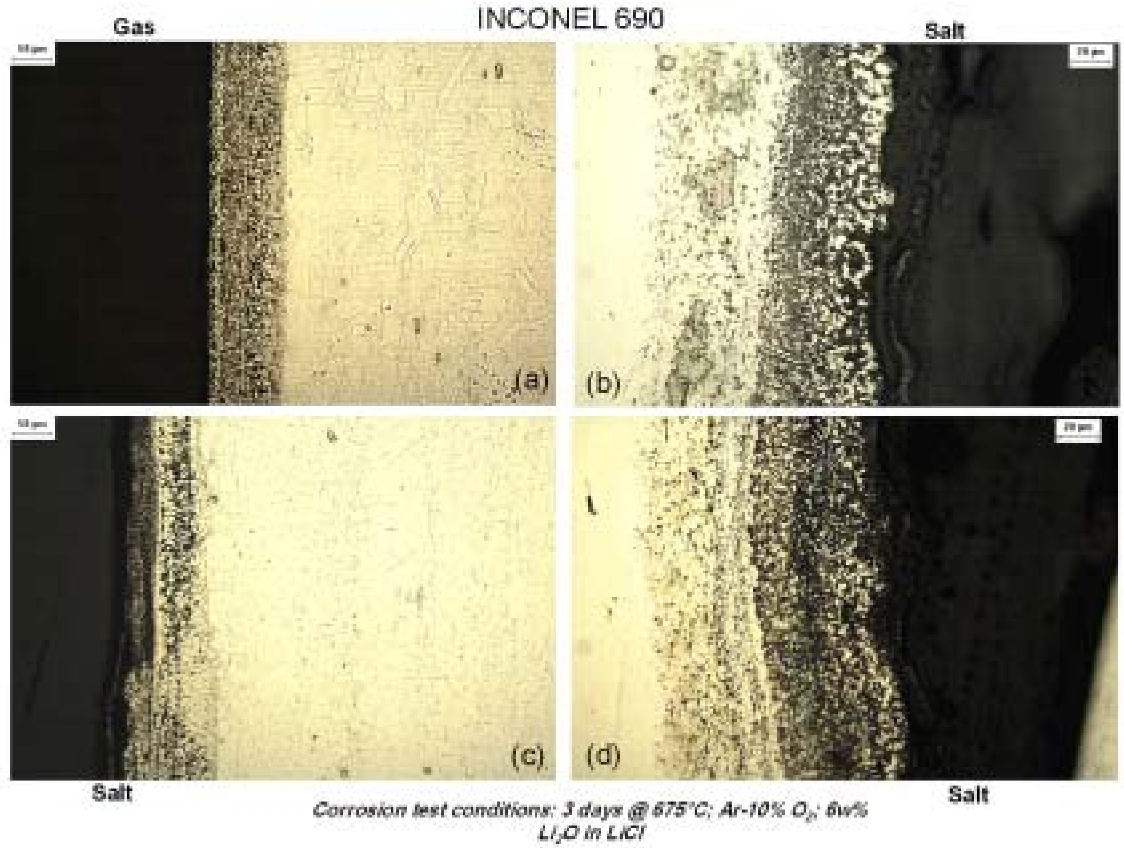


***Figure 2**.*

However, Inconel 600, which contains ~10w% Fe, was judged as failing due to the severe corrosion and delamination present at the conclusion of the test. The 9 day testing at 675ºC in underway.

Planned Activities

Research efforts in 2004 will focus on continued assessment of alloy additions for use in oxidizing hot corrosion conditions and evaluation of coating materials for “functional barrier” protection of the metal substrate. Our prime concern is to satisfy the mechanical strength requirement for structural material used in the electrochemical reduction process, coupled with the maximum corrosion resistance available among the commercial alloys before the application of a protective coating. While the literature suggests Cr_2O_3 scale-forming alloys are superior to those which form Al_2O_3 scales in hot corrosion, Al_2O_3 scales provide superior oxidation resistance. Since our initial alloy samples all contain Cr but very small amounts of Al, we plan to test 2 Ni-based alloys containing ~ 5w% Al. The alloys are IN 100 and Haynes 214. In addition, we will test a Co-based alloy, Haynes 188. Also, scale adherence and resistance to spalling and/or cracking is an important factor in oxidation resistance and can be greatly improved by adding “active” alloy elements such as lanthanum, cerium, and yttrium. Inconel MA754 contains yttrium and appears to have performed well macroscopically to date. Testing of alloys containing the recommended composition of lanthanum and/or cerium will be undertaken if available.

Selected ceramic candidates have been under consideration for testing but have not been finalized. Assessment of the heat generation expected during the electrochemical reduction process will be required to determine the type of barrier coating or surface modification needed. Materials most compatible with the metal alloy substrate and meet the needs of the heat assessment will be chosen and subjected to testing.

Appendix C

U.S./OECD Collaboration Project Summary

International Nuclear Energy Research Initiative

Project #	Title
2002-001-N	Melt Coolability and Concrete Interaction (MCCI) Program

International Nuclear Energy Research Initiative

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Research Objectives

Although extensive research has been conducted over the last several years in the areas of melt coolability and core-concrete interaction, two important issues warrant further investigation. The first concerns the effectiveness of water in terminating a core-concrete interaction by flooding the interacting masses from above, thereby affecting a quench of the core debris and rendering it permanently coolable. The second issue concerns long-term, two-dimensional concrete ablation by a prototypic core oxide melt. The goal of the MCCI research program is to conduct reactor material experiments and associated analysis to achieve the following two technical objectives:

1. Resolve of the ex-vessel debris coolability issue through a program which focuses on providing both confirmatory evidence and test data for coolability mechanisms identified in integral debris coolability experiments.

2. Address remaining uncertainties related to long-term, two-dimensional, core-concrete interaction under both wet and dry cavity conditions.

Achievement of these two objectives will lead to improved accident management guidelines for existing plants, and also better containment designs for future plants. In terms of meeting these objectives, the workscope for the second year of the program was defined to consist of: 1) conducting three additional Small Scale Water Ingression and Crust Strength (SSWICS) tests, and 2) carrying out a large-scale, 2-D, molten Core-Concrete Interaction (CCI) experiment.

Research Progress

The purpose of the SSWICS tests is to carry out fundamental analysis and experiments to determine the extent that water is able to ingress into cracks and fissures in corium during cooldown, thereby augmenting the otherwise conduction-limited cooling process. This is a generic phenomenological issue, applicable to both in-vessel and ex-vessel accident sequences. The experiment approach is to generate a prototypic core melt composition at ~ 2100 ÚC through an exothermic chemical reaction, and then flood the corium melt pool with water from above. The steam formed as a result of the interaction is condensed in an instrumented quench system; the melt/water heat flux is evaluated based on the steam condensation rate. A schematic of the SSWICS test section is shown in Figure 1, while a photograph showing key facility features is provided in Figure 2. The corium melts are not heated during the quench process. On that basis, the water ingression rate (or dryout heat flux) can readily be determined by comparing the actual corium cooling rate with well-known analytical solutions for the case of conduction-limited cooling and solidification of liquids. To date, five successful experiments have been carried out at system pressures ranging from atmospheric pressure up to 4 bar absolute. The tests are conducted with a 15 cm deep melt pool (typical melt mass is 75 kg) within a 30 cm ID refractory test section. The tests have primarily parameterized on melt composition. In general, the compositions are representative of a fully oxidized pressurized water reactor (PWR) in terms of the core and cladding contents; concrete levels in the melts have ranged from 8 to 25 wt % concrete. Both limestone/common sand and siliceous concrete types have been addressed in the test matrix.

The results of the SSWICS tests have shown that water does indeed penetrate into the material during cooldown, augmenting the otherwise conduction-limited cooling process. A photograph of one of the test specimens after radial sectioning is provided in Figure 3. The cracks in the material provide the pathway for water ingression. The normalized heat flux data obtained from the first four SSWICS experiments is compared with the conduction-limited solution in Figure 4. The data indicates that water ingression augmentation of the debris cooling rate diminishes as more concrete is added to the melt. Thus, from an accident management viewpoint, the data suggests that early water addition is important in terms of maximizing the coolability of the core debris.

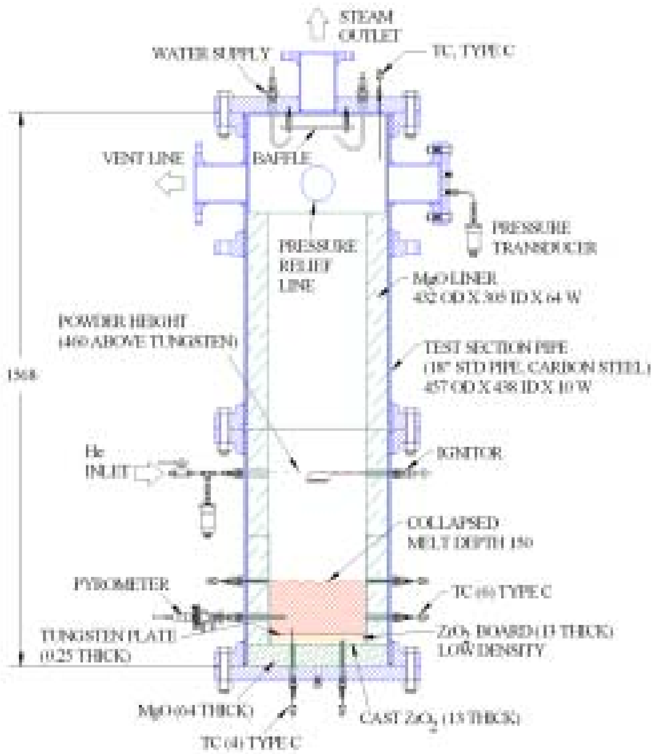

***Figure 1**. SSWICS Test Section*

***Figure 2**. SSWICS Test Facility*

***Figure 3**. Radially Sectioned Corium Sample*

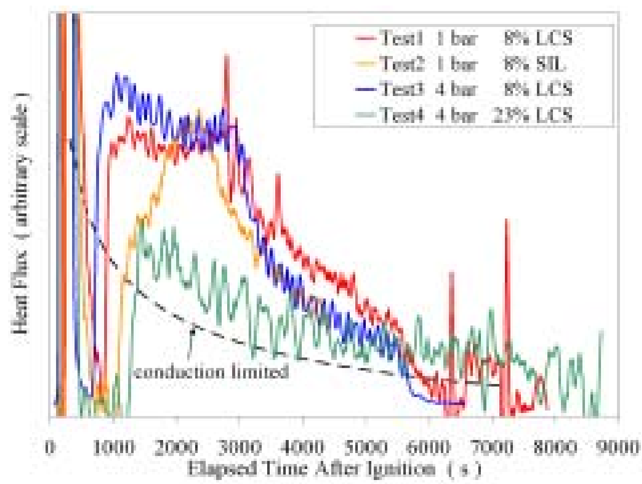

***Figure 4**. Corium/Water Heat Flux*

The large-scale CCI tests are being conducted to provide data in several areas to support the overall project objectives, including: 1) lateral vs. axial power split during dry core-concrete interaction, 2) integral debris coolability data following late phase flooding, and 3) data regarding the nature and extent of the cooling transient following breach of the crust formed at the melt-water interface. In terms of this year's activities, the facility was assembled in the test cell, and the first of two planned tests, CCI-1, was successfully conducted. This experiment investigated the interaction of a fully oxidized 400 kg PWR core melt, initially containing 8 wt % calcined siliceous concrete, with a specially designed two-dimensional siliceous concrete test section. A schematic of the facility is provided in Figure 5, while details of the 2-D concrete test section are shown in Figure 6. The core-concrete interaction results indicate that radial ablation is a key element of the overall cavity erosion process. Representative melt temperature data obtained during the test is provided in Figure 7. Late-phase flooding was found to cause a significant increase in the upwards heat flux from the debris. The normalized debris cooling rate is shown in Figure 8. Force measurements made during the crust loading sequence indicate that the crust material is structurally quite weak. Crust breach by an insertable lance caused a dramatic increase in the debris cooling rate. In particular, a large melt eruption occurred, causing the average debris cooling rate to significantly exceed the CHF limitation. These findings indicate that transient crust breach events at plant scale will lead to significant increases in the debris cooling rate.

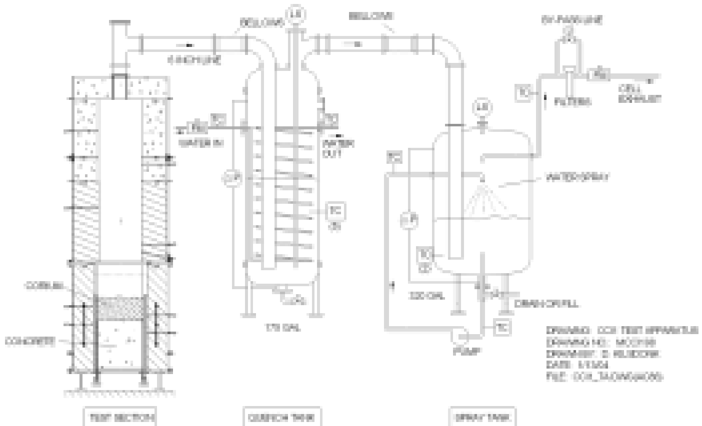

***Figure 5**. CCI Test Facility*

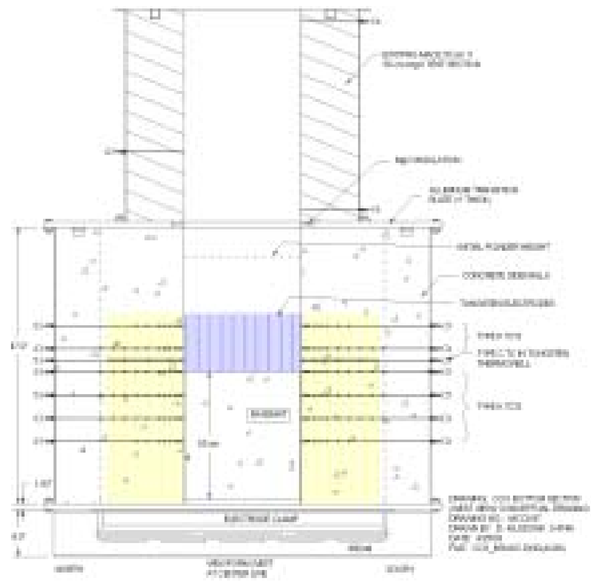

***Figure 6**. CCI-1 Test Section*

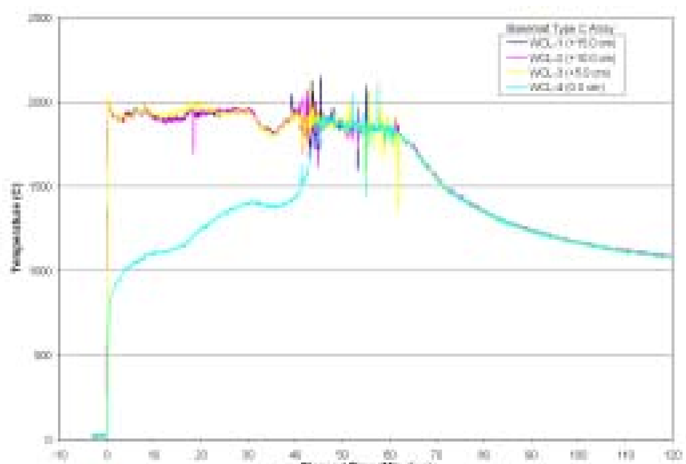

***Figure 7**. CCI-1 Melt Temperature Data*

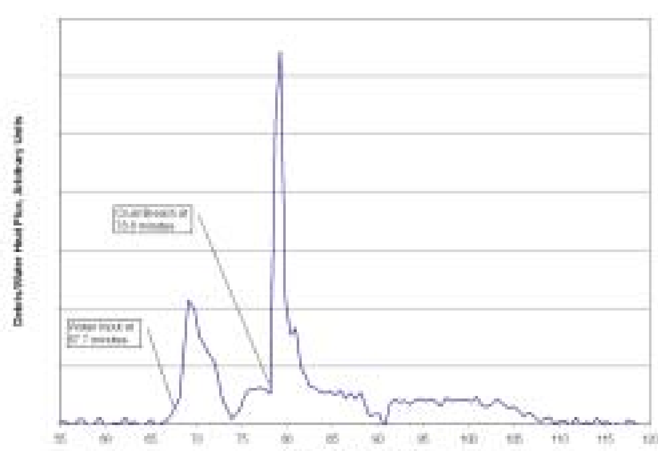

***Figure 8**. CCI-1 Corium/Water Heat Flux*

In terms of reporting, full-length data reports have been published for all tests conducted to date. A paper documenting the results of the SSWICS tests has been submitted to *Nuclear Engineering and Design,* while the results of CCI-1 were documented in a paper submitted to ICAPP '04.

Planned Activities

Next year's activities will include additional reactor material tests and supporting analyses focused on achieving the project's two main objectives. The sixth SSWICS test will be conducted to investigate water ingression cooling of a fully oxidized PWR core melt containing 15 wt % siliceous concrete at atmospheric pressure. In addition, CCI-2 will be conducted. This test will examine 2-D cavity erosion and debris coolability of a fully oxidized PWR core melt interacting with a limestone/common sand concrete crucible. The project will also initiate preparations for large-scale separate effects tests to obtain data on the melt eruption cooling mechanism.