International Nuclear Energy Research Initiative

2008

Annual Report

Disclaimer

Available to DOE, DOE contractors, and the public from the

U.S. Department of Energy
Office of Nuclear Energy
1000 Independence Avenue, S.W.
Washington, D.C. 20585

Printed with soy ink on recycled paper.

Foreword

The International Nuclear Energy Research Initiative (I-NERI) was established by the U.S. Department of Energy, Office of Nuclear Energy (DOE-NE) in fiscal year (FY) 2001 to conduct research and development (R&D) with international partners in advanced nuclear energy systems development. This annual report provides an update on the FY 2008 accomplishments under the I-NERI collaborations.

I-NERI supports scientific and engineering R&D linked to three of the principal research programs sponsored by DOE-NE: the Generation IV Nuclear Energy Systems Initiative, the Advanced Fuel Cycle Initiative, and the Nuclear Hydrogen Initiative. As international collaboration is an important part of DOE-NE's R&D efforts, I-NERI is designed to foster such international partnerships to address key issues affecting the future global use of nuclear energy. Through international collaboration, DOE can effectively leverage its limited economic resources, more readily expand the knowledge base of nuclear science and engineering, and establish valuable intellectual relationships with researchers from other countries. Forging these partnerships enhances the United States' participation in the global nuclear community and helps build an international consensus on critically important issues such as expanding the benefits of nuclear power and designing proliferation resistance into advanced nuclear systems.

Current I-NERI collaborators include Brazil, Canada, the European Union (EU), France, Japan, and the Republic of Korea. In FY 2008, DOE initiated seven more projects: three with Canada, one with France, and three with the Republic of Korea. Seven projects concluded in FY 2008, involving collaborations with Brazil, Canada, EU, France, and the Republic of Korea. This *I-NERI 2008 Annual Report* provides a description of the new projects, along with a comprehensive summary of the progress of each collaborative research project initiated since FY 2005.

R. Shane Johnson
Acting Assistant Secretary for Nuclear Energy
U.S. Department of Energy

Table of Contents

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

1.0 Overview and Program History

The International Nuclear Energy Research Initiative (I-NERI) is an international, research-oriented initiative that supports the advancement of nuclear science and technology in the United States and the world. I-NERI promotes bilateral scientific and engineering research and development (R&D) with other nations. Innovative research performed under the I-NERI umbrella addresses key issues affecting the future use of nuclear energy and its global deployment by improving cost performance, enhancing safety, and increasing proliferation resistance of future nuclear energy systems. A link to the program can be found at the U.S. Department of Energy, Office of Nuclear Energy (DOE-NE) website: http://www.nuclear.gov.

This *I-NERI 2008 Annual Report* serves to inform interested parties about the program's progress of current collaborative research projects. Following is an overview of each section:

- Section 1 discusses background information on the events that led to the I-NERI program's creation.
- Section 2 presents the countries and international organizations participating in current I-NERI collaborative agreements. This section also provides an overview of program goals and objectives, a work scope summary for the three constituent program areas, and an overview of funding since the program's inception.
- Section 3 provides a summary of recent programmatic accomplishments and a list of participating organizations.
- Section 4 presents details of the R&D work scope for current I-NERI collaborative projects with Brazil, Canada, the European Unioin (EU), France, Japan, and the Republic of Korea, respectively. For each participant, the report presents an index of projects and a summary of technical accomplishments achieved in fiscal year (FY) 2008. There are no current projects with the Organisation for Economic Co-operation and Development (OECD).

At the end of the document, there is an index of I-NERI projects listed under the FY in which they were awarded.

DOE-NE created I-NERI in FY 2001 in response to recommendations provided by the President's Committee of Advisors on Science and Technology, who issued a report entitled *Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation.* This report promoted "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 stated, "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 initial focus of the I-NERI program was on developing international collaborations, program planning, and project procurements. Since the inception of the program, 88 projects have been awarded, 54 of which have been completed. Both the number of awards and the consistency of project achievement demonstrate I-NERI's success in fostering international collaboration.

Table 1 presents the breakdown, by FY, of the number of project awards for each partnering country.

Collaborator	FY 01	FY 02	FY 03	FY 04	FY 05	FY 06	FY 07	FY 08	Total
France	4	1		11		3	1	1	**21**
Republic of Korea		6	5	6	4	4	7	3	**35**
OECD-NEA		1							**1**
EURATOM				8	2	5			**15**
Canada				7			2	3	**12**
Brazil					1	1			**2**
Japan					1	1			**2**
Total	**4**	**8**	**5**	**32**	**8**	**14**	**10**	**7**	**88**

Table 1. Number of I-NERI projects awarded.

2.0 I-NERI Program Goals and Objectives

I-NERI's mission is to sponsor innovative scientific and engineering R&D in cooperation with international partnering countries. The I-NERI program is designed to foster closer collaboration among international and U.S. researchers, improve communications, and expand the sharing of nuclear research information. The I-NERI program has established the following overall objectives:

- To develop advanced concepts and scientific breakthroughs in nuclear energy and reactor technology in order to address and overcome the principal technical and scientific obstacles to expanding the global use of nuclear energy
- To promote bilateral and multilateral collaboration with international agencies and research organizations to improve the development of nuclear energy
- To promote and maintain a nuclear science and engineering infrastructure to meet future technical challenges

Through the I-NERI program, DOE-NE has coordinated wide-ranging discussions among governments, industry, and the worldwide research community regarding the development of Generation IV nuclear energy systems, advanced fuel cycles, and nuclear hydrogen technologies. The graphic on the following page illustrates key features of the I-NERI program.

International Agreements

In order to initiate an international collaboration, a government-to-government agreement must first be in place. These agreements are the vehicles to conduct R&D under Generation IV, AFCI, and NHI programs.

To date, DOE has implemented bilateral I-NERI collaborative agreements with the following international partners:

1) Commissariat à l'énergie atomique (CEA) of France
2) Republic of Korea Ministry of Science and Technology (MOST), predecessor to the Ministry of Education, Science and Technology (MEST)
3) The Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD)
4) European Atomic Energy Community (EURATOM)
5) Department of Natural Resources Canada (NRCAN) and Atomic Energy of Canada Limited (AECL)
6) Brazilian Ministério da Ciência e Tecnologia (MST)
7) Agency of Natural Resources and Energy (ANRE) and the Ministry of Education, Culture, Sports, Science, and Technology (MEXT) of Japan

I-NERI's bilateral agreements enable U.S. researchers to establish international R&D collaborations to begin developing next-generation nuclear energy systems. Negotiations are under way to establish a new bilateral agreement with the Republic of South Africa. DOE supports multilateral R&D collaborations under the Generation IV International Forum (GIF) and the international Global Nuclear Energy Partnership. Please visit their websites: http://www.gen-4.org and http://www.gneppartnership.org.

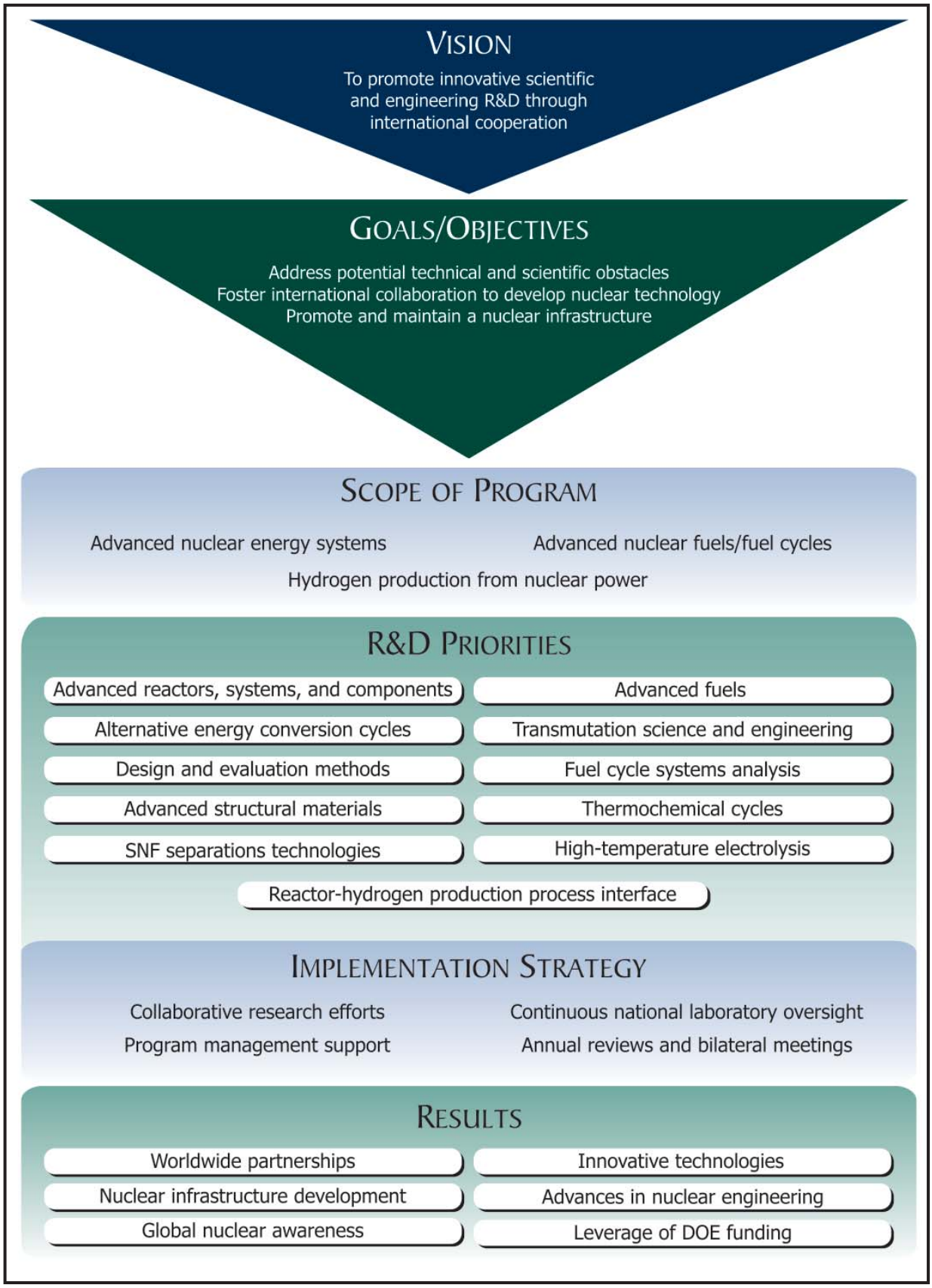
Funding

I-NERI is an important vehicle for enabling international R&D in Generation IV, AFCI, and NHI technology on a leveraged, cost-shared basis. Each country in an I-NERI collaboration provides funding for their respective project participants. The United States funds I-NERI projects through its national laboratories. U.S. contribution is based upon current-year budgets. Actual cost-share amounts are determined jointly for each selected project. The program's goal is to achieve approximately 50-50 matching contributions from each partnering country.

To date, I-NERI participants have committed a total R&D investment of $217.4 million: $116.8 million contributed by the United States and $100.6 million by international collaborators. International investment consists of

- $19.2 million from Canada
- $30.2 million from France
- $33.9 million from the Republic of Korea
- $2.7 million from Japan
- $12.3 million from the European Union
- $2.3 million from Brazil

I-NERI projects typically last three years and are funded annually by the Generation IV, AFCI, and NHI programs. Funding provided by the United States may be spent only by U.S. participants.


VISION
To promote innovative scientific and engineering R&D through international cooperation
GOALS/OBJECTIVES
Address potential technical and scientific obstacles
Foster international collaboration to develop nuclear technology
Promote and maintain a nuclear infrastructure
SCOPE OF PROGRAM
Advanced nuclear energy systems
Advanced nuclear fuels/fuel cycles
Hydrogen production from nuclear power
R&D PRIORITIES
Advanced reactors, systems, and components
Advanced fuels
Alternative energy conversion cycles
Transmutation science and engineering
Design and evaluation methods
Fuel cycle systems analysis
Advanced structural materials
Thermochemical cycles
SNF separations technologies
High-temperature electrolysis
Reactor-hydrogen production process interface
IMPLEMENTATION STRATEGY
Collaborative research efforts
Continuous national laboratory oversight
Program management support
Annual reviews and bilateral meetings
RESULTS
Worldwide partnerships
Innovative technologies
Nuclear infrastructure development
Advances in nuclear engineering
Global nuclear awareness
Leverage of DOE funding

Work Scope

The work scope of all current I-NERI projects is directly linked to the scientific and engineering needs of the Generation IV, AFCI, and NHI programs. Candidate project work scopes are jointly developed by the United States and the collaborating country based on conformance with the bilateral agreement and current Generation IV, AFCI, and NHI programmatic needs. Following is an overview of the individual work scopes for NE's three R&D programs.

Generation IV Nuclear Energy Systems Initiative. The Generation IV program is developing next-generation nuclear energy systems that offer advantages in the areas of economics, safety, reliability, and sustainability, with a goal of commercial deployment by the year 2030. Using a technology roadmap created by GIF member countries, the following six reactor concepts were deemed most promising: the molten salt reactor, lead-cooled fast reactor (LFR), gas fast reactor (GFR), supercritical water reactor (SCWR), sodium fast reactor (SFR), and very high-temperature reactor (VHTR). Current U.S. research priorities are focused on the VHTR and SFR. The VHTR concept is being pursued as the Next Generation Nuclear Plant (NGNP), designed for the economical and safe production of hydrogen and electricity without greenhouse gas emissions.

The Generation IV program has eight technology goals:

Sustainability

1) Generate energy sustainably and promote long-term availability of nuclear fuel
2) Minimize nuclear waste and reduce the long-term stewardship burden

Safety and Reliability

3) Excel in safety and reliability
4) Have a very low likelihood and degree of reactor core damage
5) Eliminate the need for offsite emergency response

Economics

6) Have a life cycle cost advantage over other energy sources
7) Have a level of financial risk comparable to other energy projects

Proliferation Resistance and Physical Protection

8) Be a very unattractive route for diversion or theft of weapons-usable materials and provide increased physical protection against acts of terrorism

Advanced Fuel Cycle Initiative. The mission of AFCI is to enable the safe, secure, economic, and sustainable expansion of nuclear energy by conducting research, development, and demonstration focused on nuclear fuel recycling and waste management to meet U.S. needs. In this way, nuclear energy can satisfy the growing energy needs of the United States while improving management of waste requiring geologic disposal. These technologies may also be of value to Generation IV.

Research under this initiative focuses on recycling, fuel treatment, and conditioning technologies that have the potential to dramatically reduce the quantity, radiotoxicity, and thermal content of used nuclear fuel materials requiring geological disposal, thus greatly expanding a repository's effective capacity.

Nuclear Hydrogen Initiative. The goal of the NHI program is to demonstrate the commercial-scale, economic viability of hydrogen production using nuclear energy by the year 2020. This initiative conducts R&D on enabling technologies, demonstrates nuclear-based hydrogen production technologies, studies potential hydrogen production schemes, and develops deployment alternatives to meet future needs for increased hydrogen consumption.

Hydrogen can be produced using a variety of technologies, each of which has its advantages and limitations. The primary advantage of nuclear energy production technologies is the potential ability to produce hydrogen in large quantities at a low cost without emitting greenhouse gases or consuming fossil fuel resources.

Figure 1 shows the distribution of projects by each of the three major program areas since the program's inception. (Prior to 2004, all of the projects were related strictly to Generation IV.)

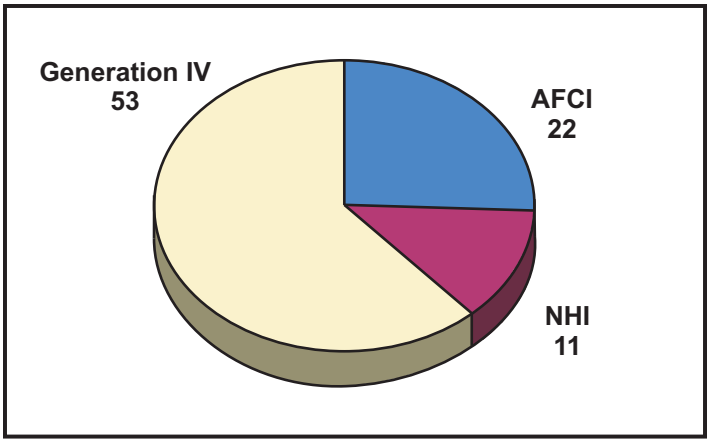


Figure 1. Project distribution by program area (FY 2001 to FY 2008).

3.0 I-NERI Program Accomplishments

I-NERI collaborative projects produce findings that would take the United States alone far more time and money to accomplish. The U.S. investment in I-NERI is matched nearly dollar for dollar, with the United States again providing leadership in both funding and research, thus helping to establish the U.S. role within the international community. Moreover, creating an international infrastructure brings multiple perspectives and priorities together to address shared obstacles. I-NERI collaborative efforts have contributed greatly to industry knowledge, furthering the ability to overcome technological challenges to nuclear energy's continued—and enhanced—role in the world energy platform.

For example, the NHI program is investigating development of the copper-chloride (Cu-Cl) cycle as part of the alternative thermochemical cycle program. I-NERI has joined the efforts of the two university programs associated with this thermochemical cycle research program—one in Canada and one in the United States. Successes thus far include updating the conceptual process design, finding ways to increase the yield of Cu_2OCl_2, and uncovering a method to improve efficiency of the hydrogen production reaction during constant-current electrolysis. (For details about this project, see Section 5.2.) The three other current NHI projects are investigating the sulfur-iodine thermochemical cycle, one of the most promising for large-scale hydrogen production; issues associated with solid-oxide electrolysis cells used to produce hydrogen from steam; and the development of a preliminary design for an integrated complex comprising a nuclear reactor, electrolytic hydrogen plant, and bitumen upgrader.

The remaining ongoing projects are evenly divided between the AFCI and Generation IV programs. The 13 Generation IV teams include two investigating construction and design of SFRs, one developing a pre-conceptual design for an LFR, two researching fuel coatings, one evaluating use of the supercritical CO_2 Brayton cycle, and four studying alloys for use in Generation IV designs—mostly the VHTR. The remaining three projects are also conducting research related to the VHTR, with one team developing advanced simulation techniques, another studying core bypass flow, and the last modeling air ingress accidents and thus providing data crucial to preventing serious safety problems.

Within the 13 AFCI project teams are three teams finding ways to improve fuel performance, thus increasing efficiency and reducing waste; two specifically focused on safety systems and safety validation; three researching transmutation and related fuels; one examining alloys for the SFR; and four examining fuel separations, fuel recycling, and new waste forms. All clearly meet the agenda of finding new fuel resources and systems for proliferation-resistant nuclear power.

DOE plans the following international activities for FY 2009:

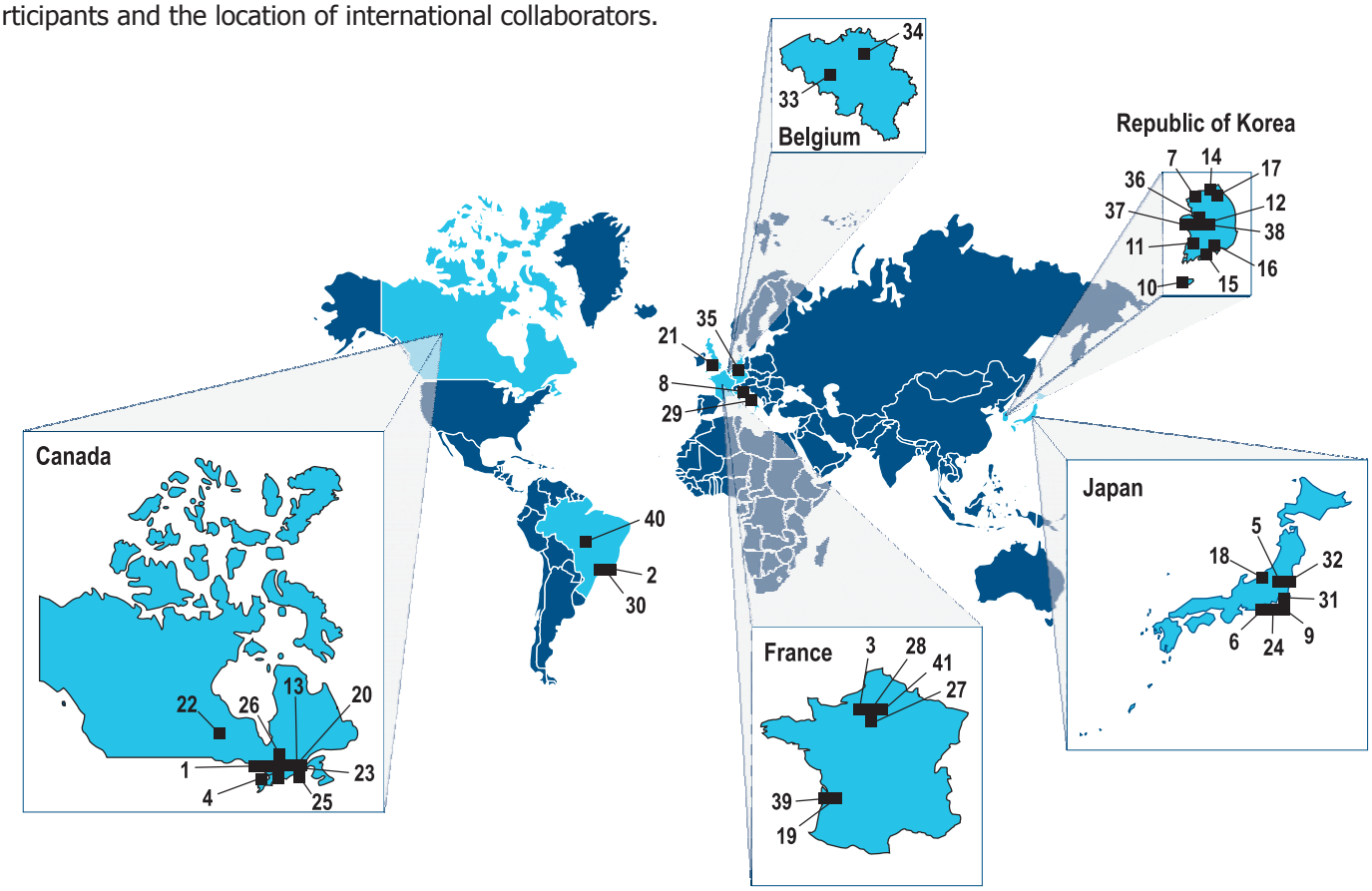
- Initiate new cooperative projects under existing agreements
- Conduct annual project review/bilateral meetings with international partners (Canada, EURATOM, France, and Republic of Korea)
- Continue pursuing new cooperative agreements with prospective partner countries, including the expected signature of the Republic of South Africa on a VHTR System Arrangement

During FY 2008, the following activities were completed:

- Expansion of GIF's cooperative enterprise through the accession of the Republic of South Africa to the GIF framework agreement, bringing the total number of participating countries to nine
- Signature of the People's Republic of China in October on a VHTR System Arrangement
- Completion of four research projects that began in FY 2004 and FY 2005 with three I-NERI collaborators:
  - Brazil (1)
  - France (1)
  - Republic of Korea (2)
- Initiation of seven new collaborative research projects:
  - Canada (3)
  - France (1)
  - Republic of Korea (3)
- Annual project performance reviews and bilateral program planning meetings with the following collaborators:
  - Canada
  - EURATOM
  - France
  - Republic of Korea

Program Participants

Following is a complete listing of the I-NERI program participants and the location of international collaborators.



U.S. Collaborators

National Laboratories

Argonne National Laboratory
Brookhaven National Laboratory
Idaho National Laboratory
Lawrence Livermore National Laboratory
Los Alamos National Laboratory
Oak Ridge National Laboratory
Pacific Northwest National Laboratory
Sandia National Laboratories

Industry Organizations

Gas Technology Institute
General Atomics
Westinghouse Electric

Universities

Iowa State University
Massachusetts Institute of Technology
Ohio State University
Pennsylvania State University
Purdue University
Rensselaer Polytechnic Institute
University of California-Santa Barbara
University of Florida
University of Idaho
University of Illinois-Chicago
University of Maryland
University of Michigan
University of Nevada-Las Vegas
University of Notre Dame
University of Wisconsin
Utah State University

International Collaborators

Industry Organizations

1 Atomic Energy of Canada Limited (AECL)
2 Eletronuclear
3 Framatome ANP
4 Gamma Engineering
5 Hitachi Works
6 Hitachi, LTD
7 Korea Hydro and Nuclear Power Company (KHNP)
8 Società Informazioni ed Esperienze Termoidrauliche (SIET)
9 Toshiba Corporation

Universities

10 Cheju National University
11 Chosun University
12 Chungnam National University
13 École Polytechnique de Montréal
14 Hanyang University
15 Korea Maritime University
16 Pusan National University
17 Seoul National University
18 Tohoku University
19 Université Bordeaux
20 Université de Sherbrooke
21 University of Manchester
22 University of Manitoba
23 University of Ontario Institute of Technology
24 University of Tokyo

Governmental Organizations

25 CANMET Energy Technology Centre-Ottawa (CETC-O)
26 Chalk River Laboratories (CRL)
27 Commissariat à l'énergie atomique (CEA)
28 Electricité de France (EDF)
29 Ente per le Nuove Tecnologie, l'Energia e l'Ambiente (ENEA)
30 Instituto de Pesquisas Energéticas e Nucleares (IPEN)
31 Japan Atomic Energy Agency (JAEA)
32 Japan Atomic Energy Research Institute (JAERI)
33 Joint Research Centre-Institute for Energy (JRC-IE)
34 Institute for Reference Materials and Measurements (IRMM)
35 Joint Research Centre-Institute for Transuranium Elements (JRC-ITU)
36 Korea Advanced Institute of Science and Technology (KAIST)
37 Korea Atomic Energy Research Institute (KAERI)
38 Korea Electric Power Research Institute (KEPRI)
39 Laboratoire des Composites Thermostructuraux (LCTS)
40 Ministério da Ciência e Tecnologia (MST)
41 Organisation for Economic Co-operation and Development-Nuclear Energy Agency (OECD/NEA)

U.S. Student Participation

One of the I-NERI program goals is to spur development of nuclear-related education and research opportunities at U.S. universities. As noted in Figure 2, a total of seven universities and colleges participated in I-NERI research projects during FY 2008. Approximately 63 students from these institutions worked on active I-NERI research projects during the year, distributed by degree level as shown in Figure 3.

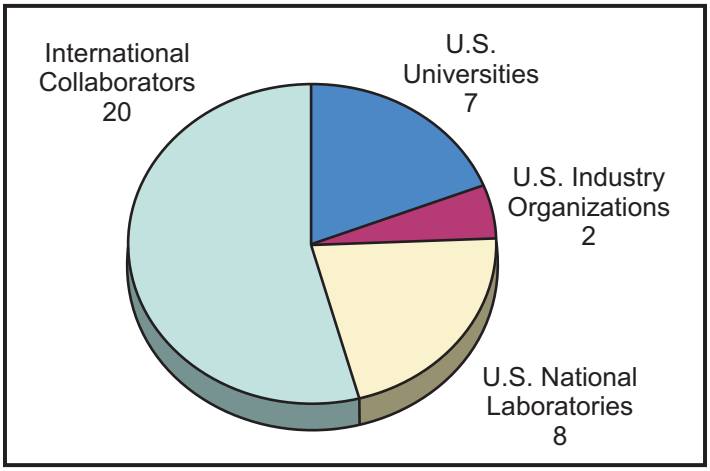


Figure 2. FY 2008 I-NERI participant profile.

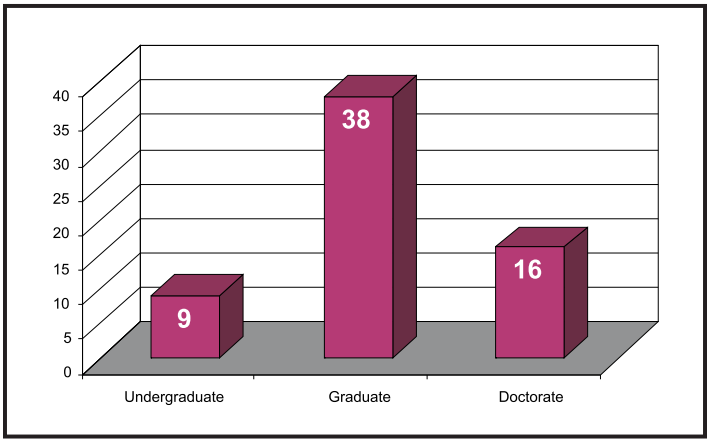


Figure 3. FY 2008 I-NERI student participation profile.

Completed I-NERI Projects

This year marked the completion of four I-NERI research projects; one began in FY 2004 and three in FY 2005:

- 2005-001-B — Development of Advanced Instrumentation and Control for an Integral Primary System Reactor, Oak Ridge National Laboratory
- 2004-010-F — PRA-Aided Design of Advanced Reactors With an Application to GFR Safety-Related Systems, Massachusetts Institute of Technology
- 2005-001-K — Supercritical Carbon Dioxide Brayton Cycle Energy Conversion, Argonne National Laboratory
- 2005-004-K — Development of Head-End Pyrochemical Reduction Process for Advanced Oxide Fuels, Idaho National Laboratory

The following section provides detailed summaries of each of these projects.

Based on the documented accomplishments, it is apparent that I-NERI's goals and objectives continue to be satisfied. Collaborative efforts between the public and private sectors in both the United States and partnering international entities have resulted in significant scientific and technological enhancements in the global nuclear power arena. International collaborations have forged lasting ties that will continue promoting the strong infrastructure necessary to overcome future challenges to the expanded use of this vital source of clean and reliable power. In conjunction with parallel efforts undertaken by the NERI program, Generation IV, AFCI, and NHI, this program has helped to revive the Nation's leadership role in international nuclear R&D. The resulting technological and scientific advances will ensure the United States remains competitive in both the global and domestic nuclear energy marketplaces.

4.0 Project Summaries and Abstracts

Sections 4.1 thorugh 4.6 provide descriptions of the work scopes, listings of funded projects, and brief project status reports for I-NERI R&D projects undertaken with Brazil, Canada, the European Union, France, Japan, and the Republic of Korea.

4.1 U.S./Brazil Collaboration

The U.S. Department of Energy (DOE) and the Brazilian Ministry of Science and Technology (MST) established a bilateral agreement on June 20, 2003. Secretary of Energy Spencer Abraham signed the agreement for DOE, and Brazilian Minister of Science and Technology Roberto Amaral signed for MST. The first collaborative project under this agreement was awarded in FY 2005.

Cooperative research projects with MST take place primarily in the areas of advanced nuclear fuels, fuel cycles, and materials, based on a bilateral agreement signed June 20, 2003. Cooperative research with Brazil entails instrumentation, operations and control, and human interaction with the integral primary system reactor, along with an investigation into shared resources for multiple modular reactor designs.

Work Scope Areas

Following are the R&D topical areas for the U.S./Brazil collaboration:

- Advanced reactor development for future-generation nuclear energy systems
- Advanced reactor fuel and reactor fuel cycle integration
- Life-cycle management and upgrading of current operating reactors
- Advanced fuel and material irradiation and use of experimental facilities
- Environmental and safety issues related to new reactor and fuel cycle technologies
- Fundamental nuclear science and engineering

Project Summaries

The last remaining project was completed in FY 2008, with no new projects awarded. The completed I-NERI U.S./Brazil project is shown on the next page, followed by a summary of the accomplishments achieved in FY 2008.

Directory of Project Summaries

* completed in FY 2008

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Advanced Instrumentation and Control for an Integral Primary System Reactor

PI (U.S.): David Holcomb, Oak Ridge National Laboratory (ORNL)

PI (Brazil): Antonio Barroso, Instituto de Pesquisas Energéticas e Nucleares

Collaborators: Westinghouse Electric Company

Project Number: 2005-001-B

Program Area: AFCI

Project Start Date: March 2005

Project End Date: March 2008

Research Objectives

Integral primary system reactors (IPSRs) have distinctive instrumentation and control (I&C) configurations and requirements compared to traditional external loop light water reactors (LWRs). The overall objective of this project was to develop specialized I&C technologies that are not directly transferable from external loop LWRs. As no systematic assessment of I&C requirements for IPSRs was available when this project began, performing a detailed review of the instrumentation requirements became a leading task.

A previously identified instrumentation challenge for IPSRs is obtaining an accurate, in-vessel water-level measurement. Conventional measurements are almost impossible because of the irregular path imposed by the shape of the vessel's internal structural components, such as the pressurizer bottom plate, riser, and control rod drive mechanisms. Two candidate systems are currently under development: 1) an ultrasonic, torsional waveguide-based level measurement technique and 2) the application of advanced signal processing algorithms to a cooled-fluid-based, lance-type probe.

An additional objective of this work is to assess areas in plant operation and control where unique IPSR features and operating modes require innovative approaches. The medium size and modularity of the IPSR provide economic incentives for deploying multiple reactor modules in a single nuclear park. Co-generation, which produces desalinated water, district heating, industrial steam, or hydrogen, in addition to electrical power, is an attractive option for modular reactors sited in areas that already have sufficient electrical generation capacity to supply baseload power requirements. In order to fully use the energy available from all reactors in a nuclear park, the balance-of-plant must be reconfigurable to vary co-generation with hourly changes in electrical load. In order to optimize multi-modular and/or reconfigurable operation, a hierarchical supervisory control system needs to be developed in order to overlay the individual unit control system. The hierarchical control development task will maximize the utilization efficiency of the power-park resources while minimizing staffing requirements.

The final technical area of this project is to develop guidelines for operator interaction with the plant controls and protection systems. Advanced IPSR designs, particularly the Westinghouse International Reactor Innovative and Secure, are unique in that they are capable of responding to almost any operational condition or accident without operator intervention. The large, thermal inertia of their primary system gives these reactors long transient evolution times. Due to the advantageous thermal characteristics of IPSRs compared to external loop LWRs, the operator's interaction with the control/protection systems needs to be redesigned. Emergency procedure guidelines and control room architecture must also take into account the possibility of controlling multiple modules from a single control room.

Research Progress

The remaining project task for FY 2008 was completing fabrication and assembly of prototype instrumentation to provide accurate and reliable measurement of in-vessel coolant levels. This has now been accomplished. Additionally, an IPSR in-vessel environmental mimic vessel was designed, fabricated, and tested. The test chamber

consists of a code-rated pressure vessel with external heaters and a control system to allow comparison water-level measurement.

The overall task objective was to develop an in-vessel, coolant-level measurement instrument. The new instrument is based on magnetostrictively generating an ultrasonic torsional wave within a shaped waveguide. The speed of the torsional wave propagation along the waveguide is proportional to the density of the surrounding fluid, and the level is thus obtained by measuring the net transit time for the wave to progress down the waveguide and reflect back to the transducer.

The key technological advancement embodied in the new instrument is that the entire transducer system is designed for direct immersion in the reactor coolant at full pressure (approximately 15 MPa) and temperature (up to 330°C). All previous ultrasonic waveguide-based level measurement techniques have required locating the transducer in the mild environment outside the reactor vessel and therefore would have required transmission of the ultrasonic torsional signal across the reactor coolant system pressure boundary, which is technologically infeasible. The high vessel interior temperature and pressure necessitated restricting the transducer materials to only metals (including a custom, high-temperature magnetostrictive alloy) and ceramics with the exception of a recently developed, commercial organic potting material employed to encapsulate the interior wiring.

In addition to employing highly ruggedized materials, the new instrument also advances the state-of-the-art in ultrasonic torsional wave generation by employing an innovative, non-contact magnetostrictive striping-based drive technique to level measurement for the first time. The magnetostrictive transducer is excited by a Blackman windowed sine wave to launch a torsional wave along the waveguide. A digital correlation filtering method is used to recover the reflected torsional wave signal.

Currently, the measurement technique employing innovative transducer hardware, electronics, and signal processing has been successfully demonstrated in a laboratory environment and a transducer fabricated that is designed for harsh environment deployment. However, the new measurement system has not yet been demonstrated in prototypic in-vessel conditions.

A series of laboratory measurements were made using the prototype instrument intended for use under IPSR in-vessel environmental conditions. The measurement results using the ruggedized hardware were virtually identical to those using the earlier laboratory-grade prototype. Figure 1 shows representative ultrasonic reflection data acquired with the prototype level measurement system in a laboratory environment. The return time of the torsional signal (peak at approximately 2.5 ms in Figure 1) varies linearly with the depth of probe immersion. The grey plane in the figure highlights the frequency used for comparison measurements.

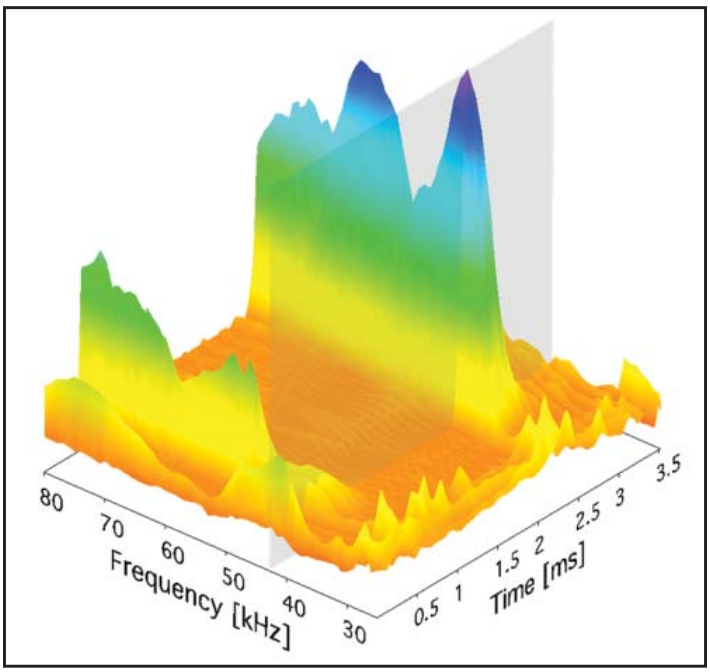


Figure 1. Ultrasonic reflection signal for prototype level measurement system.

Preliminary discussions have been held with reactor instrument vendors, who have expressed tentative interest in developing a commercial version of the new instrument—provided it can be demonstrated to function under prototypic conditions inside the vessel. With this in mind, ORNL is pursuing demonstration of the new instrument in high-pressure, high-temperature environments.

Planned Activities

This I-NERI project has concluded.

4.2 U.S./Canada Collaboration

The Director of the U.S. Department of Energy, Office of Nuclear Energy, William D. Magwood IV, signed a bilateral agreement on June 17, 2003, with the Assistant Deputy Minister of the Department of Natural Resources Canada, Ric Cameron, and the President, Research and Technology Division, Atomic Energy of Canada Limited, David F. Torgerson. The first U.S./Canada collaborative research projects were awarded in FY 2004.

Work Scope Areas

Following are the R&D topical areas for the U.S./Canada collaboration:

- Hydrogen production by nuclear systems
- Sustainable and advanced fuel cycles
- Supercritical water-cooled reactor concepts

Project Summaries

Three new projects were awarded during the past year. A listing of the I-NERI U.S./Canada projects that are currently under way follows, along with summaries of the accomplishments achieved in FY 2008.

Directory of Project Summaries

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Hydrogen Production Using High-Temperature Electrolysis

PI (U.S.): Stephen Herring, Idaho National Laboratory (INL)

PI (Canada): Sam Suppiah and Donald Ryland, Atomic Energy of Canada Limited (AECL)

Collaborators: University of Ontario Institute of Technology

Project Number: 2007-001-C

Program Area: NHI

Project Start Date: July 2007

Project End Date: March 2010

Research Objectives

This project will address research and scale-up issues associated with the use of energy-efficient, high-temperature solid-oxide electrolysis cells to produce hydrogen from steam. The project has three primary objectives: 1) to perform an economic analysis of electrolysis processes, 2) to develop methods for safely handling high-temperature oxygen in high-temperature electrolysis (HTE) systems, and 3) to identify high-temperature materials for the cell stack and heat exchangers.

HTE systems operate at high temperatures, presenting significant technical issues and hazards in handling the hot gas streams, especially oxygen. An integrated laboratory scale (ILS) experiment will dilute the 850°C oxygen stream with air and cool it to room temperature before being exhausted. Researchers will develop methods for safely handling the 3 $Nm^3 \cdot h^{-1}$ of hot oxygen produced (2.1 kg O_2/hr), along with processes for recovering heat from the oxygen stream. Proper selection of materials for the key components—high-temperature heat exchangers, cell stack components, and interconnecting piping—is critical to achieving high performance of the overall HTE process. Researchers will conduct a series of corrosion tests to analyze material performance in both the HTE cell and the balance of the plant.

The following tasks comprise the main elements of this project:

1) Conduct economic analysis of low-temperature and HTE systems
2) Provide input for safe handling of oxygen streams
3) Develop process for heat recovery from hot oxygen streams
4) Identify suitable materials for HTE systems and analyze their performance

Research Progress

Researchers are working on a project to generate hydrogen using an HTE process. In such a system, safety precautions need to be taken to handle high-temperature oxygen at approximately 830°C. Though oxygen itself is not flammable, most engineering materials, including many gases and liquids, will burn in the presence of oxygen under some favorable physicochemical conditions. At present, an absolute set of rules does not exist that can cover all aspects of oxygen system design, material selection, and operating practices to avoid subtle hazards related to oxygen. Because most materials, including metals, will burn in an oxygen-enriched environment, hazards are always present when using oxygen. Most materials will ignite in an oxygen-enriched environment at a temperature lower than that in air and, once ignited, combustion rates are greater in the oxygen-enriched environment. However, these hazards do not preclude the operations and systems involving oxygen. In fact, the probability of oxygen system fires is reported to be low—about one in a million.

During FY 2008, researchers at INL published a report addressing Tasks 2 and 3, concerning the handling and cooling of oxygen in an HTE plant. Tasks 1 and 4 will be addressed in the second and third years of this project. The report is a practical guide and tutorial for the safe operation and handling of gaseous oxygen in an HTE

system. The intent is to provide safe, practical guidance that permits the accomplishment of experimental operations at INL, while being restrictive enough to prevent personnel endangerment and to provide reasonable facility protection. Adequate guidelines are provided to govern various aspects of oxygen handling associated with an HTE system to generate hydrogen. The intent is to present acceptable oxygen standards and practices for minimum safety requirements. The report provides a summary of operational hazards, along with oxygen safety and emergency procedures.

When selecting a material for use with oxygen, the circumstances that cause oxygen to react with the material need to be understood. When the ignition energy input exceeds the configuration-dependent threshold, ignition and combustion may occur. Therefore, safe use of oxygen requires the control of potential ignition energy mechanisms within oxygen systems by judiciously selecting ignition-resistant materials. Ignition mechanisms, which can result in pipeline failures, include the following:

- Particle impact ignition caused by impingement of metallic or non-metallic materials with the metal components of the pipeline
- Adiabatic compression, acoustic resonance, and flow friction which create temperature increase
- Ignition initiated by the combustion of organic materials or contaminants entrained in the oxygen flow
- Friction caused by rubbing, as in a valve between adjacent moving and stationary parts
- Electric arcing between metallic components due to static electricity or lightning, which generates enough energy to ignite metallic or non-metallic materials

When the ignition mechanism has started, the combustion can propagate through the kindling chain. Once ignited, the combustible material or component generates heat, which can, depending on many factors, ignite the bulk material of the pressure envelope. The rate and extent of the propagation of the fire along the pressure envelope will depend on the thickness and the flammability of the material. The use of exempt materials will limit the propagation of the combustion by interrupting the kindling chain. The rate and extent of the propagation of the fire is also influenced by oxygen parameters such as pressure, purity, temperature, and the total oxygen inventory available to support combustion. For pressures below 30 psig (0.21 MPa), experimental data show that the combustion rates of potential materials used in oxygen pipeline components, such as carbon steel,

Application	Low pressure (less than 1,000 psi)	High pressure (greater than 1,000 psi)
Component bodies	Nickel alloy steel, Stainless steel	Monel Inconel 718
Tubing and fittings	Copper, Stainless steel, Steel, Aluminum, Aluminum alloys	Monel Inconel 718
Internal parts	Stainless steel	Monel Inconel 718, Beryllium copper
Springs	Stainless steel	Beryllium copper, Elgiloy, Monel
Valve seats	Stainless steel	Gold or silver plated over Monel or Inconel 718
Valve balls	Stainless steel, Tungsten carbide	Sapphire
Lubricants	Everlube 812, Microsel 100-1 and 200-1, Triolube 1175, Krytox 240AB, and 240AC Braycote 3L-38RP	Batch/lot-tested Braycote 3L-38RP, Batch/lot-tested Everlube 812, Krytox 240AC
O-seals and backup	TFE, Halon, TFE Teflon, Kel F, Viton	Batch/lot-tested Viton Batch/lot-tested Teflon
Pressure vessels	Nickel steel, Stainless steel, Steel, Aluminum alloys	Inconel 718

Table 1. Materials for conservative design standards. Materials listed in the "Low pressure" column and other materials that are not listed may be suitable for more extreme environment oxygen service. Careful engineering analysis and rationale shall be used to select alternate materials.

are very low and decrease with decreasing pressure. This effect has contributed to the excellent service experience demonstrated by properly designed carbon steel components in selected very low-pressure oxygen applications. Many causes of fire can be avoided and their consequences reduced by good design practices. Other causes of fire may be due to unsuitable maintenance and operating practices, such as:

- Overheating due to either a process failure or an oxygen leakage from the system resulting in an external fire adjacent to the pipeline
- Accidental mixing with fuel, due to either a process failure or contaminant introduction during maintenance work

Planned Activities

During the remaining two years of this project, researchers will address Tasks 1 and 4. The economic analysis (Task 1) will build on the work performed by INL during FY 2007. In addition, Task 1 will include a comparison between conventional, room-temperature electrolysis of water and HTE of steam.

Task 4 will be a collaboration among researchers at INL, AECL, and the University of Ontario Institute of Technology. INL has performed tests of various metals in either steam-hydrogen or air-oxygen mixtures at temperatures up to 925°C and for durations up to 500 hours. Post-test examination has determined the rates of oxide growth and whether or not the oxide film was adherent to the base metal. In addition, the materials used in the ILS experiment during early FY 2009 will become available for examination. The ILS experiment included heat recuperators in which either a steam-hydrogen mixture or an air-oxygen mixture was cooled from approximately 820°C to 450°C for the test duration of 1,080 hours. Discussions during FY 2009 will determine the most logical location for further tests and the set of materials, experimental conditions, and durations to be used.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Thermochemical Hydrogen Production Process Analysis

PI (U.S.): Michele Lewis, Argonne National Laboratory (ANL)

PI (Canada): Sam Suppiah, Atomic Energy of Canada Limited (AECL)

Collaborators: Pennsylvania State University, University of South Carolina, Tulane University, University of Ontario Institute of Technology, Guelph University, Toronto University, University of Western Ontario, Waterloo University, McMaster University

Project Number: 2007-002-C

Program Area: NHI

Project Start Date: September 2008

Project End Date: August 2011

Research Objectives

The development of the copper-chloride (Cu-Cl) cycle is ongoing as part of the Department of Energy's (DOE's) Nuclear Hydrogen Initiative (NHI) alternative thermochemical cycle program. The Cu-Cl cycle is a promising thermochemical cycle for heat sources that provide heat from 550°C to 600°C. The cycle consists of the three major reactions shown in Table 1.

#	Reaction Stoichiometry	Temperature (°C)
1	$2CuCl + 2HCl\ (aq) \rightarrow 2CuCl_2(aq) + H_2(g)$	<100
2	$2CuCl_2 + H_2O(g) \rightarrow Cu_2OCl_2\ (s) + 2HCl(g)$	300–375
3	$Cu_2OCl_2\ (s) \rightarrow 2CuCl(l) + \frac{1}{2}O_2(g)$	450–530

Table 1. Major reactions in the Cu-Cl cycle.

The effort at ANL has focused on meeting two objectives: 1) to optimize the yields in the hydrolysis reaction, shown as #2 in Table 1, and 2) to determine the efficiency and cost of hydrogen production. The primary objective of AECL is to develop the electrolyzer for the electrochemical reaction, shown as #1 in Table 1. There are two university programs associated with this thermochemical cycle research program, one in Canada and one in the United States. The objectives of the universities are to support the primary activities with special reference to advanced electrochemical technologies in the United States and to the development of engineering-scale equipment. Other work, such as identifying corrosion-resistant materials for construction and analyzing reliability and stability systems for distributed control for nuclear-based hydrogen generation, is also ongoing.

Research Progress

Hydrolysis Reaction. The objective of this work is to increase the yield of Cu_2OCl_2 and decrease the yield of CuCl, which may be a product of the thermal decomposition of $CuCl_2$, a competing reaction. Past work showed that the previous fixed bed reactor designs did not provide sufficient heat and mass transfer, as much of the $CuCl_2$ did not react and tended to be agglomerated in the middle of the fixed bed. In addition, up to 25 wt% CuCl was measured in some of the solid products. Researchers designed, built, and tested new hydrolysis reactor concepts that offered better mass and heat transfer.

One concept that appears very promising is a spray reactor. A spray reactor generally provides greater heat and mass transfer than a fixed bed reactor because very fine droplets/particles are dispersed within the reactor's heated zone. In this concept, a solution of $CuCl_2$ is injected into the reactor using a nebulizer or an atomizer. Researchers tested two types of nebulizers: the first one is a pneumatic

quartz nebulizer commonly used for inductively coupled plasma-mass spectrometry (ICP-MS) analyses (glass expansion); the second is an ultrasonic nozzle (Sono-Tek). The ICP-MS requires argon (Ar) as a sweep gas to create a fine mist of $CuCl_2$ droplets while the ultrasonic nozzle uses ultrasonic vibrations to atomize the solution. Superheated steam with or without Ar can be used to improve heat transfer and is added either at the top of the reactor (co-current flow design) or at the bottom (counter-current flow design). Solid products are collected at the bottom of the reactor where the temperature is kept at 150°C for quenching the reaction while keeping the material dry.

Preliminary experimental results show that high yields of Cu_2OCl_2 can be obtained, as shown by the x-ray diffraction (XRD) pattern of one of the products in Figure 1. Some un-reacted $CuCl_2$ is still present and further optimization work is continuing. However, the most intense peaks are those of Cu_2OCl_2, which is referred to as melanothallite in the XRD database.

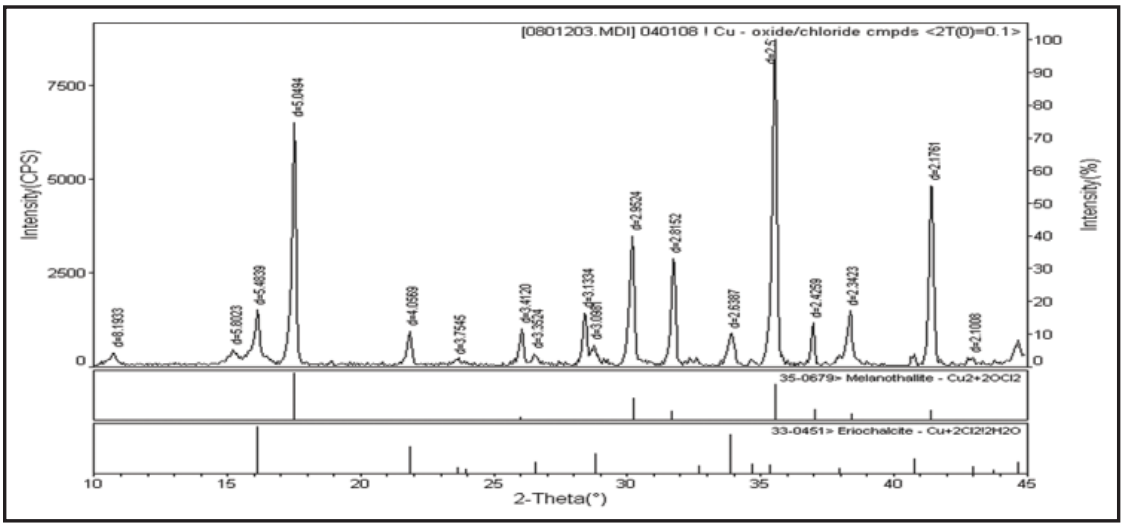


Figure 1. XRD pattern for a product of the hydrolysis reaction using a nebulizer reactor.

Efficiency and Economic Analysis. The objective of this task is to determine if the estimated efficiency and hydrogen production costs can be expected to meet DOE's efficiency and cost targets.

Researchers have updated the conceptual process design and the corresponding Aspen flowsheet. They have made several improvements to the design, including a flash unit to treat the effluent from the hydrolysis reactor, an ejector to pull a slight vacuum on the hydrolysis unit, and a direct heat exchanger to cool the molten salt from the oxychloride decomposition reactor. The addition of the flash unit to treat the effluent from the cathode gives two streams: 1) an acid-rich liquid stream, which is recycled, and 2) an acid-poor steam stream with very low HCl content, which is used as the feed for the hydrolysis reactor. The incorporation of the ejector allows researchers to reduce the steam's cupric chloride molar ratio from 16 to 8, thereby reducing the heat load required to vaporize the amount of water per mole of cupric chloride. The direct heat exchanger is used to efficiently recover the heat in the molten salt. With these changes and the assumptions that the team will meet the performance goals of 0.5 V and 500 mA/cm^2 for the electrolyzer, the efficiency was calculated as 46 percent (LHV) using the Aspen Plus flowsheet. Researchers assumed that the conversion of thermal heat to electricity is 40 percent.

$$\text{Efficiency} = \frac{\text{Mol. of } H_2 \text{ Produced} * \text{LHV}}{(\text{Pinch Heat} + \text{Electrochemical Work} + \text{Shaft Work})}$$

$$\text{Efficiency} = 125{,}000 \times 33.3/(24(191{,}000 + 73000/0.4) + 4(4000)/0.4) = 46 \text{ percent (LHV)}$$

Higher cell voltages will reduce the efficiency of producing hydrogen, so it is critical to focus on the development of the electrolyzer.

Development of Electrochemical Cells to Produce Hydrogen or Copper Powders. The objective of this task is to develop an electrochemical cell that produces hydrogen gas and $CuCl_2$ in an energy-efficient manner. The current H2A cost analysis has assumed targets for the cell electromotive force and current density of 0.5 V and 500 mA/cm^2. These targets assume that there is no copper-ion crossover, which acts as a parasitic reaction, and that the cell operates at about 100°C.

During this report period, AECL continued half-cell and single-cell electrochemical studies of the CuCl/HCl electrolysis step. The half-cell studies included an investigation into the effect of CuCl concentration and temperature on the rate of the Cu(I) oxidation reaction. In addition, the team investigated the effect of temperature on the rate of the hydrogen production reaction. The

single-cell experimental work has largely focused on modifying the experimental setup to incorporate ion exchange columns for the removal of Cu from the catholyte solution.

Half-Cell Electrolysis Studies. Researchers performed a preliminary half-cell experiment to determine the effect of temperature on the potential of the hydrogen production reaction during constant-current electrolysis at 100 mA/cm^2. They monitored the potential of the working electrode as a function of time, immersing the working electrode in a 6 M HCl electrolyte solution. The experimental data are presented in Figure 2 for three temperatures. The data show that the electrode potential for the hydrogen production reaction becomes less negative as the temperature is raised. These data indicate that higher-temperature cell operation is one method that can be used to decrease the cell operating voltage, thereby improving efficiency.

Single-Cell Electrolysis Studies. The experimental apparatus used to carry out single-cell electrolysis experiments is being modified to incorporate two ion exchange columns that will be used to maintain the purity of the 6 M HCl catholyte solution. Cu crossover from the anolyte to the catholyte has been a recurring problem in single-cell studies. The Cu that builds up in the catholyte can react at the cathode, which will affect the kinetics of the hydrogen production reaction. The problems that are being encountered with Cu in the catholyte solution are an artifact of the current scale of the electrolysis experiments and they should not be carried over to a large-scale process.

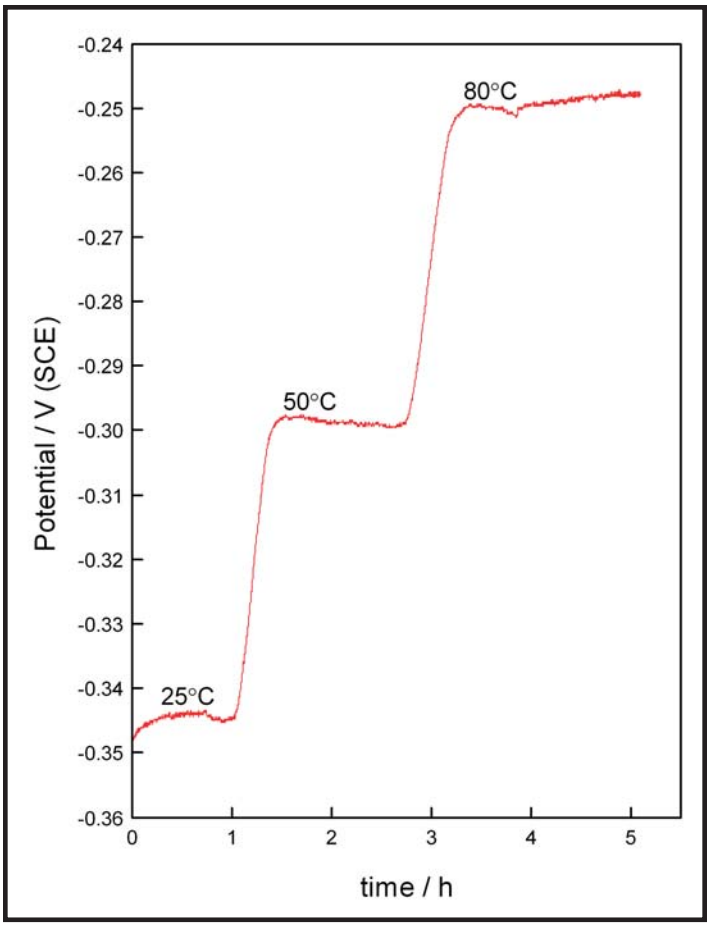


Figure 2. Potential of the hydrogen production reaction during constant-current electrolysis at the temperatures indicated. The current density was 100 mA/cm^2. The working electrode was 20 percent Pt/XC-72R on a graphite plate working electrode.

Planned Activities

Planned activities include optimizing the performance of the reactor with the ultrasonic nozzle. As new information is obtained, the economic analysis and efficiency calculations will be updated. Further development of the electrolyzer will continue with the focus on improving performance to meet the goals described above. The collaborating universities will provide support on these tasks. With respect to the electrolyzer's development, researchers at the universities in the United States will focus on developing membranes that prevent copper-ion crossover.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Upgrading of the Athabasca Oil Sands for the Production of Diesel and Gasoline

PI (U.S.): J.S. Herring, Idaho National Laboratory (INL)

PI (Canada): S. Suppiah, Atomic Energy of Canada, Limited

Collaborators: Chalk River Laboratories

Project Number: 2008-001-C

Program Area: NHI

Project Start Date: April 2008

Project End Date: April 2011

Project Abstract

This project represents a three-year collaboration to develop a preliminary design for an integrated complex consisting of a nuclear reactor, electrolytic hydrogen plant, and bitumen upgrader. It builds on a previous I-NERI project that developed a preliminary design to integrate a nuclear reactor with a high-temperature electrolysis system.

The complex being designed will provide steam for bitumen extraction and electricity for the electrolytic hydrogen plant. The hydrogen produced will be sent to the bitumen upgrader to produce synthetic crude oil, gasoline, and diesel fuel. The integrated plant will be optimized to maximize thermal efficiency and minimize the use of water.

The project is composed of two main tasks. The first is to define the requirements that the integrated complex will meet. These requirements include determination of the temperature, pressure, and quality of the steam required for the Steam Assisted Gravity Drainage process. In the first task, researchers will also select reference designs for the nuclear reactor, hydrogen production process, and process for bitumen upgrading. The bitumen upgrading process selected will specify the requirements for the hydrogen stream (i.e., temperature, pressure, and quantity) produced by the electrolytic hydrogen plant. In the second task, the team will develop the integrated model of the complex consisting of a nuclear reactor, hydrogen plant, and bitumen upgrader. The model will be developed using the process simulator HYSYS from AspenTech, Inc. HYSYS was used by INL staff for their work on integrating a nuclear reactor with a high-temperature electrolysis plant. Because HYSYS was originally designed as a process simulator for oil and gas applications, it is suitable for the simulation of a bitumen upgrading process.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Evaluation and Demonstration of Advanced Power Conversion and Hydrogen Generation Technologies for Low Emission Nuclear, Fossil and Renewable Energy Sources

PI (U.S.): Paul Pickard, Sandia National Laboratories

PI (Canada): Kourosh E. Zanganeh, CANMET Energy Technology Centre-Ottowa

Collaborators: None

Project Number: 2008-002-C

Program Area: NHI

Project Start Date: June 2008

Project End Date: June 2011

Project Abstract

This project is a collaborative research effort to evaluate crosscutting advanced power conversion and hydrogen generation technologies for application to low-emission nuclear, fossil, and renewable energy sources. The initial focus of the work will be on evaluating the potential of the supercritical CO_2 Brayton cycle for use with these advanced energy sources. Based on the first-year evaluations, researchers will select the most promising power conversion application and, in the second year, develop a preliminary design for a small-scale integrated demonstration system. Depending on the availability of resources, the selected system will be constructed and experiments will be conducted to demonstrate the key technologies in the third year. Coupling the advanced power-conversion technology being developed by the U.S. Department of Energy for nuclear applications with the next-generation fossil fuel applications being investigated by CANMET can accelerate the development of both of these potentially more efficient technologies. This research and development plan will also examine advanced thermochemical processes for hydrogen production for use with these advanced nuclear and fossil energy systems at a later stage.

Researchers have constructed the first supercritical CO_2 test loops and are conducting experiments to demonstrate the key technologies of this cycle. The initial review of candidate advanced energy sources is focusing on near-zero emission fossil fuel concepts, including natural gas-fired combined cycle systems.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Study of Burning Transuranic Elements in Heavy and Light Water Reactors

PI (U.S.): Jess C. Gehin, Oak Ridge National Laboratory (ORNL)

PI (Canada): Gary R. Dyck, Atomic Energy of Canada, Limited (AECL)

Collaborators: None

Project Number: 2008-003-C

Program Area: AFCI

Project Start Date: May 2008

Project End Date: September 2009

Research Objectives

The objectives of this project are to 1) examine and compare the isotopic evolution of actinide-bearing fuel under irradiation in light water reactors (LWRs) and heavy water reactors (HWRs), 2) understand the physics processes involved, and 3) provide a comparative analysis of the burning performance in the two reactor types. In this project, ORNL and AECL are collaborating on evaluating the capability of LWRs and HWRs for burning transuranic elements, particularly minor actinides. The two teams are focusing on their respective technical strengths in a coordinated fashion: the ORNL team is addressing the LWR aspects of this project, primarily pressurized water reactors (PWRs), while the AECL team is focusing on Canada Deuterium Uranium (CANDU) HWR technology. The work was arranged in four tasks: 1) establish goals for an intermediate actinide burner and metrics for assessing repository benefits; 2) determine a nominal composition for spent LWR fuel; 3) calculate the isotopic evolutions in LWR and HWR cores with transuranic mixed oxide (TRU-MOX) fuel and targets, considering the use of recycled uranium (RU), as appropriate; and 4) perform comparative analysis of results from LWR and HWR analyses. These results will be related to reduction in decay heat of the resulting high-level waste. ORNL is leading Tasks 1 and 2, while ORNL and AECL are sharing the lead on Tasks 3 and 4.

Research Progress

Metrics for evaluating the burner reactors have been developed based on discussions between ORNL and AECL and on metrics typically used in AFCI's systems analysis activities. These metrics include the fraction of TRU isotopes that are destroyed by the burner reactor, the reduction in decay heat, and the support ratio (the amount of energy production from primary reactors supported by 1 GWe of burner reactor). Using the SCALE code, ORNL determined the isotopic compositions for the PWR spent fuel and provided these data to AECL to be used as input specifications for the CANDU actinide-containing fuel. For the PWR fuel isotopics, the team modeled the scenario for 4 wt% ^{235}U initial enrichment 17×17 PWR fuel assemblies, burned to 50 GWd/MTHM, both at discharge and with 10 years of post-discharge cooling. AECL has developed lattice cell and full-core CANDU models to assess this and other relevant TRU fuel compositions. In addition to determining spent nuclear fuel isotopic compositions, ORNL developed lattice physics and full-core models to perform analyses comparing homogeneous and heterogeneous recycle options for LWRs. Progress has been made for both boiling water reactor and PWR applications.

In order to support the use of full-core analysis, the PWR analysis activities have involved developing lattice physics and full-core models to compare homogeneous and heterogeneous recycle options for LWRs. Researchers calculated the assembly cross sections using the TRITON lattice physics code and completed the formatting of the cross sections for input in the full-core analysis code, NESTLE. Performance of core calculation of MOX configuration to establish baseline performance numbers that represent currently acceptable designs is in progress. The assembly-level calculations provide some useful information on the transmutation performance for a PWR. For example, Figure 1 shows the depletion of ^{241}Am and other actinides in a PWR fuel assembly as the fuel depletes, showing a significant reduction in the actinide-containing pin. The full-core analysis, to be completed next year, will

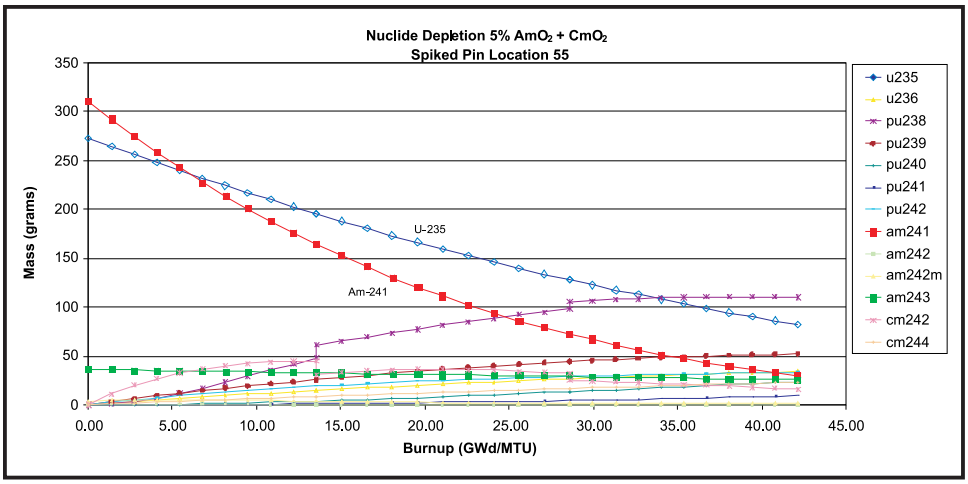


Figure 1. Depletion of ^{241}Am and other nuclides in PWR application.

provide the information needed to understand the impact on the reactor performance parameters (power peaking, cycle length, etc.) to better understand the ability of PWRs to burn minor actinides.

AECL has been developing and modeling TRU fuel concepts (TRU, TRU-MOX, and TRU-inert matrix fuel [IMF], with natural U or RU) in lattice cell models for CANDU-6 and also in full-core CANDU models. AECL has received the PWR spent fuel compositions calculated by ORNL and is using the data as inputs to their models.

Planned Activities

Activities in FY 2009 will focus on calculating the isotopic evolutions in HWR and LWR cores with TRU-MOX fuel and targets, as well as completing and documenting a comparative analysis of the results from the LWR and HWR analyses. Researchers will assess the performance of heterogeneous energy systems employing LWRs or HWRs as intermediate TRU burners, ahead of a fast burner reactor stage. A major metric of this work will be the reduction of the decay heat of the resulting high-level waste.

Specific activities at ORNL will include core calculations using homogeneous TRU-MOX assemblies that will be performed by varying loading to obtain performance numbers (i.e., cycle length, power distribution) as close to those of the Pu-MOX core design as achievable. Researchers will complete core calculations using heterogeneous target/MOX assemblies by varying target loadings to obtain performance numbers as close to those of the Pu-MOX core design as achievable. AECL developed lattice cell models for a CANDU-6 reactor fueled with TRU (Pu, Am, Np, and Cm), mixed with natural uranium (TRU-MOX), and in an inert matrix (TRU-IMF). Also, AECL developed a full-core model for TRU-MOX. Other scenarios that AECL is looking at are TRU burning in RU, and the burning of americium and curium in IMF in a number of CANDU peripheral fuel channels, with RU loaded into the rest of the CANDU core. The collaborative work between ORNL and AECL will 1) compare relevant results, 2) provide necessary calculations to complete the workscope of the project, and 3) allow the documentation of the work.

4.3 U.S./European Union Collaboration

The U.S. Department of Energy (DOE) and the European Atomic Energy Community (EURATOM), an international organization composed of the members of the European Union (EU), signed a bilateral agreement on March 6, 2003. Secretary of Energy Spencer Abraham signed the agreement for DOE, and Commissioner for Research Phillipe Busquin signed on behalf of EURATOM. In 2004, the United States and EURATOM selected the first ten projects for collaboration.

Work Scope Areas

Following are the R&D topical areas for the U.S./EU collaboration:

- Reactor fuels and materials research
- Advanced reactor design and engineering development
- R&D related to the transmutation of high-level nuclear waste
- Transmutations-related system analyses

Project Summaries

A collaboration with Italy was established in FY 2006 under the EURATOM agreement. A listing of the I-NERI U.S./EU projects that are currently under way follows, along with summaries of the accomplishments achieved in FY 2008.

Directory of Project Summaries

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Experimental Investigation of Small Break LOCAs in Coupled Vessel/Containment Integral Reactors

PI (U.S.): Milorad Dzodzo, Westinghouse Electric Corp.

PI (Italy): Fosco Bianchi, Ente per le Nuove Tecnologie, l'Energia e l'Ambiente (ENEA)

Collaborators: Oak Ridge National Laboratory (ORNL), Società Informazioni ed Esperienze Termoidrauliche (SIET)

Project Number: 2006-001-E

Program Area: AFCI

Project Start Date: May 2006

Project End Date: March 2010

Research Objectives

The purpose of this project is to verify experimentally the behavior of integral reactors during accident conditions. The Global Nuclear Energy Partnership includes international deployment of smaller-scale, grid-appropriate reactors with fully passive safety systems, such as the Westinghouse International Reactor Innovative and Secure (IRIS). IRIS offers advantages over traditional passive safety features with its inherent, design-based approach to coping with small break loss-of-coolant accidents (LOCAs) that does not rely on dedicated safety systems for coolant injection. The integral configuration of IRIS (without the primary loop external to reactor vessel) also precludes the possibility of a large break LOCA.

During a small break LOCA, the reactor vessel depressurizes due to heat removal and condensation by the integral steam generators, while the pressure in the small, spherical containment increases from the steam release through the break. The two pressures equalize relatively quickly, nullifying the pressure differential that drives the coolant egress, thus terminating the LOCA without any coolant injection or need for operator intervention. Numerous computer simulations performed under a variety of conditions indicate that the core remains safely covered at all times; however, an accurate and comprehensive experimental investigation is necessary to validate analytic tools and confirm this result.

This work will require extensive modifications and upgrades of SIET's existing AP600 test facility to represent the characteristics of an integral rather than a loop-type reactor and the vessel/containment coupling. Researchers will develop an analytical program to guide testing. The four-year duration supports the IRIS goal of submitting the design certification application in FY 2012.

The project entails the following tasks:

- Design a small break LOCA experimental facility for the coupled vessel/containment configuration that also allows investigation of other accident scenarios
- Review existing quality assurance (QA) plans and update as necessary to satisfy IRIS integral testing needs
- Perform pre-test analyses to guide and evaluate the actual tests
- Procure components and assemble the equipment necessary to modify, construct, and commission the test facility
- Conduct the tests in the test matrix, including shakedown tests
- Evaluate results and prepare a comprehensive report

Research Progress

Last year, the team created a conceptual design for the small break LOCA experimental facility, designated SPES-3. The team completed a stage gate review at SIET in Piacenza, Italy, on October 8, 2007. The completion of this major milestone allows moving forward with equipment procurement.

Numerous RELAP runs led to establishing a reference case for the direct vessel injection break in the lower portion of the vessel, as well as similar cases for the upper and automatic depressurization system breaks. Much of the effort focused on eliminating unrealistic conservatisms in the safety analysis models that would be difficult to simulate during testing. For example, the reference runs use realistic valve opening times where the safety analyses assume a maximum opening delay followed by instantaneous opening. The team used these results and scaling analysis to refine the designs for the reactor vessel and internal components to improve the scaling relationships between SPES-3 and the IRIS design. Similar activities for the remaining components (containment, tanks, refueling water storage tanks [RWST]) led to some SPES-3 design changes; most notable was a decision to use two RWST tanks. Uncertainties in the IRIS plant dimensions now limit further improvement in the SPES-3 design. At this point, the team has "frozen" a SPES-3 baseline design that SIET will use for equipment specification and procurement. The team also created a "frozen" SPES-3 reference IRIS design. This reference design will be the basis for all future IRIS/SPES-3 scaling analyses and for comparing SPES-3 and IRIS calculated transient responses.

The resulting design is for a full height, 1:100 volumetric scale model of the following components:

- Primary circuit
  - Reactor vessel and internals
  - Core
  - Reactor coolant pump
  - Circulation paths
  - Pressurizer
- Secondary circuits
  - Steam generators
  - Feed line
  - Steam line
- Emergency systems components and piping
  - Emergency boration tanks
  - Emergency heat removal systems
  - Refueling water storage tank
  - Automatic depressurization systems
- Containment system simulated by separate tanks and piping
  - Reactor cavity
  - Dry well
  - Pressure suppression system
  - Long-term gravity make-up system
  - Passive containment cooling

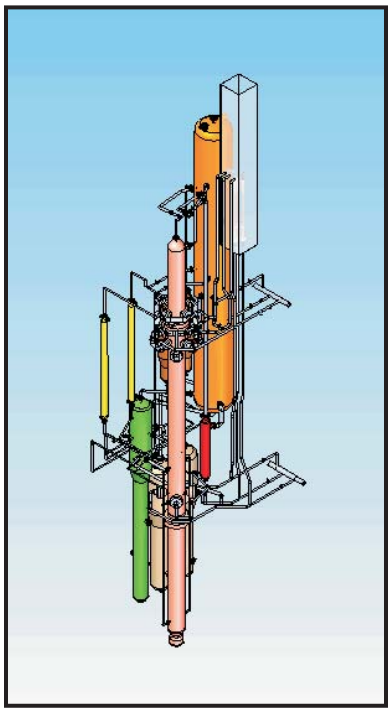


Figure 1. SPES-3 scale model.

The Italian partners have started procurement activities. SIET continues working to provide the details needed for equipment procurement. SIET is preparing valve specifications, tank specifications, instrumentation specifications, and data acquisition and control system specifications for procurement. ORNL provided technical support. Westinghouse issued a formal IRIS Integral System Test Specification that ENEA agreed to use to supersede their earlier specification to SIET.

The team is now updating the specification to reflect the latest "frozen" dimensions to which they agreed. SIET issued a quality plan of the IRIS SPES-3 project to cover their design, testing, and other activities on the project. Westinghouse will hire a QA engineer to develop an overall QA manual to cover activities and organizations associated with the IRIS program, including the SPES-3 program.

Planned Activities

The team will now 1) procure components, 2) assemble equipment, and 3) construct and commission the test facility. The Italian partners will have the main responsibility for this task, with consultation and advice from the U.S. partners. The very substantial cost of materials and refurbishing the facility will be borne by ENEA.

After commissioning, the main task will be to conduct the tests in the test matrix, including shakedown tests. The team will conduct some separate effects tests as part of the shakedown program, and then follow with various aspects and phases of the integral tests. SIET will conduct the tests with advice from Westinghouse. ENEA and ORNL will provide consultation and perform *ad hoc* analyses during and after the tests to verify, interpret, and guide the tests.

The final task will be to prepare a comprehensive test report. This will be part of the safety documentation submitted to support the IRIS application for final design approval with the U.S. Nuclear Regulatory Commission. Westinghouse will prepare the report with input from all the other organizations.

SIET and Westinghouse are very familiar with the QA requirements. Westinghouse will review and update existing QA plans as necessary to satisfy the IRIS integral testing needs and then prepare a project-specific QA manual. The QA plan will address test planning and execution, as well as the configuration control of the theoretical studies (e.g., pre- and post-test analyses).

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Advanced Nuclear Fuel Properties Measurements and Fuel Performance Modeling

PI (U.S.): J.R. Kennedy, Idaho National Laboratory (INL)

PI (EURATOM): V.V. Rondinella, Joint Research Centre, Institute for Transuranium Elements (JRC-ITU)

Collaborators: Arizona State University (ASU), Argonne National Laboratory, Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory, Pacific Northwest National Laboratory

Project Number: 2006-002-E

Program Area: AFCI

Start Date: January 2007

End Date: December 2010

Research Objectives

The behavior characteristics of various advanced fuel forms must be effectively modeled to determine in-pile performance. Considerable effort has been extended to model uranium oxide (UO_2) fuels, including recent studies of the thermal conductivity of irradiated UO_2 as a function of burnup and irradiation temperature. The next step in fuel performance code development and validation is for mixed oxide fuel (MOX), which, together with metal alloy fuel, is under consideration in the Advanced Fuel Cycle Initiative (AFCI) to act as a bridge to other advanced fuels. These include minor actinide (MA) transmutation fuel types such as advanced MOX, advanced metal alloy, and inert matrix fuel, in addition to nitride and carbide ceramic fuels that may also have application to space energy systems. The purpose of this project is to 1) promote the effective use of international resources for the fabrication and characterization of fuel types and 2) develop and validate fuel performance codes.

The U.S. Nuclear Regulatory Commission relies on the FRAPCON and FRAPTRAN codes for fuel licensing. Further development is required to support MOX and metal alloy fuels in view of the anticipated fuel definition and licensing process for AFCI and the framework of research and development activities. JRC-ITU has begun applying the TRANSURANUS code to MOX fuel. Further update with the thermophysical properties of MOX and metal alloy fuels with specific microscopic features is necessary, along with models for helium generation and release.

In order to improve understanding of the basic mechanisms behind materials properties and fuel performance and to predict the behavior of fuel material, researchers will undertake a multiscale modeling approach covering a spectrum of systems from *ab initio* atomistic calculations to macroscopic models.

Research Progress

Experimental work in the United States during 2008 was primarily directed toward fabrication and characterization of advanced metal alloy transmutation fuels and fabrication of advanced MOX transmutation fuels intended for irradiation testing within the scope of the AFC-2 test series. An x-ray diffraction diagram of one of the fuel compositions, along with the results from a Rietveld analysis of the reflection pattern, is shown in Figure 1. It can be seen that the

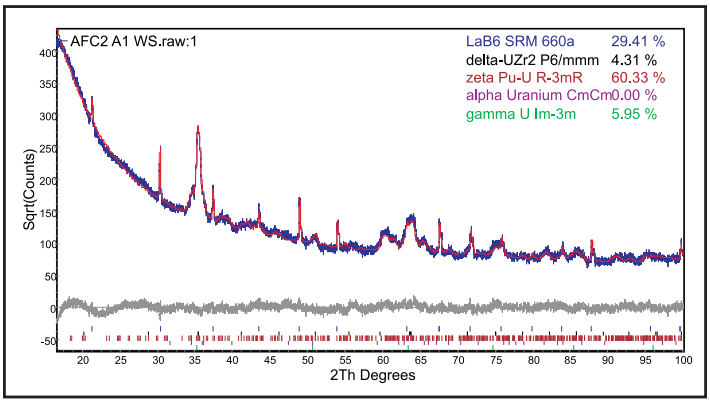


Figure 1. Rietveld refinement analysis results from an AFC-2 irradiation test series alloy.

alloy is composed of three phases: ζ-phase arising out of the U-Pu binary; δ-MZr_2 phase arising out of the U-Zr binary (and perhaps the Pu-Zr binary); and residual, high-temperature metastable γ-phase.

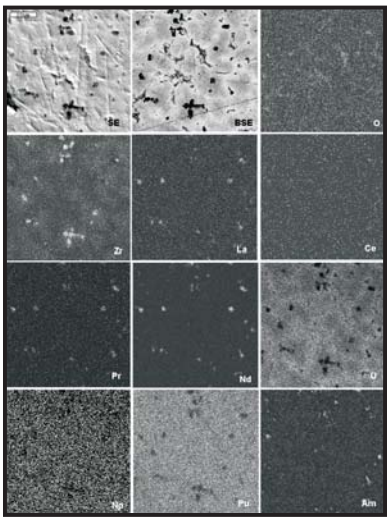


Figure 2. SE, BSE, and WDS x-ray maps from the interior of an AFC-2 irradiation test series fuel sample.

The microstructure, from secondary electron (SE) and back scattered electron (BSE) images, and element distribution mapping from wavelength dispersive spectroscopy (WDS) in an AFC2 irradiation test series fuel sample are shown in Figure 2. A two-phase lighter and darker contrast matrix can be observed, as well as dark contrast inclusions. Within the matrix phase there appears to be an inverse relation to the relative concentrations of uranium and zirconium, with perhaps an even distribution of plutonium and neptunium. Most of the dark contrast inclusions are enriched in zirconium and depleted in the actinides. The lanthanide additions can be seen to express themselves as precipitates wherein some, but not all, are associated with americium (Am).

Heating and cooling curves from ambient temperature to 800°C obtained from differential thermal analysis (DTA)-type measurements on a transmutation test fuel are provided in Figure 3. The traces are seen to be repeatable and reproducible for each alloy. The first heating cycle for each alloy is slightly different than the subsequent two heating cycles suggested to result from an annealing effect on the as-cast samples, as these alloys were not heat treated prior to measurement. This is supported by the observation that the same discrepancy among the three runs does not appear upon cooling. During all of the runs for each alloy, there are two very obvious, distinct transition zones that are reversible upon cooling. The alloys all show this similar feature, although the sharpness and intensity of each peak varies slightly depending upon the alloy stoichiometry. A very strong or sharp peak might suggest either a single phase transition or a transition through a very narrow phase field. A broad or weak peak might suggest a transition through a wide phase field, or overlap of two close, consecutive phase transitions. In all cases, the cooling curves appear to resolve the more complex phase behavior than that observed upon heating. There is reasonable consistency between the determined transition onset temperatures from the heating and cooling curves. The onset temperatures associated with the primary transitions shown in Figure 3 are approximately 555°C and 640°C. The overall heats of transition for these phase changes are about 7 J/g for the low temperature and about 21 J/g for the higher temperature transitions.

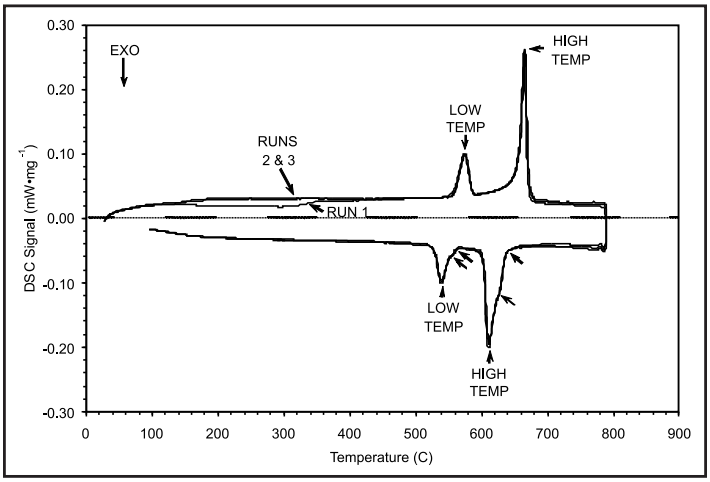


Figure 3. Heating and cooling traces obtained from the differential scanning calorimetry/DTA measurements on an AFC-2 irradiation test series fuel sample.

Interpretation of the observed transitions based on what is known for the U-Pu-Zr ternary system suggests that the thermal behavior follows the pathway $\delta+\zeta \rightarrow \delta+\gamma+\zeta$, $\delta+\gamma+\zeta \rightarrow \gamma+\zeta$, $\gamma+\zeta \rightarrow \gamma$-phase. The ongoing deconvolution of the overlapping peaks will allow a more quantitative analysis of the results.

The thermophysical property measurements on the transmutation metal alloys include determination of the thermal conductivity through measurements of the thermal expansion, density at room temperature, heat capacity, and thermal diffusivity. The final thermal conductivity plot for one of the AFC2 irradiation test fuel samples—as determined from the product of correlated data for density at temperature, heat capacity, and thermal diffusivity—is shown in Figure 4 up to 700°C. The data plotted in Figure 4 is generally consistent with what has been obtained previously on related alloys within the U-Pu-Am-Np-Zr system.

The principal investigator for the U.S. Department of Energy (DOE), Office of Basic Energy Sciences-funded Computational Materials Science Network (CMSN) visited JRC-ITU and held discussions on coordination with CMSN of shared fundamental physics-based materials science model development on fission gas release and microstructure

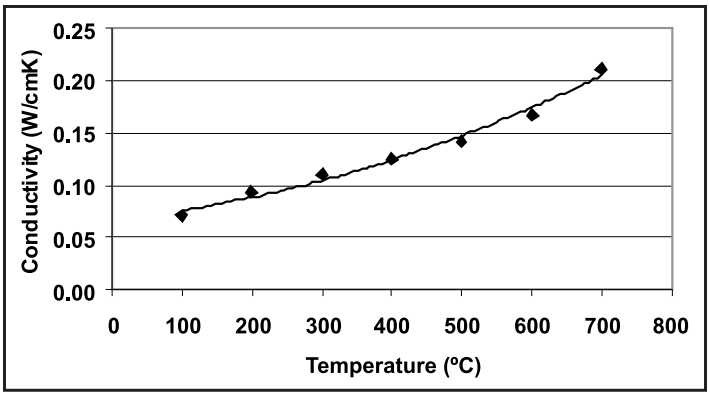


Figure 4. Plot of thermal conductivity values from correlated fitted data for an AFC-2 irradiation test fuel alloy sample.

development. The discussion will continue in FY 2009 in view of establishing a plan for joint investigations. Modeling and simulation work under DOE sponsorship relevant to the AFCI program was under reorganization in CY 2008 and future progress will be more evident.

DOE, Office of Nuclear Energy-AFCI-sponsored activities during CY 2008 included work to secure required feedstock supplies of the MAs neptunium (Np) and Am for future studies. Part of this work was to design and begin electrochemical reduction of NpO_2 in a glovebox and begin design for Am oxide reduction in a hot cell environment. JRC-ITU researchers provided valuable information through discussions and personal communications, as well as archived documentation on past efforts on the reduction of Np and Am oxides.

A key feature to the collaboration, and perhaps one of the more logistically difficult tasks to institute, will be a combined characterization effort on irradiated fuel samples. Irradiated fuels from the Fast Flux Test Facility reactor have been delivered to INL; however, researchers are still working on identifying a viable transport solution to exchange samples between INL and JRC-ITU that are essential for the deployment of the full scope of the present collaboration program.

ASU/LANL Microstructural Variability Effects/Modeling Activities. The structural integrity of ceramic fuels is important to ensure their overall performance in-pile. In this regard, the typical scatter observed on the fracture behavior of ceramics can, at least partially, be attributed to the intrinsic variability in their microstructures. A study has been undertaken at ASU in collaboration with LANL to evaluate the effect of this microstructural variability in the mechanical behavior of nitride fuel pellets and to formulate models that can also be used to evaluate directly the effects of microstructure on other phenomena of interest to nuclear fuels, e.g., heat transfer and mass transport. In order to achieve these objectives, microstructures of an appropriate surrogate material for plutonium nitride (PuN), i.e., zirconium nitride (ZrN), were characterized using electron microscopy techniques that allowed mapping geometry of pores and grains, as well as grain crystallography. A representative volume element (RVE) was defined from these measurements and then used to create a finite element model of a cylindrical fuel pellet that contained this RVE surrounded by a homogeneous material with the average properties calculated from the RVE. The model used properties for PuN and included anisotropic elasticity, thermal expansion, creep (both thermal and radiation-induced), and radiation-induced swelling. Initial calculations were performed for low levels of burnup (about one percent at a rate of one percent burnup/month) to evaluate the ability of the model to quantify variability of the local stress-strain response inside the RVE as a result of the microstructural heterogeneity. It is evident that the microstructure can indeed induce variability in the local stress distribution and the presence of stress concentrations can, in turn, lead to local fracture. Extension of these models to higher burnups and other physical mechanisms (heat and mass transfer) is under way.

JRC-ITU Fuel Properties/Modeling Activities. A comprehensive experimental study on thermophysical properties of ZrN and $(Zr_{1-x}Pu_x)N$ was carried out during the past three years with the aim of expanding and/or refining the knowledge available on advanced fuel materials. In particular, heat capacity data covering both low-temperature and high-temperature ranges and thermal conductivity curves were obtained for two nitride compounds using several techniques. As necessary development to be able to collect high-quality data, an important effort has been implemented to optimize sample handling/preparation and experimental procedures for oxidation-sensitive materials. This work was based on nitride compounds, but is applicable to other non-oxide fuel compounds as well (namely carbides, metallic alloys, etc.).

Planned Activities

Researchers will continue interactions related to post-irradiation examinations with emphasis on operation of electron probe micro-analysis and advanced technique development such as micro x-ray diffraction, focused ion beam, and thermal diffusivity measurements.

Results reported from researchers at JRC-ITU represent part of a larger program of characterization at JRC-ITU, which also includes studies on UN and on solid solutions $(U_xPu_{1-x})N$, and the program will be extended to cover both

(U, Pu, MA) N (where MA represents Np, Am, Cm) and irradiated nitride, carbide, and metallic alloy fuel samples. Information exchange and discussions of the results with possible joint publication with U.S. researchers are planned, particularly in the area of cooperation between the experimental measurements and the modeling and simulation computations on the nitride system for heat and mass transfer.

Results reported from researchers in the United States, as part of the AFCI program, will continue to focus on metal alloy and oxide systems. Researchers will also examine fundamental studies on basic actinide systems. Some emphasis will be given to high-temperature measurement development. Modeling and simulation activities will involve information exchange on computational efforts and mechanism development related to fission gas release, microstructure development, elemental redistribution, phase stability, and thermophysical property behavior of irradiated fuels.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Novel Transmutation Systems for Sustainable Nuclear Fuel Cycles

PI (US): J. Carmack, Idaho National Laboratory

PI (EURATOM): J. Somers, Joint Research Center Institute for Transuranium Elements (JRC-ITU)

Collaborators: Los Alamos National Laboratory

Project Number: 2006-003-E

Program Area: AFCI

Project Start Date: January 2007

Project End Date: January 2010

Research Objectives

This project seeks to develop transmutation fuel systems to support future nuclear reactor systems. Economic sustainability and increases in operating efficiencies may be empowered by novel developments in fuel matrices and fuel system configurations. This project provides the mechanism for research collaboration in the following areas: 1) nitride fuel development, 2) dispersion fuel development, 3) vibro-pac fuel development, 4) target system development, and 5) advanced claddings.

Nitride fuel research and development (R&D) will form fabrication processes that result in structurally stable fuel pellets. The research will focus on fabrication process development at small-scale and structural characterization of the resulting pellets. Low-temperature or low-heat fabrication processes will also be developed. Research on innovative processes (e.g., microwave processing, combustion synthesis, etc.) will continue.

The baseline transmutation concept under the Advanced Fuel Cycle Initiative relies on a homogeneous core with group TRU included in the fuel. An alternative option may be to put a non-fertile minor actinide into the reactor using inert matrix materials. A longer-term option may also be partitioning americium (Am), possibly with curium, and transmuting Am targets in a moderated pin.

Advanced nuclear fuel claddings have the potential to extend fuel life and burnup, maximizing fuel resources and actinide destruction. Oxide dispersion-strengthened (ODS) steels and advanced ferritic-martensitic materials are two such advanced claddings. R&D of advanced cladding will include fabrication and joining development as well as material property characterization. Material from in-process material irradiations will be used to begin characterization of irradiated material properties.

Research Progress

Nitride Fuels. The objective of the nitride fuel R&D is to develop fabrication processes that result in structurally stable fuel pellets and assess these fabrications through irradiation. The research focuses on fabrication process development on a small scale and structural characterization of the resulting pellets. Fabrication tests will be conducted over a range of fabrication parameters (e.g., powder size, sintering temperature, press pressure, etc). Nitride fuels may be a good alternative for the baseline transmutation fuels in the future.

Two nitride fuel compositions were irradiated in the Phenix fast spectrum test reactor during FY 2008. These two fuel compositions were irradiated to 60 effective, full-power days and will be returned to the United States in FY 2010 for post-irradiation examination. Joint planning by the United States and JRC-ITU for the post-irradiation examination of these pins was initiated in FY 2008 and will be completed in FY 2009. Shipment of the fuel pins will be conducted early in FY 2010 with post-irradiation examination coordinated with the JRC-ITU. JRC-ITU will conduct post-irradiation examination of two inert matrix fuel compositions that were also irradiated in the Phenix fast spectrum test reactor.

The performance of the two nitride fuel pins irradiated in the Phenix fast spectrum test reactor will be compared with that of a nitride fuel pin irradiated in the U.S. Advanced Test Reactor in the AFC-1G experiment. The AFC-1G nitride fuel pin was irradiated to approximately

28 percent at burnup. The AFC-1G nitride fuel pin completed irradiation in FY 2008 and is scheduled to begin post-irradiation examination in FY 2009.

Advanced Claddings. Advanced nuclear fuel claddings have the potential to extend fuel life and burnup, maximizing fuel resources and actinide destruction. ODS steels and advanced ferritic-martensitic materials are two such advanced claddings. R&D of advanced cladding includes fabrication and joining development as well as material property characterization. Material from in-process material irradiations will be used to begin characterization of irradiated material properties.

To burn fuels to greater than 20 percent burnup without changing the cladding requires development of materials to withstand doses greater than 200 dpa. During FY 2008, researchers tested advanced ODS ferritic steels using proton irradiations at temperatures of 25°C, 300°C, and 550°C. The materials tested include MA-956 and MA-957, which were compared to HT-9. Nanohardness was measured after irradiation and compared between the two materials. In the area where a proton enters the samples, there is hardening in the HT-9 sample and no hardening (except in the peak dose location) in the MA-957 sample. In addition, atom probe and small angle neutron scattering measurements have been performed on these advanced materials to characterize the oxide dispersions in these materials at the atomic scale. The same advanced materials have also been included in the MATRIX SMI and MATRIX 2 irradiations presently under way in the Phenix reactor. Total dose expected is 17 dpa and greater than 60 dpa at 400°C to 500°C while in contact with the sodium coolant.

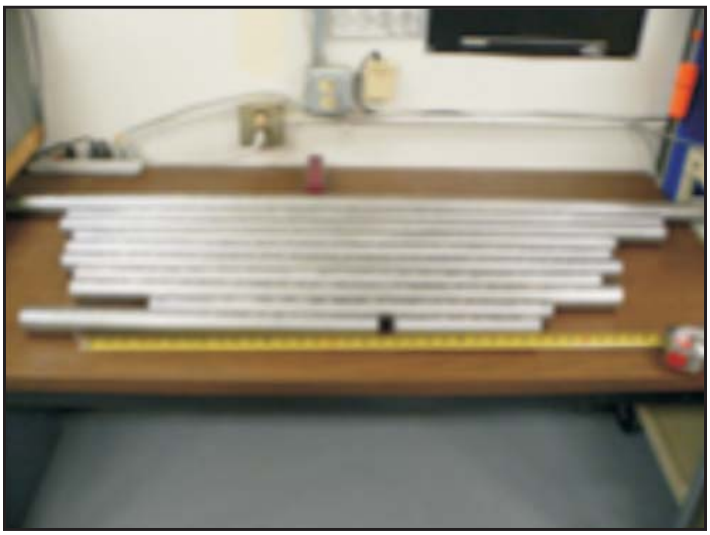


Figure 1. Bar stock drawn from 300-lb heat of HT-9 cladding.

Fast reactor cladding tubes have not been manufactured in the United States for more than a decade; it is important to revive this capability before the manufacturing knowledge is lost. To this end, a new 300-lb heat of the cladding alloy HT-9 was cast, and the tube manufacturing company Veridiam, Inc. (formerly Carpenter Technologies) has been contracted to fabricate the alloy into cladding tubes following NQA-1 standards. These cladding tubes will be used in future domestic and international fuels test irradiations that will be critical for optimizing fuel performance.

A significant barrier to achieving high burnup levels in HT-9 clad metal fuel is the chemical interaction between the fuel and the cladding that occurs once the fuel swells and contacts the cladding. Both coating and liner technologies are being explored to reduce or eliminate fuel-clad chemical interaction (FCCI). By mitigating FCCI, burnup levels of metal fuel systems will be increased, dramatically reducing fuel handling and improving transmutation efficiencies. Coating the inner surface of a long narrow tube has proven challenging, but during the year a new and novel technique to guide lasers into a cladding tube and deposit a TiN coating using pulsed laser deposition technology was developed. Successful fabrication of lined cladding with vanadium and zirconium liners through a plug-drawing process was demonstrated.

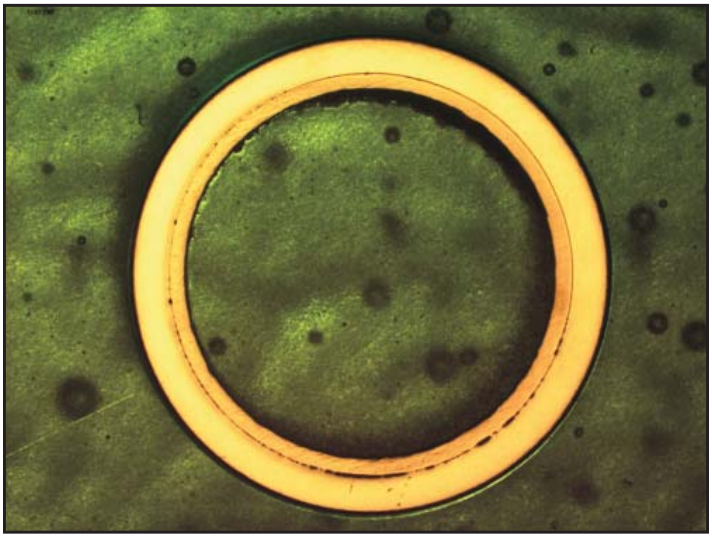


Figure 2. Zr-lined HT-9 cladding tube fabricated using a plug-drawing process.

Planned Activities

Activities in FY 2009 will focus on post-irradiation examination of nitride fuels and further development of advanced cladding systems for sodium fast reactors.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Oxide Fuels for Transmutation in Fast Reactors

PI (US): J. Carmack, Idaho National Laboratory (INL)

PI (EURATOM): J. Somers, Joint Research Center Institute for Transuranium Elements

Collaborators: Los Alamos National Laboratory (LANL)

Project Number: 2006-004-E

Program Area: AFCI

Project Start Date: January 2007

Project End Date: January 2010

Research Objectives

This project establishes the mechanism for minor actinide-bearing mixed oxide fuel development collaboration between the United States and EURATOM. The purpose is to establish cooperative research and development on oxide fuels, including conduct of the AFC-2 series of irradiations to be carried out under the transuranic fuel development program in the Advanced Test Reactor (ATR) at INL. This project will examine three main areas: 1) oxide fuel selection and fabrication, 2) oxide fuel characterization and modeling, and 3) irradiation testing and post-irradiation examination.

Oxide Fuel Selection and Fabrication. Assessment will include the analysis of neutronic behavior (plutonium [Pu] burn rate, reactivity swing), fuel thermal performance, fuel irradiation performance, and in-core corrosion resistance. Minor actinide mixed oxide fuel compositions will be fabricated for inclusion in the AFC-2 test series.

Oxide Fuel Characterization and Modeling. Experimental characterization of oxide fuel microstructure, thermal properties, and mechanical properties, corrosion testing, and ion-beam irradiation will be carried out to satisfy requirements for in-reactor irradiation testing and will provide a basis for fuel behavior modeling. Special attention will be paid to the characterization of fuels having varying oxygen (O)-to-metal ratios. Fuel modeling will be conducted using finite element analysis to determine the thermomechanical behavior of the fuel. Modeling of thermodynamic and transport properties will also be conducted.

Irradiation Testing and Post-Irradiation Examination. An assessment of transmutation fuel feasibility requires in-reactor irradiation due to the extremely complex nature of the in-core fission environment and the inability to simulate this environment with out-of-pile testing. This project will establish the basis of research for the AFC-2 irradiations to be conducted following the initial three-year period.

Research Progress

The focus in FY 2008 was on the fabrication of test pellets to be irradiated in the ATR for the AFC-2 C/D irradiation experiment.

An AFC-2 C/D fuel pellet containing 2 wt% neptunium (Np) and 3 wt% americium (Am) is shown in Figure 1. These fuel pellets are the first of their kind to be fabricated in the United States. Procedures for stoichiometry adjustment to pellets were developed to meet the irradiation test requirements. Pellets were centerless and ground to meet the tight specification of 0.191 inches to 0.192 inches and density criteria of 88+-%TD. Microstructural and scanning electron microscopy examinations found that the Am, Np, and Pu are evenly distributed throughout the matrix, showing good elemental homogeneity in the pellets. Dimensional stability during re-sintering was confirmed.

Figure 1. AFC-2 C/D fuel pellet containing 2 wt% Np and 3 wt% Am.

The oxide fuel pellets fabricated by LANL were placed in HT-9 cladding at INL and then in irradiation capsules for irradiation in the ATR. Two capsules (AFC-2 C/D) were inserted in the ATR in September 2008. These capsules contain the program's first oxide fuels with compositions relevant to fast reactor transmutation of minor actinides uranium (U), Pu, Am, Np, and O_{2-x}. In these oxide fuels, compound stoichiometry is an important test variable that will be studied. Stoichiometry in oxide fuels often controls fuel-cladding chemical interaction in oxide fuel systems that use stainless steel cladding. A total of six fuel pins were fabricated for irradiation. Table 1 provides detailed compositions of the six fuel pins in the AFC-2 C/D experiment. It is expected that this irradiation will continue through FY 2010 with post-irradiation examination occurring in the FY 2011 timeframe.

Rodlet	AFC-2C & -2D *
1	$(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.95}$
2	$(U_{0.8},Pu_{0.2})O_{1.98}$
3	$(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.98}$
4	$(U_{0.8},Pu_{0.2})O_{1.98}$
5	$(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.95}$
6	$(U_{0.75},Pu_{0.2},Am_{0.03},Np_{0.02})O_{1.98}$

***Fuel composition expressed in mole percent**

Table 1. Fuel compositions of the AFC-2 C/D experiment.

Planned Activities

Activities in FY 2009 will be focused on irradiation of the AFC-2 C/D experiment. In addition, collaborative post-irradiation examination of historical, U.S. fast reactor oxide fuels will be pursued. Planning for conduct of examination of high-burnup oxide fuels irradiated in the Fast Flux Test Facility will be undertaken as well.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Lead-Cooled Fast Reactor Concept Design and Evaluation

PI (U.S.): James J. Sienicki, Argonne National Laboratory

PI (EURATOM): Stefano Monti, Ente per le Nuove Tecnologie, l'Energia e l'Ambiente

Collaborators: Lawrence Livermore National Laboratory, Del Fungo Giera Energia S.P.A.

Project Number: 2006-005-E

Program Area: Generation IV

Project Start Date: TBD

Project End Date: TBD

Project Abstract

The purpose of this project is to conduct trade studies and develop a pre-conceptual design for a near-term deployable lead-cooled fast reactor (LFR) demonstrator. The demonstration reactor will implement innovative engineering features that exploit the characteristics of pure lead as the primary coolant to show its economic potential and industrial attractiveness. Researchers will select a suitable power level to confirm the key features of the prospective LFR fleet and enable irradiations of advanced fuels and materials. The project is focused on defining the reactor core and primary system configuration, with the following objectives:

- To identify the most promising design features of the future LFR in order to demonstrate its feasibility
- To create conditions and incentives for constructing commercial LFR prototypes
- To guide the selection of options based on experiment results

The demonstration facility will incorporate innovative features of the European Lead-cooled System concept. The facility's pre-conceptual design will be flexible to accommodate subsequent modifications and improvements in order to allow testing of new components and systems, as well as to serve as a test facility for the continuous improvement of prospective industrial/commercial reactors. The core layout will feature geometrical and functional flexibility in order to test different fuels and fuel assemblies, particularly cores with minor actinide-bearing assemblies. In addition, easy access to the core and core instrumentation will allow it to serve as a fast-spectrum-fuel and materials-irradiation facility.

4.4 U.S./France Collaboration

U.S. Secretary of Energy Spencer Abraham and Commissariat à l'énergie atomique Chairman Pascal Colombani signed a bilateral agreement on July 9, 2001, to jointly fund innovative U.S./French research in advanced reactors and fuel cycle development. The U.S./France collaboration was the first I-NERI agreement to be fully implemented; twenty-one U.S./France collaborative research projects have been awarded since FY 2001.

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 R&D that will benefit a range of anticipated future reactor and fuel cycle designs.

Work Scope Areas

Following are the R&D topical areas for the U.S./France collaboration:

- Advanced gas-cooled reactors
- Advanced fuel and materials development
- Radiation damage simulation
- Hydrogen production using nuclear energy

Project Summaries

Work on three collaborative projects initiated in FY 2006 is continuing. Work has recently begun on one FY 2007 and one FY 2008 project as well.

A listing of the I-NERI U.S./France projects that are currently under way follows, along with summaries of the accomplishments achieved in FY 2008.

Directory of Project Summaries

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

PRA-Aided Design of Advanced Reactors With an Application to GFR Safety-Related Systems

PI (U.S.): G.E. Apostolakis, Massachusetts Institute of Technology (MIT)

PI (France): F. Bertrand, Commissariat à l'énergie atomique (CEA)

Collaborators: None

Project Number: 2004-010-F

Program Area: Generation IV

Project Start Date: April 2005

Project End Date: April 2008

Research Objectives

The objective of this project was to develop a conceptual design of a decay heat removal system for the gas-cooled fast reactor (GFR). This system will function during both normal modes of operation (including shutdown and refueling) and accident conditions. Researchers evaluated the system under a range of scenarios, including loss-of-coolant accidents (LOCAs), station blackout, and anticipated transients without scram.

For GFRs to meet the high safety assurance standards expected of new reactor designs, they must employ a reliable decay heat removal system. Probabilistic risk assessment (PRA), which has matured over the last 30 years, is expected to play a key role in all aspects of system design and safety. PRA will allow the designers to build on the vast array of applications already developed for light water reactors and other reactor types.

Two major issues must be addressed in order to take full advantage of PRA capabilities for advanced reactors: 1) PRA models for passive systems must be developed and 2) probabilistic goals must be developed to determine the acceptability of a design. The use of PRA in design implies that probabilistic goals exist; however, the current licensing framework is largely deterministic, which is not expected to change substantially in the near future.

The U.S. Nuclear Regulatory Commission and the International Atomic Energy Agency (IAEA) are working to establish probabilistic goals. It is uncertain whether a new plant design would need to satisfy both deterministic and probabilistic criteria, particularly if a design option meets probabilistic goals but fails the deterministic criteria.

Research Progress

Two major issues have been addressed in this project: 1) the development of PRA methods for passive systems and 2) the use of PRA in design.

The researchers' methodology was applied to the design of a two-loop passive decay heat removal (DHR) system that was part of the iterative design process for a helium-cooled fast reactor. The accident sequence investigated was a small LOCA.

The present work uses a mechanistic model for a safety function and investigates the impact of uncertainties in the input parameters on the probability of function failure. This process reveals the significance of certain possible values of the input parameters and leads to design changes that affect these values.

An important finding was the discovery that the smaller pressure loss through the DHR heat exchanger (compared to the loss through the core) would make the flow bypass the core through one DHR loop, if two loops operated in parallel. This finding is a warning against modeling only one lumped DHR loop and assuming that n of them will remove n times the decay power.

Because the unreliability of the original design was too high, several design changes were proposed to reduce it. Sensitivity analyses showed that the heat transfer coefficient in the containment structures is important. Even after the design modifications, the calculated conditional (given a LOCA) failure probability of the passive decay heat removal system was deemed to be unacceptable and led the MIT design team to adopt an active DHR as the main mode of decay heat removal for the GFR.

The reliability methods for passive systems methodology has been successfully applied to several passive systems. In FY 2007, the CEA performed a reliability assessment of the 2,400-MWth GFR natural circulation DHR in pressurized situations.

Since FY 2006, in support of the design of its 2,400-MWth GFR, the CEA has been developing a preliminary Level 1 problamatic safety analysis (PSA) to demonstrate the ability of this tool to orientate safety design improvements at a conceptual design phase (the systems and components are still being developed and refined). The present scope of this PSA is limited to full power operating state and to a few representative initiating event families: spurious reactor trip, small and large LOCAs, and loss of off-site power. The work already performed is based on a preconceptual design of the 4th-generation GFR designed by the CEA and thus contains many assumptions with respect to the system configurations and the plant performance.

In parallel, considering that the reliability databases related to existing designs are not fully relevant, the CEA has developed a generic approach aiming at building a component failure rate database for innovative designs. An application to the CEA 2,400-MWth GFR components was carried out in FY 2007 in order to supply the above-mentioned preliminary Level 1 PSA with reliability data as relevant as possible.

In FY 2007, a preliminary safety analysis of the 2,400-MWth GFR as currently designed by the CEA was performed. Accidental situations have been investigated from the deterministic point of view, using the insights of the preliminary Level 1 PSA. These investigations allowed researchers to underline the positive features of the safety architecture and of the corresponding system to control pressurized and depressurized transients, as well as to identify the weakness of several provisions that are still to be improved.

In FY 2008, researchers planned to assess the depressurized situations in the framework of the INPRO Collaborative Project PGAP (Performance Assessment of Passive Gaseous Provisions) of the IAEA.

Planned Activities

This I-NERI project has concluded.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Sulfur-Iodine Integrated Laboratory-Scale Experiment

PI (U.S.): Paul Pickard, Sandia National Laboratories

PI (France): Philippe Carles, Commissariat à l'énergie atomique (CEA)

Collaborator: General Atomics (GA)

Project Number: 2006-001-F

Program Area: NHI

Project Start Date: April 2006

Project End Date: April 2009

Research Objectives

This sulfur-iodine (S-I) thermochemical cycle is one of the promising cycles for large-scale hydrogen production using high-temperature, advanced reactors. The U.S. Department of Energy and the CEA are investigating the S-I cycle because of the potential for high efficiency and compatibility with gas-cooled reactor outlet temperatures. The objective of this project is to develop an integrated laboratory-scale experiment for the S-I process that can be used as the technical basis for future technology selection and scale-up decisions. In FY 2006, the three major reaction sections of the S-I cycle were developed as stand-alone units: 1) the Bunsen primary reaction section, 2) the hydrogen-iodide (HI) decomposition section for hydrogen generation, and 3) the sulfuric acid (H_2SO_4) decomposition section. Stand-alone tests on these three sections established approaches for process chemistry, materials of construction, controls, and diagnostics. In FY 2007, the three sections were assembled as an integrated laboratory-scale (ILS) experiment. During FY 2008, testing of the individual sections was completed and initial integrated experiments were conducted. These experiments will provide information on system controls, the effects of cross-talk among sections, and potential corrosion product contamination effects on catalysts and process chemistry.

Research Progress

During FY 2008, assembly and integration of the S-I integrated lab-scale experiment was completed (see Figure 1). The high-temperature acid decomposer (O_2 generation section) and the HI decomposer section (H_2 generation stage) completed a sequence of tests to prepare for integrated operations. These tests provided process chemistry and control parameter information to support fully integrated testing. After early tests on the Bunsen primary reaction section, it was determined that additional diagnostics were needed; therefore, modifications were initiated in late FY 2008. The status of the testing completed to date is summarized below.

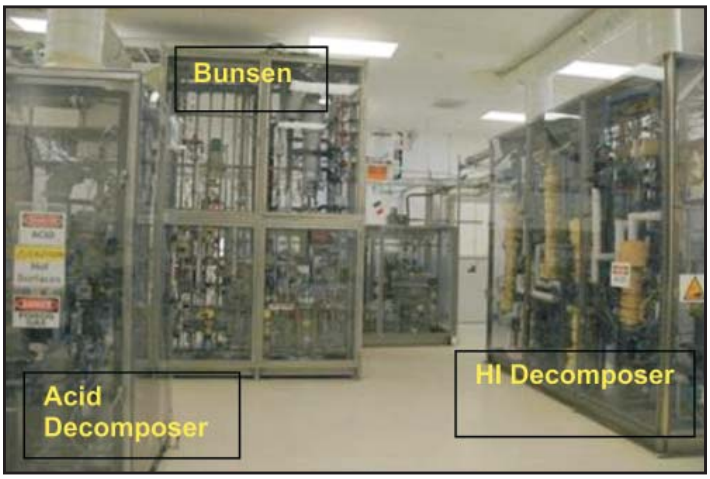


Figure 1. S-I integrated lab-scale experiment installed at GA.

Bunsen Section. Major modifications to the Bunsen section have been completed, including 1) the installation of a new Bunsen reactor equipped with Raching ring packing and differential pressure cells for enhanced process control, 2) a redesign and modification of the system for transferring iodine from a storage skid to the Bunsen section, 3) additional integrated testing with the acid decomposition, and 4) the initiation of testing for the new Bunsen reactor.

HI Decomposition Section. The HI decomposition section is based on an extractive distillation approach that separates iodine (I_2) from the HI_x feed stream before

decomposition. The HI section has now been successfully operated multiple times to produce hydrogen. The HI recycle system was successfully tested at pressures of up to 100 psi to increase hydrogen production for the same total flow rates (see Figure 2).

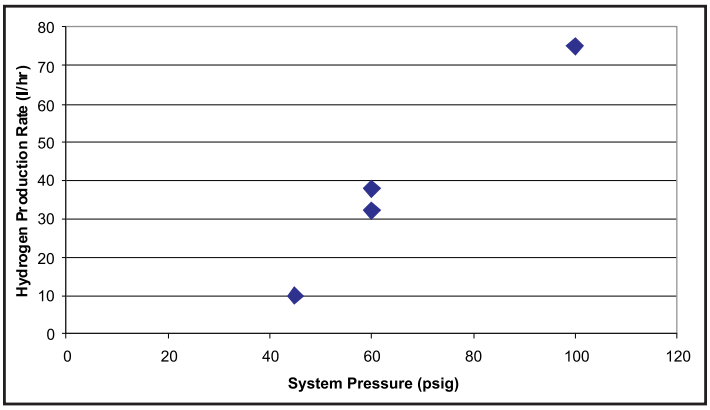


Figure 2. H_2 production rate as a function of pressure in the HI decomposition section.

Several modifications were made to improve performance. The HI reactor was replaced with a smaller unit to decrease the time for preheating the unit and reaching equilibrium while in operation. Using the smaller reactor, hydrogen has been produced at a rate from 10 to 75 L/hr. Ta alloy corrosion problems were encountered as a result of hydrogen embrittlement of Ta vessels and piping. Most of these issues have been mitigated for the ILS with either modified operational conditions or periodic replacement. Longer-term solutions are being developed and will be tested in FY 2009.

Sulfuric Acid Decomposition Section. Researchers completed sulfuric acid decomposition experiments with the integrated vaporizer, superheater, catalytic decomposer, and recuperator in the SiC bayonet. Experiments were run as a function of peak temperature to determine the SO_2 production rate. The results in Figure 3 show that reasonably high rates are achieved at temperatures over

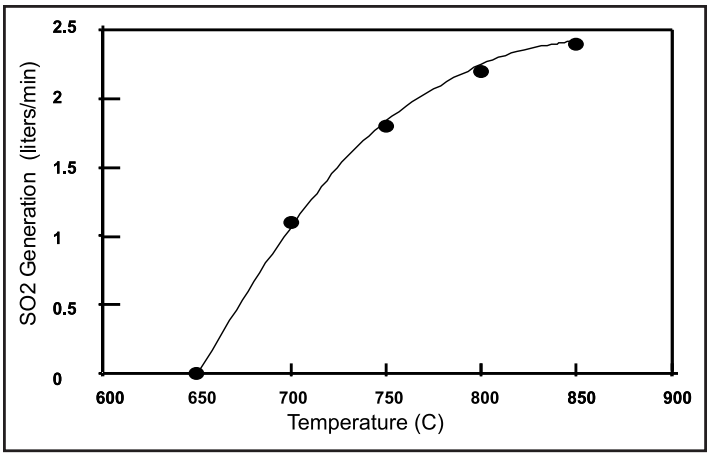


Figure 3. SO_2 generation rate as a function of peak (external) temperature in the SiC bayonet heat exchanger.

700°C. The S-I cycle could be operated with temperatures in the 750°C range with only moderate increases in recycle, which would have a small impact on overall efficiency.

System modifications included incorporation of an acid neutralization step to the process cooling water system to prevent corrosion of the circulating water chiller used in the process—an increased efficiency of the acid concentrator and the addition of a pressure control system to maintain a constant pressure in the acid decomposer. The acid decomposition process has been successfully operated multiple times over the past year in the stand-alone and integrated modes with the Bunsen section.

Planned Activities

The HI and acid decomposition sections of the S-I process are ready for additional integrated testing. All modifications have been completed and successful; stand-alone tests have been performed with the new equipment. Major equipment modifications to the Bunsen section have been completed and the process is ready to test the newly installed Bunsen reactor. Testing of the Bunsen reactor will be performed in the stand-alone and integrated modes.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

High-Temperature Nickel-Based Alloys for VHTR Applications: Mechanical and Corrosion Testing

PI (U.S.): Richard Wright, Idaho National Laboratory

PI (France): Celine Cabet, Commissariat à l'énergie atomique

Collaborators: AREVA, Electricité de France

Project Number: 2006-002-F

Program Area: Generation IV

Project Start Date: January 2006

Project End Date: December 2009

Research Objectives

The very high-temperature reactor (VHTR) is a helium-cooled system operating with outlet temperatures exceeding 950°C. There are two main advantages of this concept—high yields for energy generation and the ability to supply high-temperature process heat for hydrogen production. The high temperatures impose challenging design requirements on structural materials, particularly for the intermediate heat exchangers. Solid solution strengthened nickel-based alloys are the most suitable materials for these high temperatures, particularly Inconel Alloy 617 and Haynes 230. However, these alloys are not fully American Society of Mechanical Engineers Code of Ethics of Engineers (ASME Code)-qualified for nuclear applications. Basic data are needed to achieve a complete understanding of their behavior at high temperatures.

In this project, researchers are investigating the mechanical properties of these alloys and surface/subsurface corrosion effects caused by helium impurities. As component integrity must be demonstrated over the entire operational lifetime of the component (on the order of 100,000 hours for the heat exchanger), the team will take into account the evolution of properties over time, specifically the effects of thermal aging.

The team has carried out a number of experiments on these alloys to determine the magnitude of degradation in mechanical properties after aging. In this project, several heats of Inconel 617 have been aged for up to 10,000 hours and the resulting changes in impact properties and tensile ductility have been determined so researchers could investigate aging effects and the heat-to-heat variability in the aging effects. The team carried out shorter-term aging tests on Haynes 230 to begin characterization of potential aging effects. The ultimate goal of these tests is to use results properties over a very long duration and to predict the component lifetime. The team is also characterizing the impact of exposure to VHTR atmospheres.

There is insufficient information on creep-fatigue behavior and high-temperature weldment properties to allow ASME Code qualification of these alloys. The goals for specific activities during the year on this project included developing testing capabilities and test methods for creep-fatigue testing of alloys for potential heat exchanger applications in the VHTR systems. Cycles to failure and the variability of creep-fatigue life will be bounded for Inconel 617 and Haynes 230.

Research Progress

Figure 1 shows experimental critical temperature data for oxide stability as a function of the CO partial pressure for Inconel 617 and Haynes 230. All atmospheres within

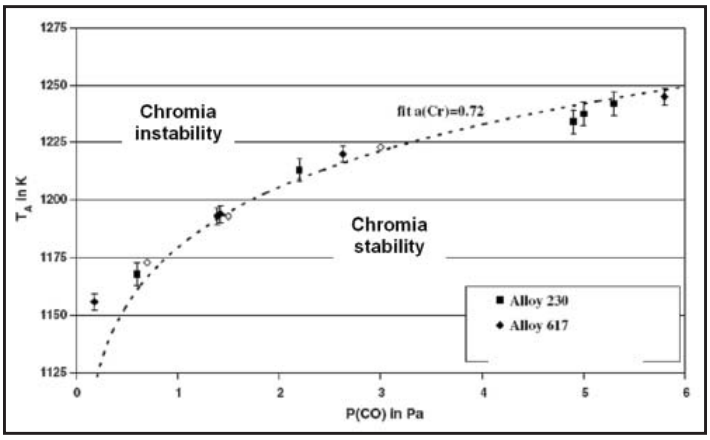


Figure 1. Critical temperature for the chromia stability versus CO partial pressure in helium; the open symbols represent Inconel 617 from the literature.

the upper domain will cause surface chromia reduction and subsequent rapid interaction with the gas phase, especially the carbon-bearing species. On the other hand, helium corresponding to the area below the curve will stabilize the surface chromia. The critical temperature thus represents the upper application temperature for the materials.

Reduction of the chromia scale can have a profound impact on the properties of these alloys. In typical VHTR atmospheres, either decarburization of the alloy can occur with reduced strength and microstructural stability or carburization and embrittlement can occur. Figure 2 shows the room temperature tensile properties of Inconel 617 after exposure at 900°C for 1,000 hours in oxidizing conditions where the protective scale is intact and in severe carburizing conditions. It is evident from the figure that carburization results in nil ductility for this alloy.

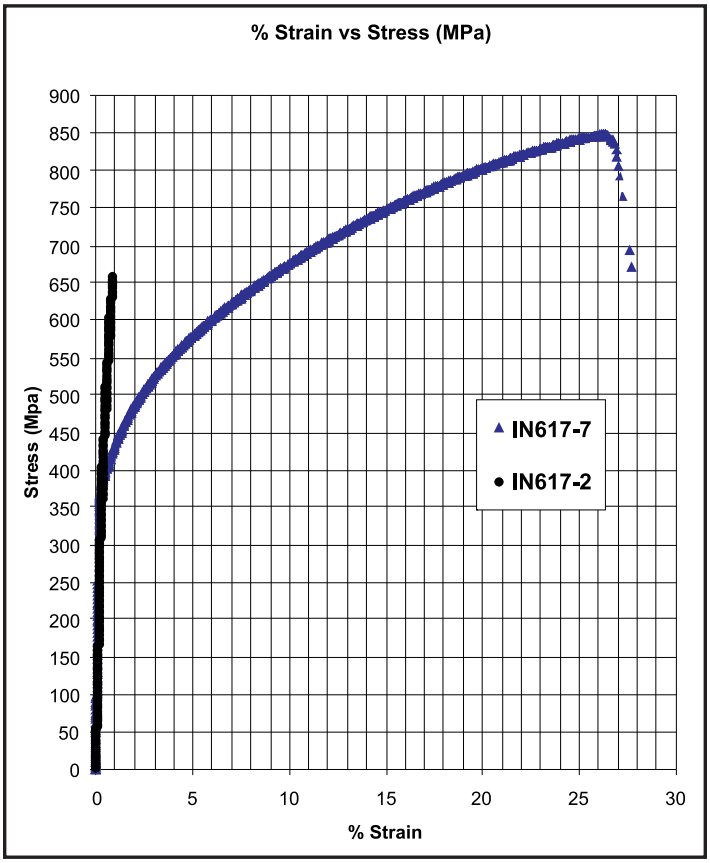


Figure 2. Room temperature tensile stress-strain curves for Inconel 617 that has been oxidized (IN617-7) and carburized (IN617-2) at 900°C for 1,000 hours.

A series of experiments is under way to generate creep-fatigue data. Multiple tests using nominally identical conditions are being run on both Haynes 230 and a controlled chemistry version of Inconel 617 to determine the variability that can be anticipated in testing. The controlled chemistry (CCA) Inconel 617 conforms to the American Society for Testing and Materials specification for this alloy; however, several of the alloy elements are near the maximum of the allowed range. This alloy was developed for a fossil energy application and is designed to have optimum strength at temperatures somewhat lower than those of interest to the Next Generation Nuclear Plant (NGNP) program. Cycle-to-failure data as a function of tensile hold time from these tests are plotted, along with values determined previously in the U.S. Generation IV testing program for Inconel 617 plate in Figure 3 for creep-fatigue tests performed at 800°C. The black symbols represent tests carried out in the NGNP research and development (R&D) program on 20 mm-thick plate material from a commercial vendor. The colored symbols (labeled with the alloy number designation) represent recent results for Haynes 230 and CCA Inconel 617.

The tests that have been completed on CCA Inconel 617 to date have lower cycles to failure compared to conventional Inconel 617. At the 800°C test, temperature of the CCA version of the alloy is expected to have lower ductility compared to the Inconel 617 plate previously tested because the CCA alloy has higher aluminum (Al) and titanium (Ti) content and will have a larger volume fraction of γ' (Ni_3Al,Ti) precipitates. Haynes 230 has low Al and Ti content and appears to have somewhat better creep-fatigue behavior at 800°C. Additional testing of both alloys

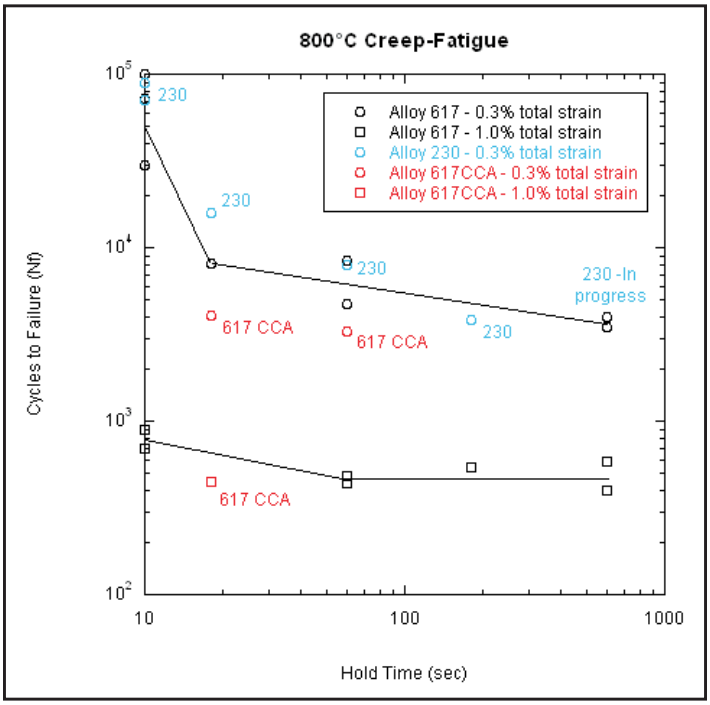


Figure 3. Cycles to failure in creep fatigue as a function of the tensile hold time. The black symbols represent tests carried out in the NGNP R&D program on 20 mm-thick plate material from a commercial vendor. The colored symbols (labeled with the alloy number designation) represent recent results for Haynes 230 and for a CCA version of Inconel 617.

at 1,000°C is under way. At this temperature, there should be little or no γ' formation and it is expected that all three of the alloys will exhibit similar behavior.

Planned Activities

Researchers will continue to focus on environmental effects to characterize the influence of longer-term exposure under benign (slightly oxidizing) conditions on the mechanical properties in post-mortem testing. There will also be a parallel effort to study the effect of decarburizing helium and to examine its effect on microstructure and properties.

The team will continue to perform systematic characterization of the creep-fatigue behavior of Haynes 230 and a CCA version on Inconel 617. They will complete testing of the alloys at 800°C and for total strain ranges of 0.3 percent and 1 percent. The primary focus of mechanical property testing in the coming year will be to investigate creep-fatigue behavior of both alloys over the same strain ranges at 1,000°C.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Comparison of Characterization Methods for Anisotropy and Microstructure of TRISO Particle Layers

PI (U.S.): John D. Hunn, Oak Ridge National Laboratory (ORNL)

PI (France): Olivier Dugne, Commissariat à l'énergie atomique (CEA)

Collaborators: AREVA, Université Bordeaux, Laboratoire des Composites Thermostructuraux

Project Number: 2006-003-F

Program Area: Generation IV

Project Start Date: June 2007

Project End Date: September 2009

Research Objectives

The resurgence of coated particle fuel as a cornerstone technology for the next generation of nuclear reactors requires improved characterization methodology for process development and product qualification. This project consists of two elements. Element 1 compares two systems for measuring the anisotropy of the pyrocarbon layers in tri-isotropic (TRISO)-coated particle fuel. Pyrocarbon anisotropy is an important parameter related to fuel irradiation performance. High pyrocarbon anisotropy can lead to non-uniform shrinkage in the pyrocarbon layers that may result in cracking and failure of the coating layers. ORNL measures anisotropy using their newly developed, two-modulator generalized ellipsometry microscope (2-MGEM). CEA uses a more traditional optical polarimeter technique called RAPAX. Element 2 investigates the use of x-ray analysis as a non-destructive technique for quality control.

Research Progress

During FY 2008, both CEA and ORNL completed optical anisotropy analysis of pyrocarbon layers grown on various TRISO particles of particular relevance to U.S. and French advanced gas reactor (AGR) fuel development programs. The ORNL technique used the 2-MGEM to obtain images of the diattenuation [$N = (R_{max}-R_{min})/(R_{max}+R_{min})$, where R_{max} (R_{min}) is the maximum (minimum) polarized reflectivity]. The CEA technique used a more traditional polarized microscope, RAPAX, with a rotating analyzer, from which the optical anisotropy factor (OAF = R_{max}/R_{min}) was obtained. The two measurements are related through the expression OAF = (1+N)/(1-N).

One of the first observations was that the RAPAX measurements resulted in a large value of OAF, even for isotropic samples. The origin of this offset was quickly found to be the beamsplitter in the RAPAX microscope that directed the incoming light to the sample. ORNL and CEA developed a calibration procedure using an aluminum mirror with a known diattenuation of N=0. This calibration procedure has allowed for a direct comparison between the two techniques.

Each laboratory examined samples of known diattenuation. The three samples chosen were 1) an aluminum mirror, 2) single-crystal rutile (TiO_2) with the c-axis in the surface plane, and 3) highly oriented pyrolytic graphite (HOPG). The error limit of each data point was $\delta N \sim 0.001$ for the 2-MGEM measurements and $\delta N \sim 0.002$ for the CEA measurements. As can be seen from Table 1, the two techniques obtained the same value well within the error limit for the HOPG measurements. There appeared to be a minor discrepancy in the RAPAX results for the rutile crystal.

	Known values	2-MGEM	RAPAX
Al mirror	0.000	0.000	0.000
Rutile at 577 nm Rutile at 640 nm	0.087 0.085	0.088 ---	--- 0.078
HOPG	---	0.576 (577 nm)	0.577 (577 nm)

Table 1. Results of measurements of standard samples.

To analyze the mean and standard deviation (SD) of the pyrocarbon anisotropy on the TRISO particles using the 2-MGEM, 1,000 pixels to 2,000 pixels were selected from the diattenuation images of the IPyC and OPyC layers. RAPAX measurements on TRISO particles used 6 points to 12 points in each layer. With a few exceptions, the data from the two laboratories agreed within the average standard deviation in the diattenuation of each particle (SD N) and the error limits of each technique (see Table 2). Further analysis of the data is ongoing.

Pyrocarbon layers of the two samples of German reference fuel have been examined using transmission electron microscopy (TEM) with selected area electron diffraction (SAED). SAED analysis showed anisotropy trends similar to the results from optical anisotropy measurements. TEM examination of the IPyC layers showed a microstructure consisting of a polynucleated core with a surrounding area of higher anisotropy.

Planned Activities

Element 1 is complete, except for detailed comparative analysis of the data and submission of a final report. One potential area of additional work is an extended collaboration with the Centre d'Elaboration de Matériaux et d'Etudes Structurales in Toulouse to investigate the use of TEM darkfield imaging to quantify pyrocarbon anisotropy at the sub-μm scale and high-resolution TEM analysis to image the microstructure responsible for pyrocarbon anisotropy at the nm scale.

Element 2 consists of an x-ray analysis of ORNL and CEA German reference fuel at AREVA and ORNL. X-ray radiography is a non-destructive technique that can obtain structural information without introducing artifacts induced by metallographic preparation. Moreover, this technique can offer 3-D information about layer thickness, density, interface structure, and coating defects not available from standard quality control techniques. Emphasis will be on using x-ray to perform non-destructive measurement of layer thickness and analysis methods for various coated particle fuel defects.

Sample Name	IPyC				OPyC			
	N	SD N	OAF	SD OAF	N	SD N	OAF	SD OAF
AGR-1 Baseline fuel before compacting	0.0074	0.0024	1.0149	0.0049	0.0061	0.0028	1.0123	0.0057
	0.0073	0.0019	1.0147	0.0039	0.0062	0.0023	1.0125	0.0047
	0.0074	0.0023	1.0149	0.0047	0.0063	0.0024	1.0127	0.0049
	0.0137	0.0075	1.0277	0.0153	0.0070	0.0076	1.0142	0.0153
AGR-1 Baseline fuel after compacting	0.0091	0.0028	1.0184	0.0057	0.0107	0.0036	1.0216	0.0074
	0.0109	0.0034	1.0220	0.0070	0.0111	0.0034	1.0224	0.0070
	0.0129	0.0077	1.0261	0.0157	0.0159	0.0095	1.0324	0.0193
AGR-1 Variant 1 fuel before compacting	0.0054	0.0020	1.0109	0.0040	0.0056	0.0026	1.0113	0.0053
	0.0044	0.0018	1.0088	0.0036	0.0047	0.0013	1.0094	0.0026
	0.0105	0.0065	1.0212	0.0132	0.0055	0.0083	1.0111	0.0167
ORNL German reference fuel before compacting	0.0141	0.0029	1.0286	0.0060	0.0076	0.0025	1.0153	0.0051
	0.0171	0.0099	1.0347	0.0201	0.0066	0.0085	1.0133	0.0172
ORNL German reference fuel after thermal anneal simulating compacting	0.0188	0.0029	1.0383	0.0060	0.0122	0.0025	1.0247	0.0051
	0.0196	0.0079	1.0400	0.0161	0.0142	0.0071	1.0289	0.0144
HRB-21 Reference fuel before compacting	0.0192	0.0030	1.0392	0.0062	0.0126	0.0039	1.0255	0.0080
	0.0204	0.0082	1.0416	0.0167	0.0129	0.0084	1.0261	0.0170
HRB-21 Reference fuel after thermal anneal simulating compacting	0.0189	0.0029	1.0385	0.0060	0.0187	0.0032	1.0381	0.0066
	0.0165	0.0093	1.0336	0.0190	0.0227	0.0084	1.0465	0.0172
CEA German reference fuel before compacting	0.0115	0.0023	1.0233	0.0047	0.0062	0.0019	1.0125	0.0038
	0.0120	0.0089	1.0243	0.0181	0.0016	0.0060	1.0033	0.0120
CEA AGR-2 Baseline fuel development sample	0.0055	0.0022	1.0111	0.0044	0.0033	0.0017	1.0066	0.0034
	0.0053	0.0085	1.0106	0.0171	0.0001	0.0085	1.0002	0.0170

ORNL results in black

CEA results in red

Table 2. Results of pyrocarbon anisotropy measurement on TRISO particles.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Advanced Dispersion Strengthened Ferritic Alloys and F/M Steels with Fine Nanoscale Dispersions

PI (U.S.): Dave Hoelzer, Oak Ridge National Laboratory, and Stuart Maloy, Los Alamos National Laboratory

PI (France): Saclay Y. de Carlan, Commissariat à l'énergie atomique

Collaborators: None

Project Number: 2007-001-F

Program Area: AFCI

Project Start Date: June 2008

Project End Date: May 2010

Project Abstract

The primary objective of this proposal is the development of high-performance ferritic alloys strengthened by the dispersion of nano-size particles for advanced high-temperature nuclear technologies such as cladding for sodium fast reactors. The project focuses on the dispersion of Ti-, Y-, and O-rich nanoclusters in mechanically alloyed ferritic alloys containing 13%Cr to 18%Cr and nano-size nitrides (and possibly carbides) in 9%Cr tempered martensitic/ferritic alloys. The key to developing these alloys will be to obtain 1) fundamental knowledge of the processing and fabrication methods for achieving the nano-size particle dispersions in the microstructure, 2) the corresponding relationship between the microstructure and the deformation and fracture properties, and 3) ultimately the stability of the microstructure and mechanical properties during exposure to neutron irradiation.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

SiC/SiC Composites for Gen-IV Reactors

PI (U.S.): William E Windes, Idaho National Laboratory, and Lance Snead, Oak Ridge National Laboratory

PI (France): Frederic Ravel, Commissariat à l'énergie atomique

Collaborators: University of Bordeaux, Snecma Propulsion Solide

Project Number: 2008-001-F

Program Area: Generation IV

Project Start Date: April 2008

Project End Date: April 2011

Project Abstract

Ceramic composites, both carbon-based and silicon carbide (SiC)-based, are considered advanced materials for use in Generation IV nuclear systems at temperatures beyond the operating temperature limits for metallic heat-resistant alloys. SiC fiber-reinforced SiC matrix (SiC/SiC) composites are very promising for high neutron loading and fast flux in Generation IV applications (some portions of very high-temperature reactor control rods and most gas-cooled fast reactor core internals). The overall project objective is to address some of the most critical design issues for SiC/SiC components.

A similar I-NERI collaboration with France was initiated in FY 2005. The program consisted of three task elements, but only Task 1 was completed before funding was temporarily discontinued. The current project will draw from the previous one, although the plan has evolved since the initial collaboration.

In the current project, Task 1 is evaluation with a U.S. emphasis on long-term environmental and synergistic effects. Through both theoretical modeling and experiments, researchers will attempt to establish a reliable scheme for life prediction of SiC/SiC under a combined environment of high-temperature, mechanical loading, radiation, and weakly oxidative atmosphere. In Task 2, the research team will analyze, both experimentally and theoretically, relations between mechanical behavior of tubes and planar tensile specimens of the SiC/SiC composite.

During the previous collaboration period, both the United States and France designed and fabricated their reference composite materials and test articles. The U.S. reference SiC/SiC is a Hi-Nicalon™ Type-S, multilayered interphase, chemically vapor-infiltrated SiC matrix composite, while the French reference is a monolayered pyrocarbon interphase composite (otherwise identical with the U.S. reference material). During the course of this project, both research teams plan to design the composite architecture and components based on requirements and thermomechanical analyses. In this task, the teams will perform design and analysis independently and exchange information.

4.5 U.S./Japan Collaboration

An exchange of notes was signed on April 22, 2004, by U.S. Assistant Secretary John Wolf for Secretary of State Colin Powell, and by Mr. Keiichi Katakami, Minister of the Japanese Embassy, for Ambassador Kato.

To implement this bilateral collaboration, the Japanese government decided to sign implementing arrangements using the following two organizations: the Agency of Natural Resources and Energy of Japan (ANRE) and the Ministry of Education, Culture, Sports, Science, and Technology of Japan (MEXT). ANRE is the office responsible for nuclear technology for the Ministry of Economy, Trade and Industry (METI).

On May 26, 2004, the implementing arrangement with ANRE was finalized and signed by Mr. William D. Magwood IV, Director of the U.S. Department of Energy, Office of Nuclear Energy (DOE-NE), and Mr. Kusaka, Director-General of ANRE. An annex regarding I-NERI collaboration was signed on June 10, 2004, by Mr. Shane Johnson, DOE-NE's Deputy Director for Technology, and Mr. Shigeru Maeda, ANRE/METI's Nuclear Policy Division Director for Nuclear Energy Policy.

On February 8, 2005, DOE and MEXT signed an Implementing Arrangement concerning cooperation in innovative nuclear energy technologies.

Work Scope Areas

Following are the R&D topical areas for the U.S./Japan collaboration with ANRE:

- Supercritical water-cooled reactors
- Innovative light water technologies
- Oxide fuel processing for light water reactors
- Fuel technologies using solvent extraction
- Radioactive waste processing

Following are the R&D topical areas for the U.S./Japan collaboration with MEXT:

- Innovative nuclear reactor technologies
- Innovative processing technologies
- Fuel technologies using solvent extraction

Project Summaries

Work continues on the only remaining project in collaboration with MEXT, awarded in early FY 2006. This is the last of the I-NERI U.S./Japan projects currently under way. All projects awarded under the cooperative agreement with ANRE in FY 2005 were completed as of last year. A summary of this project follows.

Directory of Project Summaries

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Assessment of Irradiation Performance of ZrC TRISO Fuel Particles

PI (U.S.): Y. Katoh and L. Snead, Oak Ridge National Laboratory (ORNL)

PI (Japan): K. Sawa, Japan Atomic Energy Agency (JAEA)

Collaborators: Idaho National Laboratory (INL)

Project Number: 2006-001-J

Program Area: Generation IV

Project Start Date: March 2006

Project End Date: February 2010

Research Objectives

Zirconium carbide (ZrC) is a leading candidate to replace silicon carbide (SiC) as a coating material for tristructural-isotropic (TRISO) fuel particles. However, researchers must first determine the fundamental radiation effects of ZrC in order to assess its viability as a fuel coating. The objectives of this project are to 1) clarify the fundamental irradiation response of ZrC, both in the form of a coating on surrogate TRISO particles and in a zone-refined, high-purity, polycrystalline form; 2) evaluate statistical fracture strength properties of developmental ZrC coatings before and after irradiation; and 3) develop a failure model of ZrC-TRISO particles for fuel performance evaluation and safety design.

This project consists of the following tasks:

- Conduct post-irradiation examinations (PIEs) of zone-refined, high-purity ZrC samples to understand baseline irradiation effects
- Characterize mechanical properties of unirradiated developmental ZrC coatings
- Perform high-temperature neutron irradiation of ZrC-coated surrogate fuel particles and conduct a PIE
- Develop fuel performance models and study the thermomechanical response of ZrC-coated particles
- Examine chemical interactions between ZrC and key fission products to identify potential degradation

Research Progress

Optimization of ZrC Coating Process. Continuous deposition of the outer pyrocarbon (OPyC) layer immediately after the ZrC deposition is preferred in order to avoid coating failure during processing and handling. The ZrC/OPyC continuous coating process was successfully developed up to thicknesses of the ZrC and OPyC layers of about 27 μm and 48 μm, respectively, in the batch scale of 100 g (see Figure 1). In addition, it was found that the rough surface of the ZrC caused the large pores in the OPyC layer and the decrease in the uniaxial compressive failure strength of the coated particles as the potential issue.

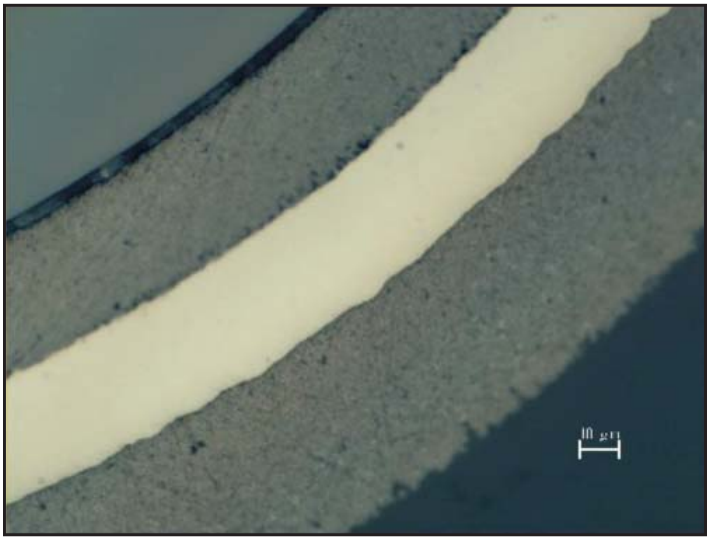


Figure 1. Cross-sectional micrograph of ZrC and pyrocarbon layers produced in a continuous deposition process.

Development of ZrC-TRISO Performance Model. A computational code to evaluate a pressure vessel failure fraction has been available for conventional SiC-coated TRISO fuel particles. However, plastic deformation of the third structural layer will have to be treated for ZrC TRISO-coated fuel particles because of the extensive ductility of ZrC at elevated temperatures. A modified model that can

treat the extensive plastic deformation of the ZrC layer was recently developed by JAEA, employing a finite element method (FEM), and was named Code-B. The new code basically consists of 1) an internal pressure calculation part, 2) a stress calculation part, and 3) a failure fraction calculation part. ABAQUS was chosen as the FEM code to calculate the stresses and the strains in the coating layers in Code-B. A benchmark calculation was carried out between Code-B and INL with a virtual stress-strain curve of the third layer. The calculation results fit well with each other, as shown in Figure 2.

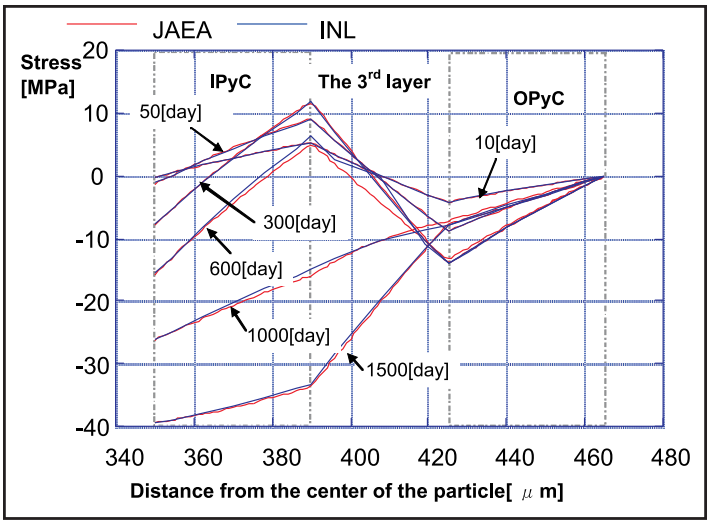


Figure 2. Radial stress distribution in TRISO fuel coating system calculated by fuel performance codes developed by JAEA and INL.

Development of Irradiation Capsules and Experimental Matrix. Capsules specifically for neutron irradiation of the tiny surrogate fuel particles at very high temperatures were newly designed in the High Flux Isotope Reactor (HFIR) at ORNL. These capsules were designed to hold approximately 50 particles, each of eight different lots, and to be accommodated in fixed rabbit tubes to be irradiated in flux trap of the HFIR. The target irradiation temperatures and the fast neutron fluences were 800°C and 1,250°C and up to approximately 6 x 10^{25} n/m^2, respectively. The surrogate particles were prepared by JAEA. Some particles had been heat-treated at 1,800°C for one hour in order to simulate the condition during the fuel compacting process. Particles with near-stoichiometric composition (C/Zr ratios in 1.03) were chosen to investigate the stoichiometric ZrC properties under irradiation, with the exception of one material with a C/Zr ratio of 1.35, which was included for comparison.

	Irradiation at <~1,000°C	Irradiation at >~1,300°C
Lattice parameter	Slight increase likely	Very slight increase?
Hardness	Increase	No change
Elastic modulus	Increase	No change?
Fracture toughness	Increase	No change
Thermal diffusivity	Decrease	Slight decrease
Irradiation creep	Significant	Significant

Table 1. Summary of neutron irradiation effects on various properties of zone-refined ZrC.

Characterization of Irradiated Properties of Zone-Refined ZrC. The effects of neutron irradiation at elevated—high to very high—temperatures on various properties and microstructures of ultra-high purity, zone-refined ZrC samples were examined. Specifically, changes in lattice parameters, indentation hardness, elastic modulus, indentation fracture toughness, thermal diffusivity, and bend stress relaxation irradiation creep were extensively examined. Results of the examination in the present study are summarized in Table 1.

Importantly, the previously identified possibility of very significant radiation-induced lattice swelling was denied. Moreover, the ambient temperature mechanical property appeared to be unchanged or slightly improved after irradiation, denying the possibility of grain boundary micro-cracking, which had historically been observed in sintered ZrC. Many of the irradiation-induced phenomena exhibited major transitions between the irradiation temperatures of <~1,000°C and >~1,300°C. As an example, thermal diffusivity is presented in Figure 3.

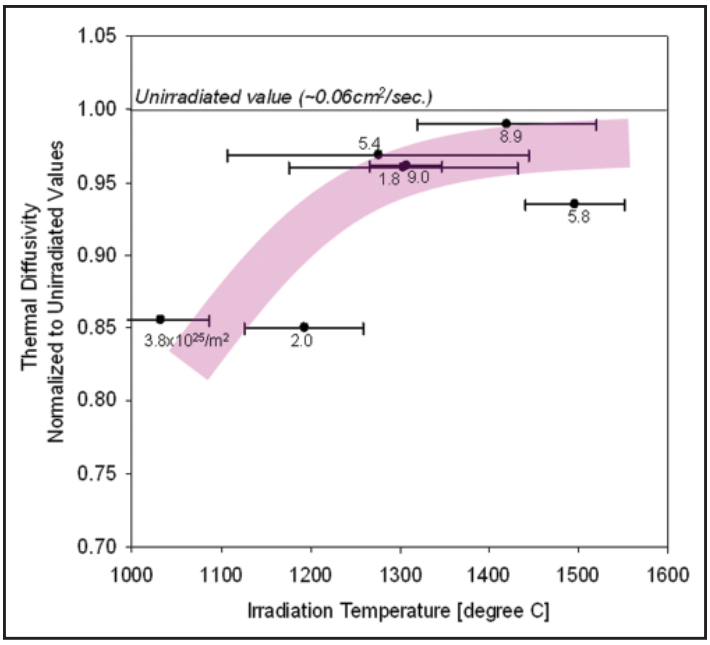


Figure 3. Thermal diffusivity of neutron-irradiated ZrC normalized to non-irradiated values. The horizontal error bars correspond to uncertainty of irradiation temperature.

Planned Activities

Researchers plan to accomplish the following activities during FY 2009:

- Continue non-irradiated and irradiated characterization of the developmental coatings
- Conduct modeling and code development for fuel performance evaluation
- Study fission products/ZrC chemical interactions
- Continue post-irradiation examination of the zone-refined ZrC samples that were previously irradiated
- Perform neutron irradiation study of surrogate ZrC-TRISO fuel particles

4.6 U.S./Republic of Korea Collaboration

William D. Magwood IV, Director of the U.S. Department of Energy, Office of Nuclear Energy, signed the first bilateral I-NERI Agreement on May 16, 2001, with Dr. Chung-Won Cho, Director General of Korea's Atomic Energy Bureau, signing for the Republic of Korea's Ministry of Science and Technology (MEST, which replaces the former MOST). The first U.S./Republic of Korea collaborative research projects were awarded in FY 2002, with a total of 35 projects awarded to date.

The U.S./Republic of Korea collaboration encompasses advanced technologies for improving the cost, safety, and proliferation resistance of nuclear energy systems. Research projects for collaboration have been selected competitively through an independent peer-evaluation process.

Work Scope Areas

Following are the R&D topical areas for the U.S./Republic of Korea collaboration:

2005 (4 projects):

- Hydrogen production by nuclear systems
- Advanced fuels and materials development

2006 (4 projects):

- Sodium-cooled fast reactor
- Advanced fuels and materials development

2007 (7 projects):

- Advanced gas-cooled fast reactor
- Sodium-cooled fast reactor
- Advanced fuels and materials development
- Next-generation reactor and fuel-cycle technology

2008 (3 projects):

- Advanced computational methods R&D
- Thermal-hydraulics
- Nuclear data/safeguards

Project Summaries

The final two remaining FY 2005 projects were completed during the past fiscal year. Research continued on four other collaborative projects awarded to Republic of Korea partners in FY 2006 and seven awarded in FY 2007. Three new projects were initiated in FY 2008.

Following is a listing of I-NERI U.S./Republic of Korea projects—those that are currently under way, those completed last year, and those newly awarded—along with summaries of FY 2008 accomplishments and abstracts of the new projects.

Directory of Project Summaries

* Completed in FY 2008

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Supercritical Carbon Dioxide Brayton Cycle Energy Conversion

PI (U.S.): James J. Sienicki, Argonne National Laboratory

PI (ROK): Seong-O Kim, Korea Atomic Energy Research Institute

Collaborators: None

Project Number: 2005-001-K

Program Area: Generation IV

Project Start Date: October 2005

Project End Date: October 2008

Research Objectives

The objectives of this project were to develop supercritical carbon dioxide (S-CO_2) Brayton cycle energy conversion systems and evaluate their performance when coupled to advanced nuclear reactors. Researchers focused on three Generation IV designs: 1) the sodium-cooled fast reactor (SFR), 2) the very high-temperature gas reactor, and 3) the lead-cooled fast reactor (LFR). The S-CO_2 Brayton cycle is expected to increase plant efficiency and reduce balance-of-plant costs relative to a Rankine steam cycle operating at the same reactor core outlet temperature. The project was organized into the following tasks:

- Development of design concepts for S-CO_2 Brayton power conversion systems
- Evaluation of S-CO_2 turbine and compressor performance
- Development of design concepts for compact sodium-to-CO_2 heat exchangers, recuperators, and coolers for use in the S-CO_2 Brayton cycle
- Analysis of plant transient behavior for an SFR coupled to an S-CO_2 Brayton cycle power converter
- Development of a system-level plant dynamics code for a S-CO_2 Brayton cycle energy converter coupled to an LFR
- Investigation and development of control strategies for the S-CO_2 Brayton cycle
- Dynamic analyses of plant transient behavior for specific control strategies
- Improvement of S-CO_2 Brayton cycle turbomachinery modeling based on interactions with a commercial turbomachinery vendor

Research Progress

All objectives for this three-year project have been met. The following is a summary of progress during the third year. Researchers developed a concept for an innovative, compact diffusion-bonded heat exchanger that provides a significantly reduced pressure drop relative to a Printed Circuit Heat Exchanger™(PCHE™). The concept incorporates airfoil type fins and crossflow between interconnected microchannels instead of separate zigzagged channels (Figure 1). The calculated pressure loss is reduced by 75 percent compared with a heat exchanger similar to a PCHE™ at the same heat duty (Figure 2).

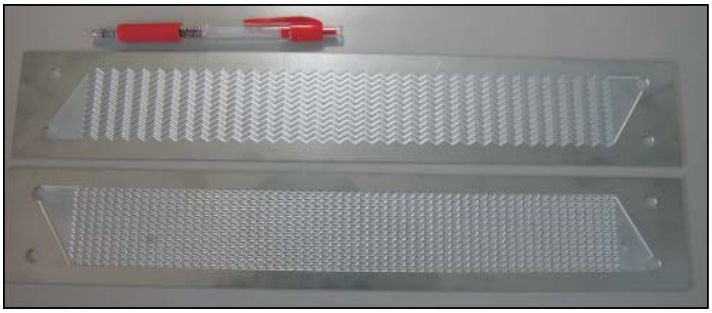


Figure 1. Chemically etched heat exchanger plates with separate zigzagged and interconnected airfoil channels.

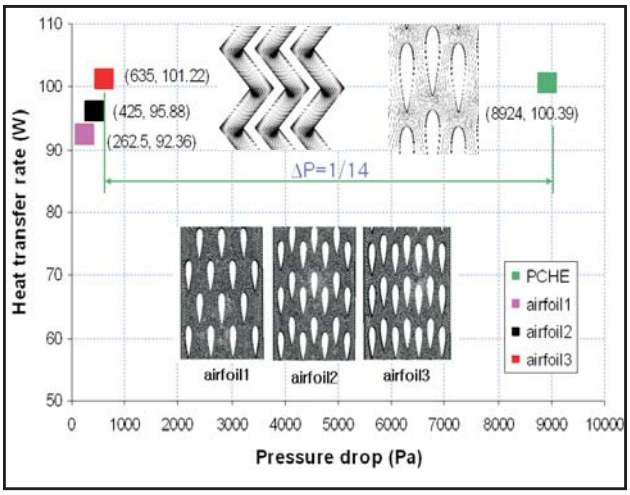


Figure 2. Calculated pressure drop for different channel configurations.

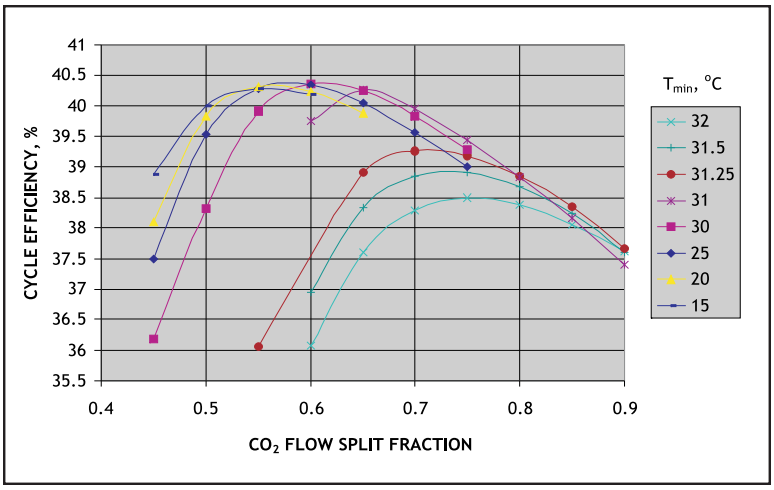


Figure 3. S-CO_2 Brayton cycle efficiency versus temperature and flow split between main and recompressing compressors for minimum pressure of 7.4 MPa.

Researchers continued developing a system-level plant dynamics code to model the S-CO_2 Brayton cycle. While previous efforts were directed at an S-CO_2 power converter coupled to a natural circulation LFR, they adapted the code to the advanced burner test reactor SFR preconceptual design. They further improved the automatic control strategy for optimal response of the overall S-CO_2 Brayton cycle through optimization of each individual control mechanism based on calculations of operational transients and postulated accidents involving additional accident initiators.

Researchers investigated the means for improving S-CO_2 Brayton cycle performance through changes in cycle layout and optimization of operating conditions. They found that a double recompression cycle with three compressors and three recuperators, intercooling between main compressor stages and reheating between turbine stages, did not improve cycle efficiency. The investigation reconfirmed the efficiency benefits of increasing heat exchanger and turbomachinery sizes, raising the cycle high-end pressure, and optimizing the cycle low-end temperature and/or pressure to operate as close to the pseudo-critical point as possible. Cycle performance can sometimes be improved if the cycle operates below the critical temperature on the low end, although this requires a larger and costlier cooler as well as the availability of a heat sink below 30°C, which is dependent upon the climate at the specific reactor plant site. For example, Figure 3 shows the cycle efficiency versus the minimum cycle temperature for a fixed minimum cycle pressure; an optimal minimum temperature of 30°C is indicated. Similarly, a minimum pressure of 7.45 MPa is determined if the minimum cycle temperature is fixed at 31.25°C, as assumed in previous evaluations.

Planned Activities

This I-NERI project has concluded.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Head-End Pyrochemical Reduction Process for Advanced Oxide Fuels

PI (U.S.): S. D. Herrmann, Idaho National Laboratory (INL)

PI (ROK): B. H. Park, Korea Atomic Energy Research Institute (KAERI)

Collaborators: University of Idaho

Project Number: 2005-004-K

Program Area: AFCI

Project Start Date: October 2005

Project End Date: September 2008

Research Objectives

Pyroprocessing can be a very effective method for recovering heavy metals and producing stable waste forms from the highly radioactive fission products in spent nuclear fuel. This is of great interest for the purpose of closing the fuel cycle for Generation IV fast reactors. However, pyroprocessing was originally developed for treating metal fuels, while the primary feed material for this process may be oxide spent fuel. A proposed solution to this problem is to develop a process for converting spent oxide fuel into a metallic form. This research project advanced the design of an economical, high-throughput oxide reduction process by focusing on two technical issues: 1) the behavior of fission products and 2) process scalability.

The United States and the Republic of Korea have been active in developing pyrochemical conversion methods for producing feed material that is compatible with pyroprocessing. Both countries favor an electrolytic reduction method, referred to as "oxide reduction." While significant advances have been made in oxide reduction over the last decade, including electrolytic reduction, a number of important technical issues need to be resolved in order to properly assess implementation of this technology.

Research Progress

To assess the behavior of fission products in the pyroprocessing of oxide fuels, researchers conducted a series of five bench-scale electrolytic reduction experiments with irradiated fast reactor mixed oxide (MOX) fuel. The MOX fuel was fabricated from UO_2-PuO_2 pellets with an initial Pu-to-total heavy metal fraction of 0.293. Three of the four fuel elements used in the reduction experiments were irradiated to a burnup of 17.7 percent, and the fourth was irradiated to 3 percent. The researchers separated the MOX fuel from its cladding by cutting, crushing, and sieving the fuel into particles less than 4 mm, as shown in Figure 1. They loaded the MOX fuel particulate into five permeable stainless steel baskets and immersed each basket in succession into a molten pool of LiCl-Li_2O at 650°C. The fuel in each basket was electrolytically reduced to metal, liberating the oxygen ions from the fuel to the salt phase where they were simultaneously oxidized to gas at a platinum anode. Analysis of the salt samples revealed the partitioning of cesium, barium, strontium, rubidium, and tellurium from the fuel matrix to the salt phase following each electrolytic reduction run. Analysis of the fuel following each run revealed a lower extent of metal oxide reduction than previously observed with spent light water reactor (LWR) fuel. Specifically, the extent of uranium oxide reduction ranged from 30 percent to 90 percent in the MOX fuel, compared to 99 percent reduction of uranium oxide in previous experiments with spent LWR

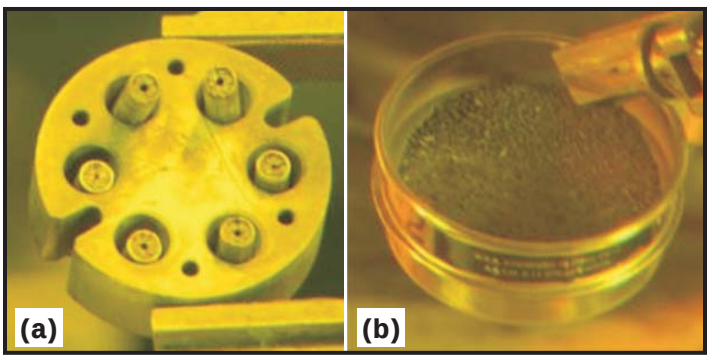


Figure 1. (a) Irradiated MOX fuel sections and (b) crushed and sieved MOX fuel particles.

fuel. Also, the electrical resistance on the platinum anode was higher during the MOX fuel runs than that observed during spent LWR fuel runs. The higher cell resistance and lower extents of reduction could be due to higher fission product concentrations in the system from the high burnup MOX fuel.

Researchers investigated three primary disposal options for the fission products that accumulate in the electrolytic reduction salt: 1) treating the reduction salt for reuse with zeolite-A to remove fission products via ion exchange and then removing the fission product-bearing zeolite for further processing into a ceramic waste form; 2) processing the salt into a ceramic waste form and discarding; and 3) allowing the reduction salt to carry over into the subsequent electrorefining process, where systems are already in place to convert spent electrorefining salt into a ceramic waste form. The researchers also investigated techniques to treat the reduction salt for reuse via salt-zeolite ion exchange. They conducted experiments in which zeolite-A particles were contacted with molten and solid particulate forms of LiCl – 1 wt% Li_2O loaded with up to 15 wt% CsCl and 5 wt% $SrCl_2$. Contact of zeolite-A with molten salt at 650°C revealed poor ion exchange of cesium and strontium due to the transformation of the zeolite structure at that temperature. However, no such transformation occurred at 500°C, and up to 99 percent of the cesium and strontium transferred from the salt to the zeolite. However, separation of salt and zeolite particulate following ion exchange at 500°C was inadequate to promote the reuse of the reduction salt with this technique. Consequently, an evaluation of the latter two disposition options was performed, and researchers concluded that carryover of the electrolytic reduction salt to the electrorefiner would alleviate the need for a separate waste form from the electrolytic reduction process, and would have a smaller impact on total ceramic waste volume generation.

Researchers completed bench-scale experiments that investigated the kinetics of uranium oxide to metal conversion in the electrolytic reduction process. Results of the two separate studies were used to support the development of a kinetic model for the process. Team members developed two models for assessing scale-up issues in the electrolytic reduction process: 1) a mathematical model for the electrolytic reduction of uranium oxide that was consistent with results from the bench-scale kinetic experiments and 2) a reactor model to define the fluid dynamics of the liquid and gas phases and the electric field in an electrolytic reduction process.

Researchers performed three electrolytic reduction runs at engineering scale with SIMFUEL in a modified reactor at KAERI's Advanced Spent Fuel Conditioning Process (ACP) hot cell facility. They first modified the equipment to simplify and improve electrolytic reduction operations, as shown in Figure 2. The SIMFUEL was comprised of U_3O_8 and nonradioactive fission product oxides in concentrations that were representative of spent LWR fuel. Since the electrolytic reduction process follows a voloxidation process at the test facility, the metal oxide compounds in the SIMFUEL were those that would be expected to form in voloxidation conditions. In the first engineering-scale electrolytic reduction run, the molten LiCl – 3 wt% Li_2O was sparged with argon in an attempt to increase the mass transfer of lithium oxide from the cathode to the bulk salt phase. However, no such benefit was observed, and the run was terminated. The second run proceeded without argon sparging and ran to completion with two intermittent lithium oxide additions. The third run was performed with a higher initial lithium oxide concentration (i.e., 5 wt%) and was completed without any intermittent lithium oxide additions. Post-test fuel samples from discrete locations in the cathode basket during the second run revealed an extent of uranium oxide reduction to metal ranging from 94.6 percent to 100 percent with an applied charge of 139 percent of theoretical. Reduction during the third run was 90.1 percent to 100 percent with an applied charge of 132 percent of theoretical. During the second and third runs, salt samples revealed the partitioning of cesium and barium from the fuel baskets to the salt phase.

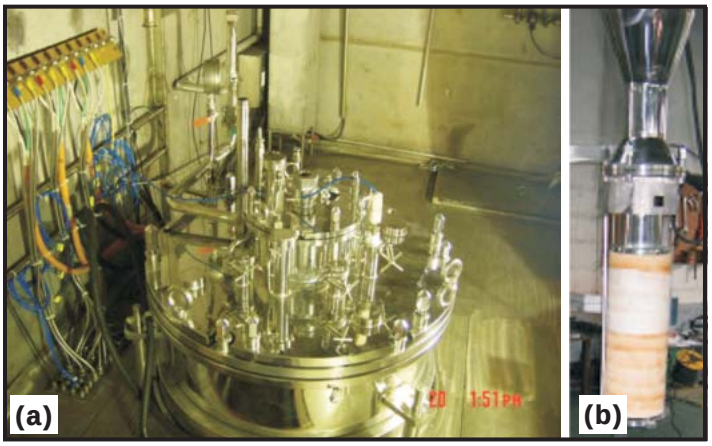


Figure 2. Modified engineering-scale equipment for electrolytic reduction of oxide fuels in KAERI's ACP: (a) top flange and (b) cathode.

The two principal research teams initially approached the electrolytic reduction process differently. Specifically, KAERI researchers operated its system with a nominal 3 wt% Li_2O concentration in LiCl with U_3O_8-bearing fuel in porous magnesia fuel baskets, while INL researchers

operated at 1 wt% Li_2O with predominantly UO_2-bearing fuel in permeable stainless steel baskets. Participants expected to settle on a common set of conditions for scaling up and advancing an electrolytic reduction process. However, at the conclusion of this project, the researchers determined that there is no need to change their respective approaches because of the programmatic directives driving each organization's design. This project has proven beneficial to expanding the understanding of oxide reduction fundamentals to support the design of economical, high-throughput oxide processes based on respective program needs.

Planned Activities

This project concluded in September 2008. Negotiations are under way to prepare and submit a new I-NERI proposal for oxide reduction development activities.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Core Design Studies for Sodium-Cooled TRU Burner Reactors

PI (U.S.): Won Sik Yang, Argonne National Laboratory (ANL)

PI (ROK): Yeong-Il Kim, Korea Atomic Energy Research Institute (KAERI)

Collaborators: None

Project Number: 2006-001-K

Program Area: AFCI

Project Start Date: October 2006

Project End Date: September 2009

Research Objectives

This project will develop conceptual core designs for sodium-cooled fast reactors (SFRs) to transmute recycled transuranic (TRU) elements. Transuranics are the dominant contributors to spent fuel radiotoxicity, long-term heat, and dose. The objectives are to 1) develop the core designs for TRU burner reactors, 2) perform verification and validation analyses, and 3) evaluate performance enhancements provided by innovative safety design features.

The burner core design task will examine various concepts to enhance the TRU transmutation rate under practical design constraints and will develop optimum core designs for different conversion ratios and power levels. Researchers will also investigate design options that minimize burnup reactivity loss and improve reactivity feedback coefficients.

The design method verification and validation task will consist of high-fidelity simulations and analyses of fast critical experiments. In particular, the team will perform detailed benchmark analyses for the ZPPR-15, ZPPR-21, BFS-73-1, and BFS-75-1 critical experiments. Researchers will analyze the sensitivities of cross sections and computational methods on neutronics performance parameters and evaluate the uncertainties of performance parameters.

To evaluate safety performance enhancements, researchers will analyze unprotected accident sequences with the ANL SASSYS-1 and KAERI SSC-K safety analysis codes for various combinations of innovative design features, focusing on inherent plant performance. They will also demonstrate enhancements to safety margins provided by advanced modeling techniques.

Research Progress

In order to assess the impact of power on core performance and safety characteristics of the SFR containing transmutation fuel, researchers upgraded the power from 1,000 MWth to 2,000 MWth. They developed the 2,000-MWth core concepts using both ternary metal and mixed oxide fuels, iteratively determining core size to yield power density similar to that of the 1,000-MWth core and a reasonably low sodium void worth. Figure 1 provides the radial layout of the 2,000-MWth core concept. The core consists of 384 driver assemblies divided into inner, middle, and outer zones of different TRU enrichments to achieve a flat power distribution. In order to reduce the sodium void worth, the active core height was reduced by 16 percent from the 1,000-MWth core and the number of driver assemblies was increased by more than double.

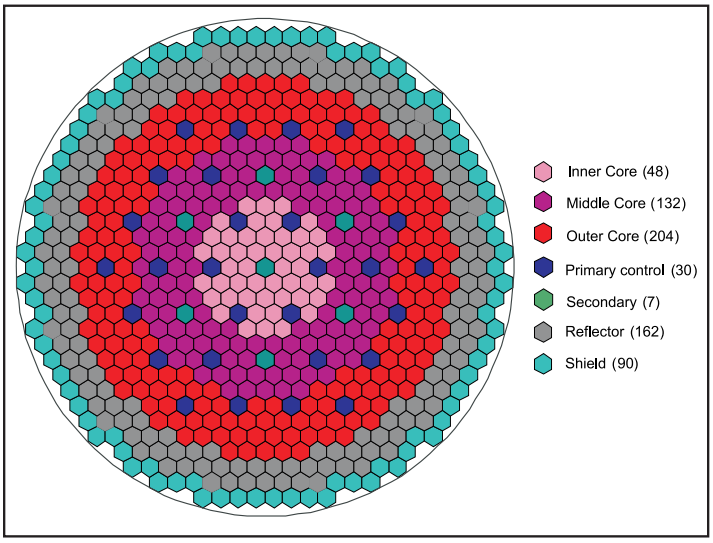


Figure 1. Radial Layout of 2,000-MWth Core Concept.

Researchers determined the fuel residence time to maximize discharge burnup within the peak fast fluence limit of HT9 cladding: 45 months and 60 months for the metal and oxide cores, respectively. The fuel volume fraction is 28.4 percent for the metal core and 34.94 percent for the oxide core, and the required average TRU enrichments are 23.1 percent and 27.1 percent, respectively. The core performance parameters are generally comparable to those of the 1,000-MWth core; however, the heavy metal inventory, TRU consumption rate, and burnup reactivity loss are higher. The reduced active core height results in increased linear power density, although the peak values remain within limits. The reactor kinetics parameters and reactivity feedback coefficients are comparable to those of the 1,000-MWth core. The effective delayed neutron fraction is 0.0032 for the metal core and 0.0031 for the oxide core. For both the metal and oxide cores, the sodium void worth is less than $7.00. Although detailed safety analyses are required, the 2,000-MWth advanced burner reactor (ABR) core concepts have similar passive safety trends as the 1,000-MWth core. The metal core has favorable passive safety features, while design fixes are required for the oxide core to meet sufficient passive safety conditions.

Researchers performed a core design study for large monolithic reactors in order to identify the most limiting factor in scaling up transmutation core concepts under a range of TRU conversion ratios between 0.44 and 0.75. They also studied the dependence of core performance and safety parameters on the TRU conversion ratio. They examined two design approaches: 1) a pancake core to maintain the power level-independent sodium void worth and 2) annular fuel pin designs with irradiation-resistant materials in the center to enhance reactivity feedbacks with single fuel enrichment. Pancake cores adopted region-dependent cladding thickness to flatten the power distribution with single fuel enrichment. With the pancake design approach, preliminary core designs were developed with metal fuel at three power levels: 600 MWe; 1,200 MWe; and 1,800 MWe. The annular fuel design was investigated exclusively at 600 MWe. The cycle length in both design approaches was 332 effective full power days. The TRU enrichments in the pancake core approach were 22 wt%, 30 wt%, and 38 wt%, while enrichment in the annular fuel approach was about 30 percent. The TRU conversion ratio was in the range of 0.44 to 0.75, and the burnup reactivity loss was in the range of 2.2 %Δk to 4.8 %Δk. For the same power rating, the pancake core design approach yielded a smaller core volume and thus a smaller fuel inventory. The calculated reactivity coefficients were adequate to provide negative reactivity feedback. The sodium density coefficient is almost invariant, and the axial coefficients became slightly less negative with increased power. As the radial expansion coefficient became slightly more negative in proportion to the power level, the net effect of power level increase did not cause the deterioration of reactivity feedback. In addition, an enhanced TRU transmutation in a low conversion core did not pose any serious safety concerns, except for a high burnup reactivity swing. To ameliorate this condition, the annular fuel core design study suggested either employing a moderator such as graphite, vanadium, or niobium, or using B_4C as an incomplete burnable poison function. However, neither of these approaches is entirely satisfactory, and a thorough design study to improve the burnup swing in a low conversion core needs to be undertaken.

In an effort to verify and validate the design analysis methods, the researchers analyzed the six benchmark problems for the ZPPR-21 critical experiments: Phases A through F specified in the *Handbook of Evaluated Criticality Safety Benchmark Experiments*. This analysis is based on the newly processed 2,082 group MC^2-2 files originating from a recently released ENDF/B-VII.0 basic nuclear data library. For all six configurations, the core multiplication factor determined with deterministic design tools agreed with Monte-Carlo solution within 0.27 %Δk, and there was no indication of any systematic bias. The statistics of differences between calculated values and specified benchmark experimental values showed similar agreement for deterministic and Monte-Carlo methods. However, both calculational methods deviate from the benchmark experimental values by approximately 0.66 %Δk for the pure uranium core. This deviation is considered partially due to the benchmark values determined with the correction factors derived from Monte-Carlo calculations based on the ENDF/B-V.2 data.

The research team also analyzed the BFS-73-1, BFS-75-1, and BFS-55-1 experimental data, including criticality, reaction rate ratios, fission rate distributions, and sodium void worth. They developed benchmark specifications for the fully heterogeneous assembly models for all three critical assemblies and will analyze the problems with ENDF/B-VI.6, ENDF/B-VII.0, JEFF-3.1, and JENDL-3.3 libraries. Predicted results by MCNP with the ENDF/B-VII.0 data are currently available. The calculated multiplication factors of BFS-73-1 and BFS-75-1 agreed well with measured values at 380 pcm and 270 pcm differences, respectively. However, researchers observed a large discrepancy of 930 pcm in the predicted multiplication

factor of BFS-55-1 (it is reported that the BFS-55-1 information will be updated). The heterogeneity effects on the multiplication factors, determined by comparing the results of heterogeneous and homogeneous configurations, are 211 pcm, 714 pcm, and 1,528 pcm for the BFS-73-1, BFS-75-1, and BFS-55-1, respectively. The value for BFS-73-1 is consistent with the IPPE result, about 150 pcm by the MMKKENO multi-group Monte-Carlo code. The spectral indices are in good agreement with measured values, but minor actinide reaction rate ratios (e.g., Am-241 and Am-243 fissions) are very poor. A BFS-55-1 assembly evaluation by the Monte-Carlo and deterministic transport calculations is now included to expand the validation activity as recommended. The TRANSX/TWODANT deterministic method has limited accuracy in reproducing the strong heterogeneity effect of BFS-55-1, but the deterministic calculations by homogeneous sub-assembly models predicted the effect of localized sodium voiding in a conservative manner. The reaction rate ratios predicted using three of the latest nuclear data libraries (ENDF/B-VII.0, JEFF-3.1, and JENDL-3.3) were in good agreement within 2 percent for plutonium (Pu-239) and uranium (U-235) fission reactions and significantly improved the prediction accuracy for Pu-240 fission reaction. However, the calculated values showed large discrepancies from measured data for U-238 fission reaction, which was not observed in the ENDF/B-VI.6-based library.

To evaluate safety performance enhancements, the research team analyzed the 600-MWe design, focusing on inherent protection against damaging consequences in low probability accident sequences involving multiple equipment failures. The primary analysis result is that a ternary metal-fueled core design demonstrated sufficient safety margins to coolant boiling and fuel damage in unprotected loss of flow accident sequences and unprotected loss of heat sink sequences. This is a direct consequence of the high thermal conductivity and low operating temperature of metal fuel, and the favorable negative feedback due to thermal expansion. However, the unprotected transient over-power (UTOP) indicated that margins to coolant boiling would not be adequate without additional mitigation features. The safety analysis determined that the total reactivity insertion of $1.02 upon the UTOP is the maximum allowed value. In accordance with this requirement, the primary control rod system is subdivided into startup and regulating banks, limiting the reactivity insertion of the regulating bank to less than $1.02.

Planned Activities

Researchers at KAERI will conduct the design optimization study for the large monolithic core concepts with iterations between core design and safety analysis, including safety analysis with the SSC-K code. The innovative fuel design and a large pancake core in a low TRU conversion ratio will be integrated to provide an optimum performance (maximum TRU transmutation with a reduced burnup swing). Considering various aspects of the evaluation results and the plant economics, an optimum core design will be suggested. The team will also analyze the ZPPR-15 experiments using design tools that KAERI developed and the recently-released ENDF/B-VII.0 nuclear data file.

ANL researchers will continue investigating the impact of power on core performance, including the development of alternative core concepts to improve the plant economy. This activity will entail a trade-off study between core size and sodium void worth. In addition, they will analyze the sensitivities of cross sections on neutronics performance parameters and evaluate the uncertainties of performance parameters for the 1,000-MWth ABR design.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Separation of Fission Products from Molten LiCl–KCl Salt Used for Electrorefining of Metal Fuels

PI (U.S.): Steven Frank, Idaho National Laboratory (INL)

PI (ROK): In-Tae Kim, Korea Atomic Energy Research Institute (KAERI)

Collaborators: University of Idaho

Project Number: 2006-002-K

Program Area: AFCI

Project Start Date: January 2007

Project End Date: September 2009

Research Objectives

The purpose of this project is to develop technologies for selectively separating fission products from salt used for the electrochemical treatment of spent nuclear fuel. During the second year, researchers completed equilibrium studies for salt-zeolite ion exchange. This enabled the use of an equilibrium model to calculate performance of the ion exchange process under various process configurations. Researchers initiated kinetics experiments to determine size and time requirements for an ion exchange unit operation for the electrochemical treatment flow sheet. They optimized experiments for fission product precipitation in electrorefiner salt, Group I fission product removal using the mineral analcime, and investigations of a zone-freezing method for increased concentration of fission products in the salt. These efforts have resulted in the selection of preferred technologies for experimental demonstration using radioactive salt during the third year of research.

Research Progress

Salt-Zeolite Ion Exchange. Researchers have completed the LiCl-KCl-MClx salt-zeolite contacting experiments with added surrogate fission product species La^{3+}, Pr^{3+}, Y^{3+}, and mixed salts containing Cs^{+}-Nd^{3+} and Cs^{+}-Sr^{2+}. Modeling development involved revising the equilibrium ion exchange model. They used new data obtained this year, along with previously collected data, collectively, to fit and verify the new equilibrium model. The ternary salt contacting tests were used to determine best-fit model parameters, while quaternary salt contact tests were used to validate the model. Comparison of quaternary salt experimental results to the new model predictions are shown in Figures 1a and 1b.

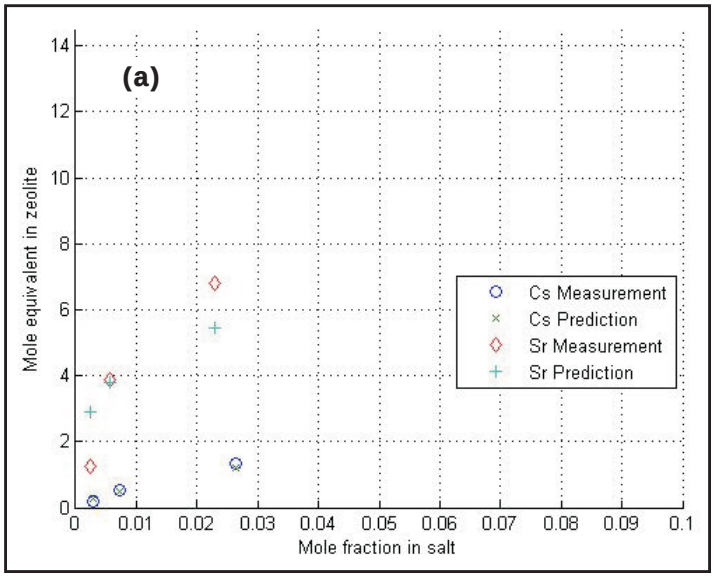


Figure 1a. Quaternary salts LiCl-KCl-CsCl-$SrCl_2$ measurements and predictions.

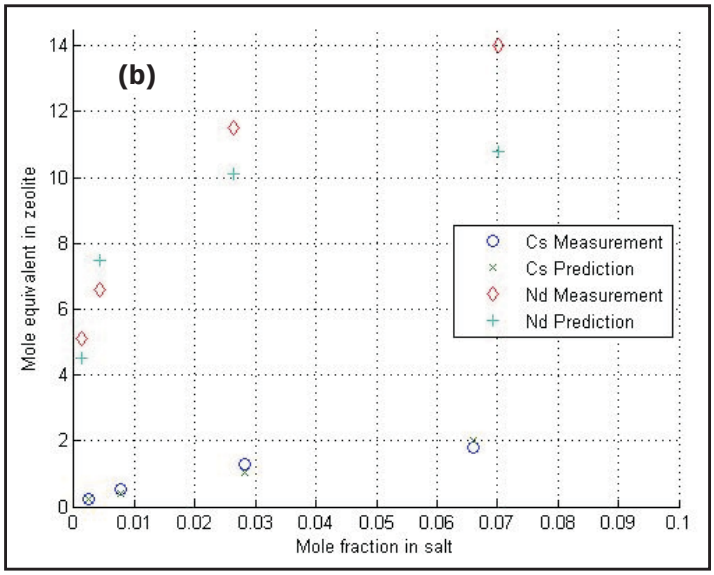


Figure 1b. LiCl-KCl-CsCl-$NdCl_3$ measurements and predictions.

Fission Product Precipitation. Researchers designed and installed lab-scale, rare-earth precipitation equipment with a 4 kg maximum batch size at KAERI. Using this apparatus, they carried out oxide and oxychloride precipitation experiments for an eight-rare-earth element (Y, La, Ce, Pr, Nd, Sm, Eu, and Gd) eutectic salt system. Reaction with oxygen gas converted Y, La, Pr, Nd, Sm, Eu, and Gd to their rare-earth oxychloride forms, while Ce and Pr were converted to their oxide forms (CeO_2, PrOCl). Since neither these rare-earth oxichlorides nor oxides are soluble in the molten salt, they were collected by free settling. Investigators also studied the removal behavior of analcime for radionuclides whose products are feasible for a solidification process. After contact with the molten salt, the analcime was transformed into leucite ($KAlSi_2O_6$) or pollucite ($CsAlSi_2O_6$). The removal capacity of analcime for Cs was approximately 0.2 meq/g to 0.8 meq/g at a given concentration. The amount required to remove 80 percent of Group I radionuclides in 100 g of salt waste is approximately 15 g to 25 g of analcime.

Zone-Freezing. Adaptation of a zone-freezing process resulted in 80 percent separation efficiency of Cs and Sr from electrorefiner salts. This zone-freezing method, when combined with oxide/oxychloride precipitation, has the potential to remove all the fission products, including Group I/II and lanthanides. A sequential separation process that consists of an oxygen sparging and zone freezing has been devised, as illustrated in Figure 2. Work continues on zone-freezing experiments in the eutectic salt-CsCl-$SrCl_2$-NaCl-$BaCl_2$ system and sequential separation experiments in the salt-Group I/II-rare-earth system to evaluate their fission product co-removal ability.

Planned Activities

Final Separations Technology Assessment. Investigation of batch fission product removal from electrorefiner salt by ion exchange with zeolite has been shown effective with the use of non-radioactive surrogates. Likewise, chemical treatment of fission products in electrorefiner salt to form insoluble precipitates followed by precipitate removal from the salt has great potential for salt cleanup. The next phase of the project will be to perform testing using these two methods to remove fission products from actual high-loaded fission product electrorefiner salt. These tests will allow comparison between the two methods to determine the best means for fission product removal from electrorefiner salt. The overall experimental plan for this phase of the project is shown in Figure 3. Experiments will be conducted in hot cells in INL's hot fuels dissolution apparatus. Both ion exchange and oxygen sparging experiments will be run sequentially using separate batches of high-loaded fission product electrorefiner salt to allow a direct comparison of separation efficiencies.

Salt Removal Using Vacuum Distillation. This task will 1) develop a process for removing salt from the fission product precipitate layer and 2) recycle the purified salt back to the electrorefining process. In support of this task, fundamental studies of distillation were performed over the first two years, including the vaporization behavior of eutectic salt at different conditions, salt recovery efficiency, and characterization of refined salt. Efforts for next year will involve various experiments performed at KAERI with the lab-scale salt distillation/condensation apparatus to optimize the operating conditions for salt removal/condensation with respect to temperature, vacuum pressure level, and time.

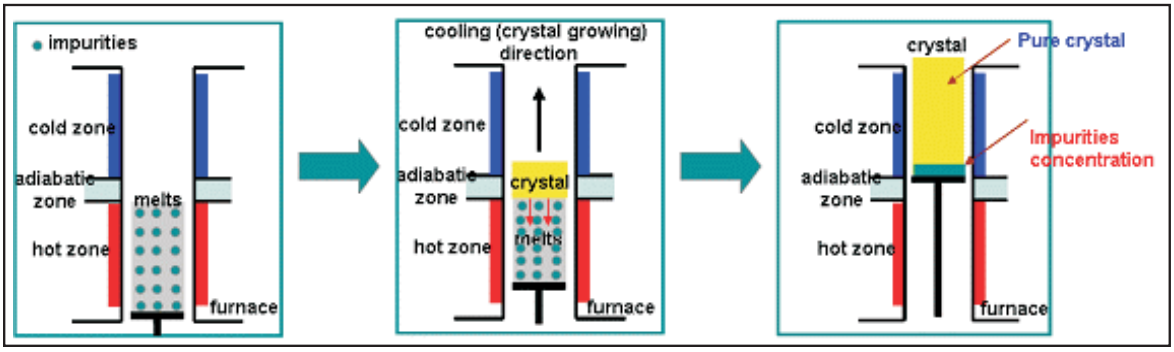


Figure 2. Zone-freezing process for fission product separation/electrorefiner salt purification.

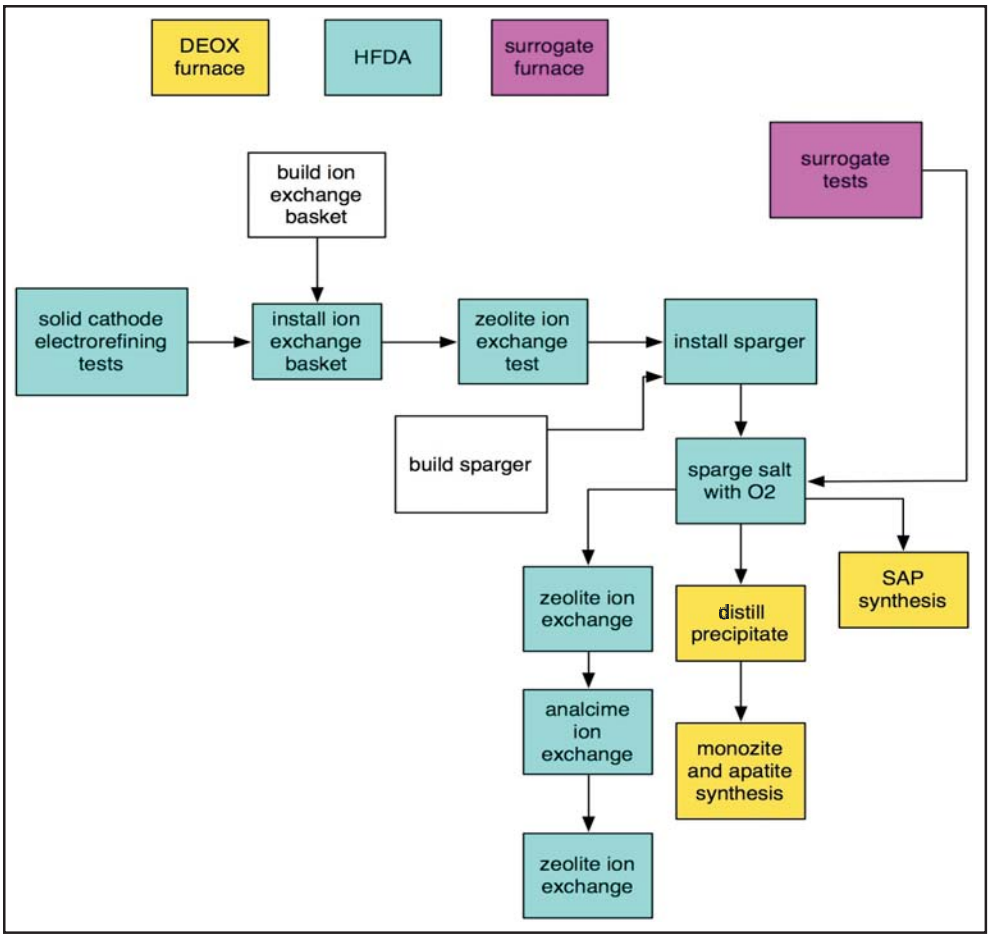


Figure 3. Proposed experimental plan for fission product removal from actual electrorefiner salt.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Crosscutting Materials for the Electrochemical Reduction of Actinide Oxides Used in Advanced Fast Burner Reactors

PI (U.S.): Christine T. Snyder, Argonne National Laboratory

PIs (ROK): Eung-Ho Kim and Jong-Hyeon Lee, Korea Atomic Energy Research Institute

Collaborator: University of Illinois at Chicago

Project Number: 2006-003-K

Program Area: Generation IV

Project Start Date: January 2007

Project End Date: September 2010

Research Objectives

The objective of this research is to develop new corrosion-resistant alloys and advanced surface engineering systems for use in pyrochemical processing technology for the treatment of spent nuclear fuel. Conditions found in the oxide reduction step during pyroprocessing represent one of the most aggressive thermal and chemical environments encountered by structural materials. The high-temperature molten salts, coupled with oxygen gas and convective flows, have been shown to cause corrosion in the most advanced nickel-based alloys. This milieu, along with exposure to uranium, plutonium, and other reactive elements, is a particularly challenging environment for the most advanced nickel-based alloys and ceramic coatings. These reactive elements may reduce oxide components of the vessel and coating material, resulting in degradation and production of low-temperature eutectic phases. Therefore, the main focus of this project is to develop novel corrosion-resistant alloys and a coating system that combines the thermal barrier qualities of graded ceramic layers with the corrosion protection of a metallic bond coat. While the specific goal is to develop a robust and stable materials system to advance the design limits presently imposed upon structural alloy materials during spent nuclear fuel oxide reduction, it is likely to benefit multiple technologies supported by the Department of Energy.

Research Progress

Alloy Fabrication. The strategy for high-temperature alloy design centers on the addition of elements to the nickel base metal that are the most reactive with oxygen in order to preferentially form a surface oxide. A continuous protective oxide scale on the metal alloy surface acts as a barrier to protect the underlying matrix from the Li_2O/LiCl and the flowing oxygen at the surface interface. For long-term use above 600°C, Cr_2O_3, Al_2O_3, and SiO_2 are the oxides of choice. Previous research focused on four new alloys—N1, N2, N3, and N4—with compositions similar to Inconel® 713LC. Subsequent testing showed that heat treating improves the corrosion resistance of the as-cast Alloys N1 and N2. Researchers have continued to survey elements with oxygen reactivity as alloy additions, using rare-earth elements to increase the high-temperature oxidation resistance of chromia- and alumina-forming alloys. They fabricated six additional alloys and corrosion tested them in Li_2O/LiCl with flowing oxygen. The rare-earth element cerium (Ce) at 0.1 wt% and 0.5 wt% was added to three of the six new alloys. Silicon (Si) additions ranged from 0.5 wt% to 2 wt%, while the Cr additions remained the same as the previous values. The most resistant, as-cast alloys based on weight loss were those with the same overall composition as Alloys N1 and N2 with additions of 0.5 wt% Ce and 1 wt% Ce, respectively.

The as-cast Alloys N5 through N10 each contained segregated phases and precipitates. In order to homogenize the microstructure, researchers performed five different heat treatments in argon gas for each alloy. The highest temperature profile produced the most improvement in homogenizing the segregated phases.

Alloy Casting Mold Comparison. Alloys cast from a metal mold exhibited a shrinkage cavity and a visible center line crack owing to the high cooling rate. Researchers rectified these problems by casting alloys using a sand mold inside a vacuum induction furnace. These ingots displayed no shrinkage cavities or cracks owing to a slower cooling rate and improved distribution of the heat from the

center to the outside of the mold. Melt contact with the cooler walls of the metal mold caused surface solidification to occur.

Corrosion Testing Results. Corrosion testing methodology consisted of suspending sample coupons in a molten LiCl-3 wt% Li_2O salt at 650°C with Ar/10 percent O_2 gas flowing through the melt. After the testing period, the coupons were removed and cleaned by ultrasonic washing, dried for 24 hours, and weighed. Researchers then analyzed the corrosion products and coupons using x-ray diffraction (XRD), scanning electron microscopy (SEM), and energy dispersive x-ray spectroscopy.

Figure 1 shows the calculated corrosion rate of as-cast Alloys N5 through N10. In evaluating the developed material for use in an engineering application, the corrosion rate should ideally be less than 0.5 mm/year. As the corrosion rate of Alloys N6 and N7 greatly exceeded this requirement, those alloys were excluded from further analysis.

Figure 2 shows the sample weight loss over time. The linear increase of the weight loss suggests spalling of corrosion products from the surface of the matrix metal is occurring. If a continuous oxide scale was developing over the alloy surface, it would create a protective barrier and the rate of growth would decrease with time as the diffusion distance increased with increasing scale thickness.

Researchers analyzed the corrosion products obtained during the cleaning step of the adhered salt by XRD. Cross-sectional SEM images of the alloys corroded for 168 hours show that all the alloys developed an oxide layer on the matrix surface; however, the oxide layers of N9 and N10 appear more dense and continuous than those of N5 and N8. This result agrees with the weight loss data.

Coatings. The research team is focusing on plasma spray feedstock size effects on coating characteristics and on developing layered multicomponent oxide coatings. Of particular interest is the study of bond-coating material compatibility and strength of adhesion of the ceramic coating to the metal substrate.

Researchers prepared feedstock granule powder of yttria-stabilized zirconia (YSZ) and alumina-yttria (Al_2O_3-Y_2O_3) by spray drying in both conventional- and small-sized spherical powders of 10 µm to 50 µm and 10 µm to 30 µm, respectively. Using a mixture of Ar and He gases, they plasma-sprayed a Ni alloy substrate at 37 kW with YSZ and 5-alumina 3-yttria using both powder sizes. Conventional-sized feedstock powder

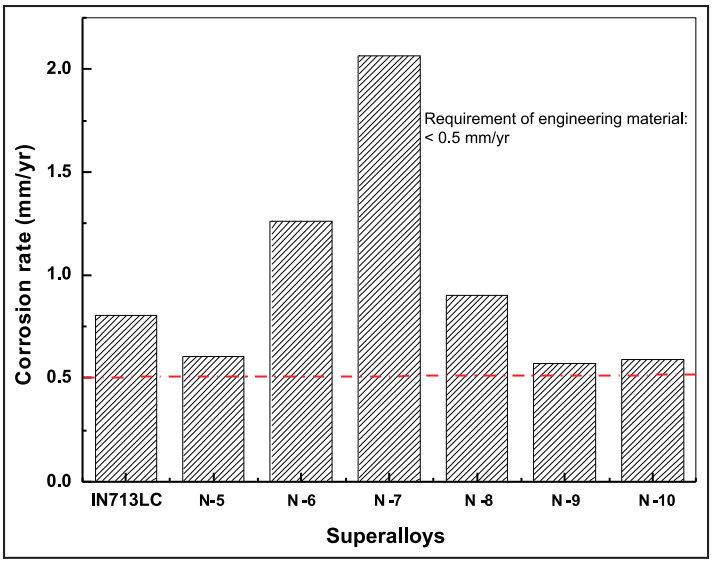


Figure 1. Corrosion rate of Alloys N5 through N10.

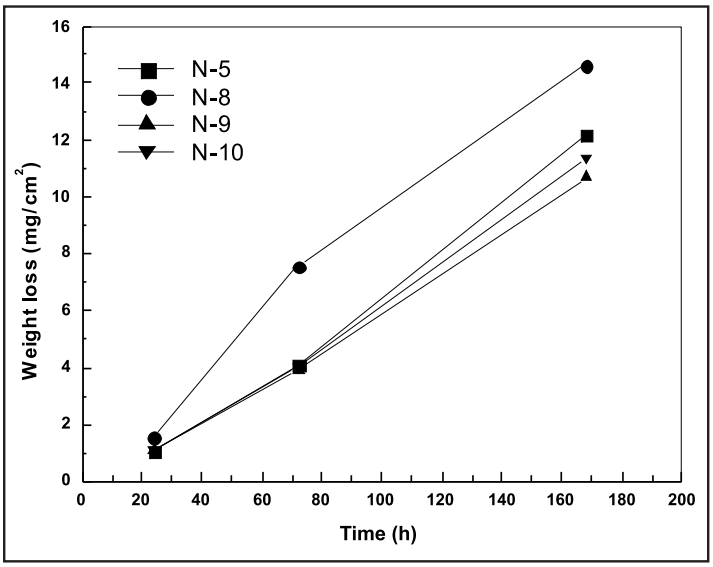


Figure 2. Weight loss of cast Ni-base alloys corroded at 650°C as a function of time.

for YSZ produced a coating with interconnected three-dimensional pores covering 3.2 percent of the area and splat boundaries. The small-sized powder coating porosity was reduced to one percent of the area and splat boundaries were non-distinct. Although both feedstock powders produced a highly dense coating for Al_2O_3-Y_2O_3, the conventional-sized coating displayed approximately 15 percent partially melted splats, a defect that significantly lowered the microhardness.

This study developed four functional gradient YSZ coating systems and three Al_2O_3-Y_2O_3 coating systems. The thickness of YSZ top coating was nearly the same in each coating system, that is, 230 µm to 250 µm. Commercially available gas-atomized CoNiCrAlY powder with an average

particle size of approximately 40 µm was used as a bond coat material, and YSZ powder was used as the ceramic top-coat material. The intermediate layers between the bond coat and the top coat were produced using a mechanically blended powder from MCrAlY and YSZ.

The results of interfacial bond strength of each coating system are shown in Figure 3. Overall, the rougher the substrate surface, the higher the interfacial bond strength. Substrate surface roughness of Ra = 2.5 µm had similar interfacial bond strength and all failed at the substrate-bond coat interface. Surfaces of Ra = 5.0 µm had increased interfacial bond strength between the substrate and bond coat and separation occurred at other interfaces. The duplex coating failed at the interface between the bond coat and the YSZ, while three- and five-layer gradient coatings failed in the intermediate layer. The functional gradient coatings had a higher interfacial strength up to 16 MPa. The five-layer coating containing the MCrAlY/Al_2O_3-Y_2O_3 intermediate layer had the highest strength.

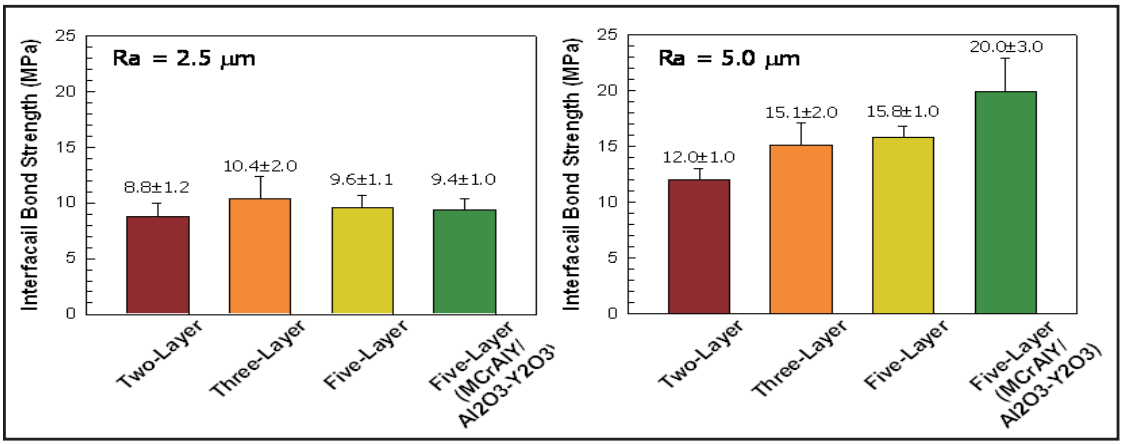


Figure 3. Interfacial bond strength of functional gradient YSZ coatings.

Researchers also designed three functional gradient amorphous Al_2O_3-Y_2O_3 coating systems. Similar to YSZ, the five-layer system had a fine distribution of both metallic and ceramic splats. The amounts of pores in the intermediate layers and the top Al_2O_3-Y_2O_3 coatings were much lower than those in the YSZ coating system. Bond strength increased gradually with the number of intermediate layers, with the highest bond strength up to 22±2 MPa for the five-layer coating.

Corrosion Testing of FGM Coatings. The coated sample coupons described above were submersed up to approximately 75 percent of vertical length in LiCl-3 wt% Li_2O molten salt at 650°C. Slow evaporation of the molten salt took place over 186 hours, leaving only 20 percent of the coupon vertical length submersed. The coupons were removed, allowed to cool in an Ar atmosphere for 24 hours, and ultrasonically washed in water. The coating was intact and adherent in all tested samples until cooling to room temperature, when spalling/delamination occurred during ultrasonic washing. Thermal shock may have generated cracks.

SEM of the two-layer coating showed separation starting at the interface of the ceramic top coat and the underlying MCrAlY. The five-layer gradient coating remained intact, but transformation from amorphous Al_2O_3-Y_2O_3 into crystallization yttrium aluminum garnet ($Y_3Al_5O_{12}$) began to occur along the vertical axis of the coating. This transformation usually occurs above 950°C; however, there was evidence of increased concentrations of Al in areas of apparent crystallization and differing contrast at 650°C.

Cross-sectional images confirmed that the delamination occurred just below ceramic coating. XRD showed no reaction products formed on the delaminated MCrAlY surface, indicating that the ceramic layer protected the underneath MCrAlY and Ni alloy in the molten salt and the interface delamination of ceramic layer occurred after removal from the molten salt.

The five-layer YSZ coatings and the three- and five-layer Al_2O_3-Y_2O_3 coatings remained intact following the high-temperature molten salt exposure and cooling. Surface damage occurred owing to an aggressive corrosion/erosion environment and was limited to the top surface layer.

Planned Activities

Next year's work will include fabricating component parts from the new alloys and additional alloying with rare-earth elements, along with further studies on the effects of silicon in optimizing a protective oxide scale growth. Pre-spray surface conditions, post-spray heat treatments, more diverse combinations of coating gradient layers, improved coating robustness and resistance to erosion during the corrosion process, and the crystallization behavior of amorphous Al_2O_3-Y_2O_3 at 650°C for a longer term will all be investigated in detail.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

VHTR Environmental and Irradiation Effects on High-Temperature Materials

PI (U.S.): William R. Corwin, Oak Ridge National Laboratory (ORNL)

PI (ROK): Woo-Seog Ryu, Korea Atomic Energy Research Institute

Collaborators: Idaho National Laboratory (INL), Korea Advanced Institute of Science and Technology (KAIST)

Project Number: 2006-006-K

Program Area: Generation IV

Project Start Date: October 2006

Project End Date: September 2009

Research Objectives

This project will select and qualify high-temperature materials for the next generation of reactors. The collaboration will include work to analyze irradiation and environmental effects and to develop a materials handbook containing the properties of high-temperature metallic materials.

Specifically, researchers will study the effects of helium environments on the mechanical properties of high-temperature metallic alloys proposed for use in the very high-temperature reactor (VHTR). The helium primary coolant in an operating VHTR is expected to be contaminated by small amounts of gaseous impurities from a variety of sources. Corrosion of structural alloys by these impurities at elevated temperatures can be significant. Researchers will also evaluate the effects of irradiation on these alloys using the Republic of Korea's High-Flux Advanced Neutron Application Reactor (HANARO) and the Irradiated Materials Evaluation Facility (IMEF). Reactor operating conditions are very challenging for these materials and they require qualification against the effects of irradiation. Materials testing of key components such as the reactor pressure vessel will provide the necessary design data to develop the *Generation IV Materials Handbook*, which will establish a comprehensive database of high-temperature materials.

The project consists of the following three tasks:

- Environmental effects on high-temperature materials
- Irradiation effects on high-temperature materials
- High-temperature materials handbook development

Research Progress

Environmental Effects on High-Temperature Materials. During FY 2007, researchers performed a series of experiments in a helium loop with controlled impurity chemistries to characterize the effect of gas chemistry on Inconel 617 and Haynes 230, two potential heat exchanger materials. They evaluated the influence of oxygen partial pressure and carbon activity on the microstructure of both materials and evaluated the mechanical properties of Haynes 230. They also studied evolution mechanisms of the oxide layer and carburization/decarburization and installed the recirculating helium loop at KAIST.

Testing was carried out with fixed gas chemistry for 500 hours or 1,000 hours at temperatures from 800°C to 1,000°C. One test at 1,000°C was designed to result in oxidation and decarburization of the alloys, while the remaining test conditions were selected to result in carburization (only Haynes 230 tested) with little or no formation of oxide scale on the alloys. Post-exposure analyses included optical and scanning electron microscopy and x-ray diffraction.

At 1,000°C, the materials under oxidizing condition displayed continuous black scales, while those under carburizing conditions showed a metallic appearance with little or no evidence of a surface scale or carbon deposition. Inconel 617 showed a relatively thick chromium oxide scale

with significant formation of grain boundary aluminum oxides, whereas Haynes 230 showed less surface oxidation and a reduced tendency for formation of grain boundary oxides. Both alloys exhibited a decarburized region, which was deeper for Haynes 230.

Tensile tests on extensively carburized Haynes 230 specimens at exposures of 900°C and 1,000°C had zero ductility at room temperature and exhibited a reduction in area at lower temperatures. The fracture strength decreased from about 700 MPa at room temperature to 500 MPa at a test temperature of 800°C for both carburization temperatures. The results of these experiments reinforce the need for maintaining control of the helium gas chemistry to avoid harmful effects on heat exchanger materials.

Researchers also studied the oxidation and evolution mechanisms of Alloy 617 and Haynes 230 oxide layers at 900°C and 1,100°C in air to establish any differences from a pure helium environment. Environmental degradation of Alloy 617 was studied under constant tensile stress and researchers characterized changes in mechanical properties.

Irradiation Effects on High-Temperature Materials. The team successfully irradiated the first high-temperature material capsule (Mod.9Cr-1Mo) in HANARO at 30 MWth and a fast neutron fluence up to 4.4×10^{19}(n/cm^2) (E>1.0MeV). The dpa of the irradiated specimens was evaluated to be approximately 0.034 to 0.07. Based on these test results, researchers designed and fabricated a second capsule for an irradiation of 9Cr-1Mo-1W specimens. Tensile and impact tests of the irradiated Mod.9Cr-1Mo (G91) were done in the hot cell of the IMEF. Hardening of yield and tensile strength by irradiation in the G91 steel was higher at the heat-affected zone. Reduction of elongation was higher at the weldment. The degradation of impact properties by irradiation was severe at the weldment compared with the base metal.

The team procured a second forged heavy 9Cr-1Mo-1W (G911) steel sample and performed narrow gap welding according to the modified procedure for COST E steel. They performed tensile and Charpy impact tests with the G911 base metal and weldment before irradiation. The base metal of the G911 steel showed higher yield and tensile strength, as well as lower total elongation, than the G911 weldment. The base metal also exhibited more improved impact properties than the weldment.

High-Temperature Materials Handbook Development. Researchers upgraded the *Generation IV Materials Handbook*, a web-accessible database, in preparation for its use as a repository for international data associated with the VHTR Materials Project Arrangement. To comply with user access protocols allowing for global data collaboration, a new access control system was developed that is independent of the ORNL system. A new demonstration version of the database was developed and released with a companion user guide to member countries of the Generation IV International Forum for their evaluation and as an initial step in preparation for receiving international data contributions. Significant improvements to the handbook were also made in the areas of structure development, content expansion, international collaboration, and preparation for massive data population.

Researchers have conducted a comprehensive literature survey of several Generation IV nuclear reactor candidate materials to identify and extract applicable data and relevant information to be included in the handbook. They also incorporated some of the results for 316H SS, Hastelloy X, Haynes 230, Incoloy 800H, and Inconel 617 into Part A-Materials to replace the tentative placeholder contents of the beta version. This included general material descriptions; physical, thermal, and environmental resistance properties; chemical compositions; material biographies (e.g., development history, chemistry features, strengthening mechanism, alloy type, crystal structure, base phase, secondary phases/functions, metallurgical stability, treatment, weldability, applications); product forms; and applicable specifications, codes, and standards (ASTM, ASME, international).

Planned Activities

Subsequent work will 1) evaluate environmental degradation mechanisms of Alloy 617 as well as other candidate superalloys for VHTRs under simulated impure helium environments; 2) perform detailed design of a second irradiation capsule and manufacturing it, a second irradiation test, and post-irradiation testing using the second procured lot of Cr-Mo steel; and 3) continue to design and construct database structure for remaining parts/chapters, enhance handbook data management functionalities, incorporate additional materials property data into the handbook, and initiate use for international VHTR PMB and ROK I-NERI *Generation IV Materials Handbook* users. Researchers also plan on continuing to establish the datasheets using tensile and impact testing results of the irradiated 9Cr-1Mo-1W steel.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Experimental Validation of Stratified Flow Phenomena, Graphite Oxidation, and Mitigation Strategies of Air Ingress Accidents

PI (U.S.): Chang H. Oh, Idaho National Laboratory

PI (ROK): Hee Cheon No, Korea Advanced Institute of Science and Technology (KAIST)

Collaborators: None

Project Number: 2007-001-K

Program Area: Generation IV

Project Start Date: October 2007

Project End Date: November 2010

Research Objectives

This project is developing and validating advanced models for air ingress accidents in very high-temperature gas reactors (VHTRs). Research is focused on 1) analytical and experimental study of air ingress caused by density-driven, stratified, countercurrent flow; 2) advanced graphite oxidation experiments; 3) experimental study of burn-off in the bottom reflector; 4) structural testing of burn-off in the bottom reflector; 5) implementation of advanced models developed during the previous tasks into the gas multicomponent mixture analysis (GAMMA) code; 6) full air ingress and oxidation mitigation analyses; 7) development of core neutronic models; 8) coupling of the core neutronic and thermal hydraulic models; and 9) verification and validation (V&V) of the coupled models.

The development of VHTR advanced air ingress models is a very high priority. Following a loss of coolant and system depressurization, air will enter the core through the break, leading to oxidation of the in-core graphite structure and fuel. Oxidation will accelerate heat-up of the bottom reflector and the reactor core and eventually cause the release of fission products. The potential collapse of the bottom reflector because of burn-off and the release of carbon monoxide (CO) gas can lead to serious safety problems. Estimating the proper safety margin requires experimental data and tools, including accurate multidimensional thermal-hydraulic and reactor physics models, a burn-off model, and a fracture model. It will also require effective strategies to mitigate the effects of oxidation. The results of this research will provide these crucial inputs.

Research Progress

Density Difference-Induced Stratified Flow Analysis. Researchers performed a preliminary computational fluid dynamics (CFD) analysis to understand density-driven stratified flow in the VHTR air ingress accident. Various parameters were taken into consideration, including the turbulence model, core temperature, initial air mole-fraction, and flow resistance in the core. The 600-MWt gas turbine modular helium reactor (GT-MHR), with a simplified 2-D geometry, was selected as the reference reactor. Following the preliminary CFD results, the researchers performed an analysis of the air ingress accident by two different codes: GAMMA (system analysis code) and FLUENT (CFD code). The analysis showed that the onset of natural convection is approximately five minutes, significantly earlier than the 150-hour result of previous calculations based on the molecular diffusion air ingress mechanism. This leads to the conclusion that the consequences of this accident will be much more serious than previously expected.

Experimental Study on the Stratified Flow. Researchers established an experimental plan for density-driven stratified flow to validate the computer codes for air ingress analysis. They identified important phenomena to be validated using a design matrix, and established the basic concepts of the experiments. The experiments were divided into two parts: isothermal and non-isothermal tests. The isothermal tests focused on the separate effect of the stratified flow. In the isothermal experiments, the facility dimensions and sizes were determined by a simple

scaling analysis in order to generate the similar flow to the reference VHTR. The non-isothermal tests focused on the coupling effects with the stratified flow including heat transfer, porous media, and chemical reactions. In the non-isothermal experiments, the onset of natural convection was selected to be the main measuring parameter. Preliminary CFD analysis showed that the onset natural circulation can be measured by the detection of mass flow and temperature changes. All the experiments were designed to be transparent for instrumentation and detection by optical devices.

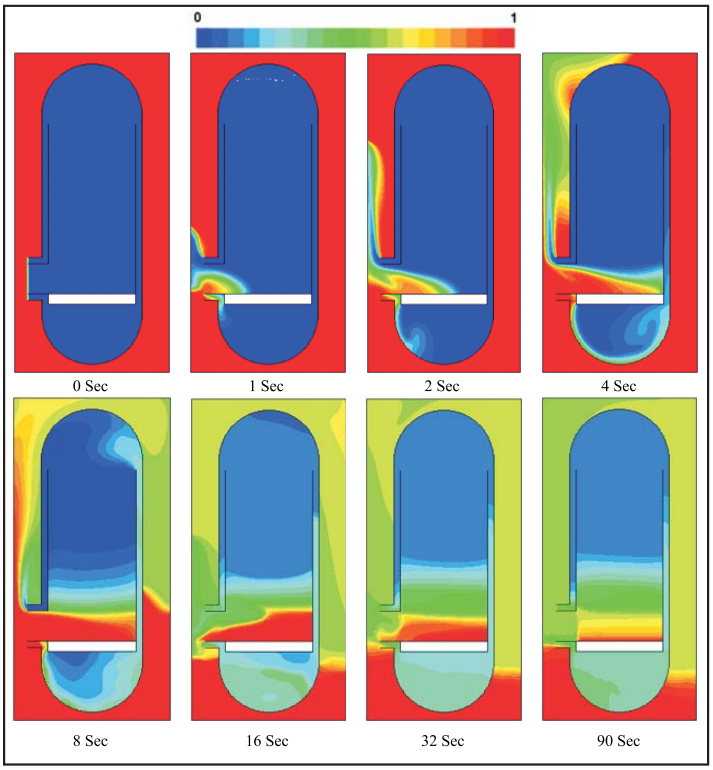


Figure 1. Variations of air distributions in the reactor core during stratified flow.

Advanced Graphite Oxidation Study. Various characteristics on the graphite oxidation have been studied for detail analysis of an air ingress accident. The characteristics being considered included the effect of oxidation degree on the graphite strength, the effect of oxidized graphite density on the oxidation rate, and surface area density in the graphite internal pores. During this task, researchers developed an experimental methodology to validate the previous correlations related to oxidized graphite strength, which is essential for analysis of graphite structure collapse. Following the graphite experiment, the collapse of the graphite structure was estimated for the reference VHTR by two computer codes: GAMMA and ABAQUS (stress analysis code). The GAMMA code predicted graphite oxidation and corrosion, which was implemented into ABAQUS to estimate core collapse. The predicted collapse time of the graphite structure is about five days after natural convection starts.

Experiment of Burn-Off in the Bottom Reflector. Researchers conducted experimental investigations of burn-off in the bottom reflector graphite of the VHTR. There are several candidate graphite materials for the bottom reflector; IG-110 and IG-430 were selected for testing. Kinetics, mass diffusion, and their combined effect were investigated in order to estimate the order of reaction (n), activation energy (E_a), and heat/mass transfer rate coefficient. The researchers also measured reaction rates and CO/CO_2 in a temperature range of 700°C to 1,500°C and an oxygen/helium condition. IG-110 data from previous investigations was compared.

Structural Tests of Burn-Off in the Bottom Reflector. Mechanical testing of bottom reflector graphite included an investigation of compressive strength degradation due to burn-off of graphite, and determination of the buckling limit and strength degradation of the oxidized graphite support column in temperatures ranging from 600°C to 1,050°C. As a result, researchers determined 79.46 MPa as the compressive strength of IG-110. They also estimated the correlation between slenderness and buckling ratio and the strength of various oxidized graphite samples. Graphite columns of the same slenderness ratio have the same strength but different diameters. The slenderness ratio can thus be the scaling parameter for a simple cylindrical graphite column in strength. The team performed GAMMA code calculations to predict the trend and condition of oxidation in the bottom reflector. The ratio of critical strength of uniformly oxidized graphite to that of fresh graphite can be expressed by the Knudsen relation. It turns out that the ratio is independent of geometry, while the strength degradation of an oxidized graphite structure is dependent on the initial strength of the structure and the bulk density change. The volume measurement technique was studied for analysis of graphite oxidized in the in-pore diffusion regime. This method will be used in the analysis of graphite that is not uniformly oxidized.

Coupling Neutronics/Thermal-Hydraulics Tools. Team members proposed coupled neutronics/thermal-hydraulics tools to analyze the VHTR. They developed both a multidimensional GAMMA code and a multi-group 3-D hexagonal geometry neutronics (COREDAX) code. A strategy was developed to couple the two codes into a system code, GAMMA/COREDAX, by determining coupling factors, mapping nodes between the two codes, and organizing calculation logic.

Core Neutronics Model. The COREDAX code is based on the analytic function expansion nodal (AFEN) method in 3-D hexagonal geometry. The AFEN method includes the use of node-interface flux moments as nodal unknowns to increase the accuracy of calculation results. In this project, multi-group extension based on matrix function theory and coarse group rebalance acceleration were incorporated into the COREDAX code. Researchers verified the COREDAX code implementing the AFEN method by testing the VVER-440 benchmark problem, a simplified VVER-1000 benchmark problem, and the SNR-300 benchmark problem.

In preparation for V&V, the neutronics of a simple GT-MHR core was calculated. This core is geometrically based on the GT-MHR benchmark problem, but the cross sections are informally obtained.

Planned Activities

Experiment of Burn-Off in the Bottom Reflector. The burn-off effect, moisture effect, and Boudouard reaction will be investigated next year. In FY 2009, IG-430 data will be obtained and compared, and the other graphite material will be investigated.

Structural Tests of Burn-Off in the Bottom Reflector. A fracture model of oxidized graphite structure will be proposed for FY 2009.

Coupled Core Model V&V. This task involves the verification of the GAMMA code coupled with the COREDAX code and the validation of prediction results of thermal power distribution in the hexagonal reactor core. The coupling task will be performed in FY 2009 at KAIST.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of an Advanced Voloxidation Process for Treatment of Spent Fuel

PI (U.S.): Brian Westphal, Idaho National Laboratory (INL)

PI (ROK): Jang Jin Park, Korea Atomic Energy Research Institute (KAERI)

Collaborators: None

Project Number: 2007-002-K

Program Area: AFCI

Start Date: October 2007

End Date: September 2010

Research Objective

This research project addresses the head-end treatment of spent nuclear fuel undergoing pyroprocessing. The objective is to develop an advanced voloxidation process that 1) provides a means to recover fuel from the cladding, 2) prepares fuel for subsequent processing, 3) simplifies downstream processing by removing volatile fission products prior to the pyroprocess, and 4) safely traps the volatile fission products. The basis of the voloxidation process is to oxidize spent oxide fuel at low temperatures (about 500°C) in order to remove tritium and increase the dissolution rate during aqueous processing. Advanced voloxidation not only performs these functions but also removes and collects specific fission products such as krypton, xenon, cesium, rubidium, iodine, technetium, molybdenum, and ruthenium.

The proposed research focuses on three primary areas: 1) evaluation of the effects of advanced voloxidation on pyroprocessing of spent oxide fuel, 2) optimization of the off-gas trapping capabilities for fission products with respect to pyroprocessing, and 3) development of operational parameters for advanced voloxidation with respect to pyroprocessing. Advanced voloxidation will also address fuel types other than light water reactor (LWR) fuels, such as fast reactor oxide fuels.

Research Progress

Assuming an advanced voloxidation process is utilized as a precursor to pyroprocessing, researchers investigated the reduced effects of radioactivity and decay heat on the spent oxide fuel. They realized a reduction in radioactivity and decay heat of 43 percent to 57 percent, respectively, for typical pressurized water reactor fuel after cooling for five years. A shielding analysis of the spent fuel following advanced voloxidation resulted in a 28 percent reduction in shielding material. In addition, an evaluation was performed for four possible head-end scenarios to pyroprocessing with respect to their downstream effects on pyroprocessing. Although the overall differences in the four scenarios were minimal, the scenario involving advanced voloxidation was given the highest grade.

The research team continued testing to optimize the trapping efficiency of volatile off-gases. Using non-radioactive materials, the trapping efficiency of the fly-ash filter for volatile cesium was increased by a factor of 10 by increasing the zone temperature to 1,100°C. For radioactive materials, the trapping temperatures and pressurization cycle were modified to maximize the collection of volatiles with the off-gas trapping system during mixed oxide (MOX) testing. Researchers performed three experiments, represented in Figure 1, with high burnup MOX spent fuel in the INL Hot Fuel Examination Facility for the purpose of investigating the oxidation and decladding characteristics while removing and collecting volatile fission products. The team achieved successful oxidation and decladding of

Figure 1. Photograph of MOX fuel following advanced voloxidation testing.

irradiated MOX fuel and determined the necessary parameters to accomplish the task. Sufficient oxidation and decladding of the D9-clad MOX fuel occurs with a minimum temperature of 600°C for three hours under an oxygen atmosphere.

Considering the relationship of advanced voloxidation to pyroprocessing, the ability to control the particle size of the voloxidation product is important for containment of the product during both oxide reduction and transfer operations. Experiments to increase particle size were carried out with a rotary-type reactor and a bismuth-oxide (Bi_2O_3) powder. The Bi_2O_3 powder was chosen based on thermal limitations of the rotary reactor equipment. Particle size was adjusted by controlling temperature and rotational speed of the reactor. Additional experiments were performed with U_3O_8 powder in a static environment to determine the degree of agglomeration above 1,000°C under oxidizing and reducing (vacuum) conditions. Significant densification was achieved at 1,200°C for 12 hours under an air atmosphere.

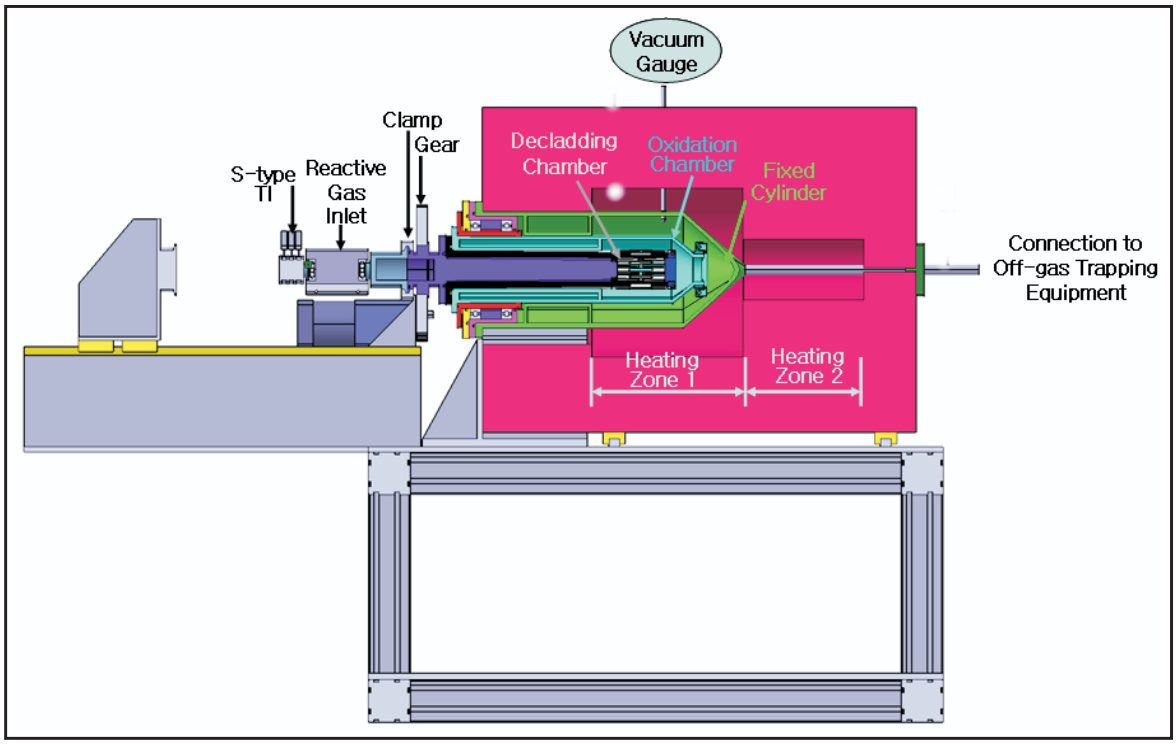


Figure 2. Schematic of advanced rotary-type voloxidizer designed by KAERI.

Based on the initial powder agglomeration testing, researchers have designed a rotary-type voloxidizer capable of oxidative decladding, off-gas trapping, and particle size control (Figure 2). The unit is designed to voloxidize up to 250 g of UO_2 at 500°C under oxidizing conditions, followed by a temperature increase to 1,200°C under rotary vacuum conditions for the removal of volatiles and control of particles.

Planned Activities

Experimentation with the advanced voloxidizer will begin using non-radioactive materials in the near term. Experiments with radioactive materials will be performed with LWR spent oxide fuel to determine the effect of the oxidation temperature on particle size distribution.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Performance Evaluation of TRU-Bearing Metal Fuel for Sodium Fast Reactors to Achieve High Burnup Goal

PI (U.S.): J. Rory Kennedy, Idaho National Laboratory (INL)

PI (ROK): Byoung-Oon Lee, Korea Atomic Energy Research Institute (KAERI)

Collaborators: None

Project Number: 2007-003-K

Program Area: AFCI

Project Start Date: November 2007

Project End Date: November 2010

Research Objectives

This project will evaluate the high burnup performance of transuranic (TRU)-bearing metal fuel for the sodium fast reactor (SFR). The scope involves 1) benchmarking of a performance analysis code using AFC-1 test data, 2) fabrication of barrier-cladding material, and 3) out-of-pile performance testing against metal fuel.

Benchmarking of Fuel Performance Code. The objective of this task is to develop suitable models and to benchmark the fuel performance code against irradiation data from AFC-1 tests. Fuel performance codes are essential tools in the analysis of fuel behavior and integral fuel performance. These codes make it possible to analyze the complicated irradiation behavior of the fuel. Development and verification of a computer code specific to TRU fuel will allow for the comprehensive analysis of metallic fuel irradiation behavior.

As TRU content in the AFC-1 tests range up to 60 wt%, existing fuel performance codes are not suitable. Development of models and benchmarking of the fuel performance code against irradiation data are needed. This task involves three subtasks: 1) preliminary performance analysis of metal fuel, 2) benchmarking of fuel performance codes by AFC test data, and 3) upgrading of models based on the benchmarking results.

Fabrication of Barrier Cladding. This second proposed task focuses on the fabrication of barrier-cladding materials and tubes. One of the potential factors limiting alloy fuel burnup is fuel-cladding chemical interaction (FCCI). To resolve this issue, one of the methods considered is the use of a barrier on the cladding inner surface. This task involves two subtasks: 1) fabrication of barrier-cladding material and 2) fabrication of barrier-cladding tubes.

Diffusion Testing of Metal Fuel Against Barrier Cladding. The objective of this task is to conduct the diffusion couple testing of metal fuel against barrier cladding. Rare-earth (RE) element attack of the cladding is a potential life-limiting factor for metallic fuels and eutectic melting is a potential high-burnup issue for SFR fuels. Use of barrier-cladding material may increase fuel burnup performance.

Screening studies will establish the diffusion behavior of RE alloys in contact with proposed barrier materials (e.g., V, Cr, Cr_2O_3). Researchers have proposed using mischmetal, Ce, Nd, and/or Ce-Nd as the RE alloy and will conduct diffusion couple tests of RE-doped U-TRU-Zr fuel against the barrier cladding. This task consists of four subtasks: 1) U-TRU-Zr diffusion tests against barrier-cladding materials, 2) screening studies of the diffusion behavior of RE in contact with proposed barrier materials, 3) diffusion tests of RE-doped metal fuel against barrier cladding tubes, and 4) joint evaluation of the performance of barrier cladding.

Research Progress

Benchmarking of Fuel Performance Code. Researchers jointly analyzed results of the AFC-1 tests, including fuel description and operational history. They determined input data and operating conditions, such as the fuel pin specifications, power history, thermal-hydraulic conditions, and maximum steady state cladding temperature. The MACSIS code was used to evaluate preliminary metal fuel performance and preliminary parametric results were analyzed.

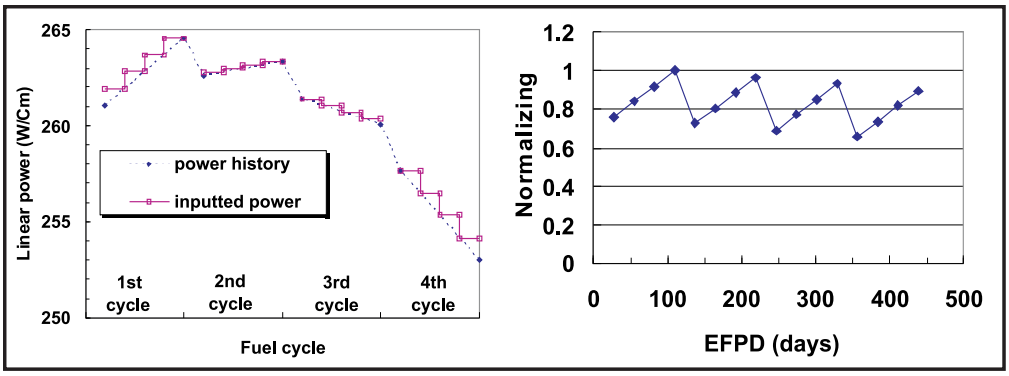


Figure 1. Linear power for the MACSIS input.

Figure 1 shows the linear power of the AFC-1 test for MACSIS. The pattern reflects the power due to Cd depletion and was inputted into MACSIS. This figure also shows the linear power rate for KALIMER-600, which is a typical pattern for the SFR.

Researchers performed detailed parametric analyses for the design parameters and operating conditions of the AFC-1 test. They analyzed important performance parameters such as temperature, fission gas release, fuel swelling, creep strain, cumulative damage fraction and Zr migration according to the variation of inlet coolant temperature, pin power, plenum-to-fuel volume ratio, smeared density, cladding thickness, Zr content, Pu fraction, and cladding material.

Fabrication of Barrier Cladding. Researchers fabricated barrier-cladding materials and provided eight samples of Cr-coated HT-9 and Cr_2O_3-coated HT-9 disks for interdiffusion tests. In order to overcome the FCCI problem and eutectic melting of metallic fuel, candidate diffusion barrier materials such as Zr, V, Cr, etc., were preliminarily tested by diffusion couple annealing tests with commercial metallic foils. Comparing the interactions between U-Zr-X and various diffusion barrier materials, researchers concluded that an element with limited uranium solubility and no intermediate phases with uranium should be used as a diffusion barrier material; V and Cr are considered strong candidates.

Diffusion Testing of Metal Fuel Against Barrier Cladding. The initial fuel sample composition tested was 50U-30Pu-20Zr. Since ferritic martensitic steels have excellent irradiation performance, researchers used disks of HT9 ferritic steels as cladding materials in a diffusion couple. They conducted the diffusion couple tests of the U-TRU-Zr fuel against the barrier-cladding materials at 700°C for 100 hours. Microstructures of diffusion couple specimens will be observed by scanning electron microscopy (SEM), and the elemental composition of any interaction layers will be measured by energy dispersive x-ray spectroscopy (EDS) and wavelength dispersive spectroscopy.

Figure 2 shows microstructures of the coated disks. SEM observation revealed several micro-cracks inside the layer and the layer-cladding interface of the Cr-coated sample. EDS analysis showed Cr all along the layer in the Cr-coated specimen. The presence of oxygen along the layer in the Cr_2O_3-coated specimen indicates that the major constituent is indeed the Cr oxide.

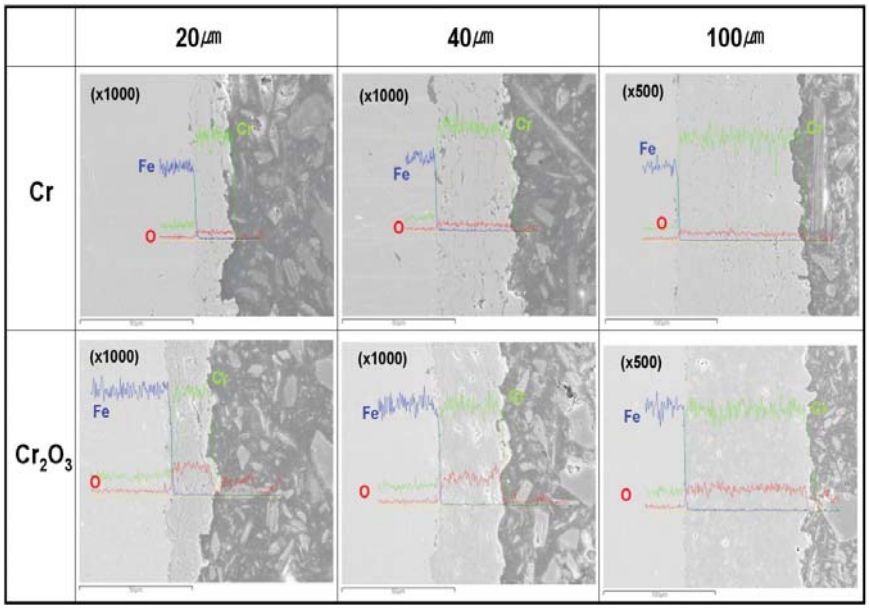


Figure 2. Microstructures of Cr and Cr_2O_3 plasma spray coating.

Planned Activities

Benchmarking of Fuel Performance Code. Follow-on work will be to review the experimental data available and benchmark the fuel performance code using the available AFC-1 test data.

Fabrication of Barrier Cladding. The fabrication process used to line or coat the cladding tubes will be investigated and evaluated. Researchers at KAERI will fabricate the barrier-cladding tubes and provide the tubes to INL researchers for interdiffusion tests.

Diffusion Testing of Metal Fuel Against Barrier Cladding. The analysis of the initial U-TRU-Zr diffusion couple tests against the barrier-cladding materials will be completed by the end of 2008. KAERI researchers may conduct the screening studies of the diffusion behavior of RE in contact with proposed barrier materials (Cr, V, Cr_2O_3, etc.). INL will conduct the diffusion couple tests of RE-doped U-TRU-Zr fuel against the barrier-cladding tubes.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development and Characterization of New High-Level Waste Form for Achieving Waste Minimization from Pyroprocessing

PI (U.S.): Terry Battisti, Idaho National Laboratory (INL)

PI (ROK): Yung-Zun Cho, Korea Atomic Energy Research Institute

Collaborators: None

Project Number: 2007-004-K

Program Area: AFCI

Project Start Date: October 2008

Project End Date: September 2011

Research Objectives

The objective of this project is to develop new high-level waste forms and fabrication processes for the disposal of active metal fission products resulting from the pyroprocessing-based fuel cycle. These fission products are chemically separated from electrorefiner (ER) salts during pyroprocessing. The current technology for disposing of them involves non-selectively discarding the fission product-loaded salt in a glass-bonded sodalite ceramic waste form. Selective removal of fission products from the molten salt would reduce the amount of high-level radioactive waste generated. Novel methods, including chemical precipitation, are currently being developed to achieve this selective separation of fission products, but no investigation has yet been performed regarding suitable waste forms for the separated fission products. The precipitates are expected to be oxides, sulfates, and/or phosphates and would be incompatible with the sodalite ceramic waste form. Thus, a completely novel approach to waste form synthesis is required.

Research Progress

Depending on the technology for removing fission products from the salt, three kinds of wastes for which the immobilization method should be developed can be expected to be generated—eutectic salt waste, rare-earth (RE) precipitates, and a mixture of the two. Researchers investigated different approaches for each waste stream to design suitable waste forms and to find a proper fabrication method to fashion monolithic forms. For the case of eutectic salt waste, the dechlorination of metal chlorides was considered as the target reaction to overcome Cl-induced disadvantages on the immobilization. In this study, the SiO_2-Al_2O_3-P_2O_5 (SAP) composite was used as a chemical agent for dechlorination. X-ray diffraction results from preliminary tests indicated that KCl gradually disappeared from the reaction with SAP to produce an amorphous phase.

It is desirable to immobilize the RE oxide waste stream in robust natural minerals. Several candidates for a host matrix have been considered, but testing was focused on apatite and monazite. In order to obtain a monolithic waste form, the fission product-loaded apatite/monazite would need to be mixed with glass and vitrified. Several different glass compositions were tested as immobilizing agents. Modified glasses with high CaO and Al_2O_3 content were found to be optimal for this purpose. A two-step waste form process was adopted that included first low-temperature chemical conversion to the crystalline product, followed by high-temperature consolidation using a reliable glass.

For the mixed salt/RE oxide waste, SAP was found to react with both the salt and the RE oxides. In the presence of metal chlorides, the RE oxides react with SAP to form stable monazite at temperatures as low as 650°C. Given these considerations, the immobilization procedure for the expected wastes can be summarized as shown in Figure 1. Mixed waste can be treated by the same procedure as the salt waste.

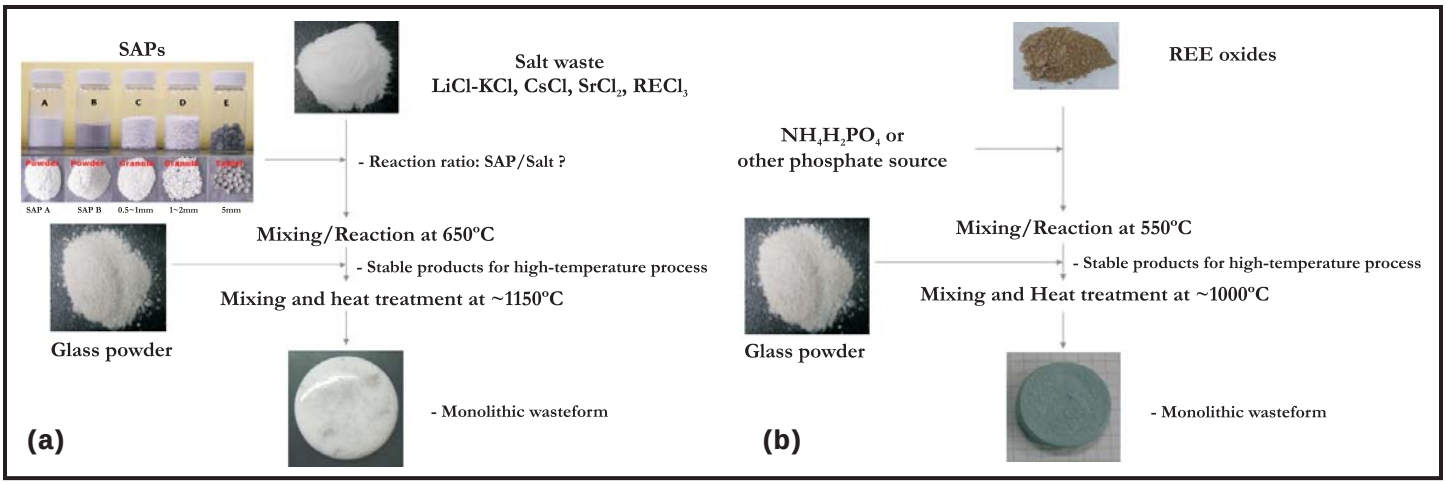


Figure 1. Immobilization procedure for the expected wastes: (a) salt waste and (b) RE oxide waste.

As U.S. funding was unavailable for this project in FY 2008, both parties agreed that researchers at INL would initiate work in FY 2009 and continue for a total of three years.

Planned Activities

In order to verify and optimize the above procedure, researchers will study several technical issues in FY 2009. They will optimize SAP composition to increase reactivity with the eutectic salt and assess vaporization during the process. The glass composition may be changed to improve compatibility with certain reaction products. Leach resistance of the waste forms will be studied using the standard PCT-A test method.

The developed procedures will be verified by performing surrogate tests. Both monazite and apatite samples will be made and vitrified. SAP will be mixed with surrogate salt and then with glass to make SAP-based waste forms. Efforts will also be carried out to prepare in-cell systems such as the Hot Fuel Dissolution Apparatus (HFDA) and the Declad and Oxidize (DEOX) furnace for supporting fabrication of waste forms containing real fission products from ER salts. The HFDA will generate the salt/RE oxide wastes to be used in these tests, and the DEOX furnace (after modification) will form the crystalline species and perform the final vitrification step.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Technology for Viable International Deployment of Small Sodium-Cooled Fast Reactors

PI (U.S.): James J. Sienicki, Argonne National Laboratory

PI (ROK): Kune Y. Suh, Seoul National University

Collaborators: None

Project Number: 2007-005-K

Program Area: Generation IV

Start Date: November 2007

End Date: November 2010

Research Objectives

Small, modular, transportable sodium-cooled fast reactors (SFRs) can play a major role in helping to meet the significant projected growth in worldwide nuclear capacity while reducing the risk of proliferation. Particular applications for which such reactors are well-suited include powering industrial enterprises such as mining in remote regions and providing electricity to consumers in developing nations that do not have major grid connections. The objectives of this project are to 1) identify approaches that minimize the construction time, labor, and costs for small, modular, transportable SFRs and 2) develop component modularization techniques, transportation, and site installation sequences that reduce onsite assembly time and labor.

Research Progress

During the first year of the project, researchers decided to initially use the advanced burner test reactor (ABTR) as the reference preconceptual design for a small modular SFR. The ABTR is a 250-MWt metallic-fueled, pool-type reactor with core outlet and inlet temperatures of 510°C and 355°C, respectively, and a 60-year service lifetime. The ABTR incorporates a supercritical carbon dioxide (S-CO_2) Brayton power conversion cycle to 1) eliminate sodium-water reactions, 2) provide more economical power conversion, and 3) enhance plant efficiency.

The team developed a 3-D computer-aided design model for the ABTR preconceptual design based on 2-D drawings. The ABTR's primary coolant system, containment building, and containment polar crane have been modeled and assembled utilizing the commercial tool, CATIA. The energy conversion system model has yet to be expanded to include piping details. Researchers built a test apparatus (Figure 1) to measure fundamental thermo-hydrodynamic characteristics of flowing S-CO_2 near and exceeding the critical and pseudo-critical points. The apparatus is mainly designed to measure the heat transfer coefficient for upward flow in a circular pipe at 8 MPa and 120°C. Construction has been completed and leakage testing is in progress. Figure 2 shows a drawing of the circular channel surrounded by six heaters providing a heat flux through the channel wall. Various test section designs having different flow channel geometries are being considered to enhance applicability of the test results to

Figure 1. Test apparatus for measurement of heat transfer coefficients for supercritical CO_2.

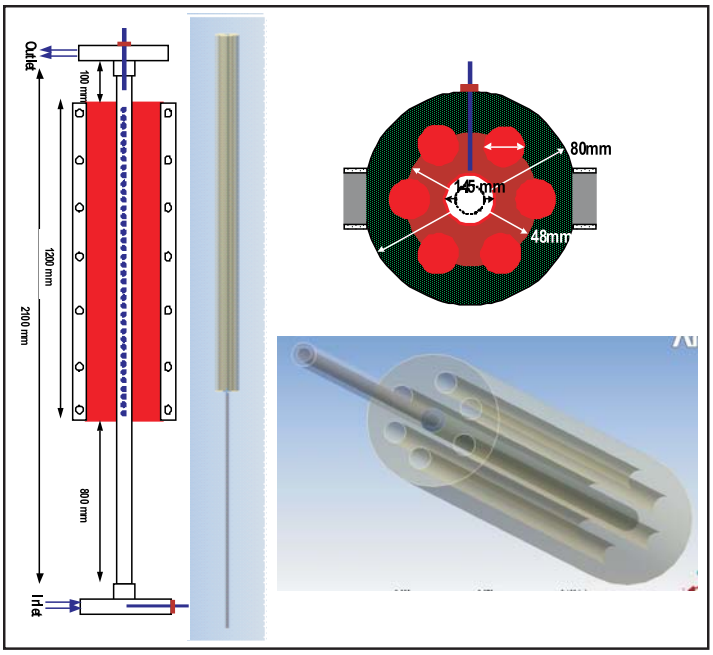


Figure 2. Circular channel test section for measurement of heat transfer coefficients for supercritical CO_2.

S-CO_2 Brayton cycle power converters. Automatic control of the S-CO_2 cycle involves the opening and closing of valves such as the turbine bypass valve, valves to and from the inventory control tank, and cooler bypass valves. Scaled valve operation tests were performed to measure the flow coefficients with air. Based on these test results, researchers developed an operational algorithm, which they implemented in a computer code for a hypothetical configuration of two stop valves and four control valves to predict the gas mass flow rate. The measured controlled mass flow rate is not proportional to the valve opening, as it turns out. It is thus non-trivial to control the mass flow rate in practice. Linearization of the valve curve is typically useful in facilitating the control valve operation. The valve curve was approximated by straight lines and two different algorithms were simulated utilizing the Simulink software.

Planned Activities

After specific 3-D models are completed, digital process management will be exercised, and the associated processes will be optimized using the DELMIA software in the framework of the 4+D Technology™. Since this application is similar to actual sequences of the construction process, this work will represent realistic processes. Various sorts of trial-and-error activities will be carried out to optimize the construction process. The next test sections for the S-CO_2 heat transfer coefficient measurements will encompass compact heat exchangers, in general, and Printed Circuit Heat Exchanger™-type channels, in particular. Thermo-hydrodynamic characteristics will be observed for different flow directions, pipe cross sections, and shapes.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Computational Models for Pyrochemical Electrorefiners of Nuclear Waste Transmutation Systems

PI (U.S.): Michael F. Simpson, Idaho National Laboratory (INL)

PI (ROK): Kwang-Rag Kim, Korea Atomic Energy Research Institute

Collaborators: Seoul National University (SNU), University of Idaho

Project Number: 2007-006-K

Program Area: AFCI

Project Start Date: November 2007

Project End Date: September 2010

Research Objectives

This research project will develop computational models of electrorefiners based on the fundamental understanding of the processes. Electrorefiners are the crucial components of pyroprocessing, a process in which spent fuel is separated into four streams: 1) uranium metal, 2) U/TRU metal, 3) metallic high-level waste containing cladding hulls and noble metal fission products, and 4) ceramic high-level waste containing sodium and active metal fission products. The primary goal of this proposed project is to develop rigorous yet flexible electrorefiner models. To support this objective, two research teams will 1) develop a computationally light and portable 2-D computational model and 2) investigate approaches to develop a computationally intensive, 3-D computational model for detailed and fine-tuned simulation of electrorefining processes. Successful completion of the proposed research should be beneficial for improving operations of current electrorefiners and for a better understanding of the fundamental aspects of electrorefining.

Research Progress

Data Compilation. Two engineering-scale electrorefiners (Mark-IV and Mark-V) in INL's Fuel Conditioning Facility have been used to treat spent fuel from the Experimental Breeder Reactor-II (EBR-II). Researchers selected the Mark-IV electrorefiner to model for its relative simplicity of spatial geometry, vast amount of data available, and accumulated operational experience. One of the repeatability runs performed for the EBR-II spent fuel treatment demonstration was selected as the reference case to be modeled. The tests were run with typical conditions and were extensively sampled. The team collected and distributed configuration and operational data from the Mark-IV electrorefiner in addition to species data. Based on their weight fractions within the salt, the top 15 major species involved in the process were tracked in the models. The related sample analysis results on fuels, salts, and products were collected, distributed, and reviewed by the participating organizations. The researchers also identified relevant thermodynamic, physical, and electrochemical properties.

Computational Platform Assessments. In view of the spatial characteristics provided, Mark-IV operational data, and model requirements, the researchers decided to use a combination of applications weighing on flexibility, long-term sustainability, and upgradeability. They selected REFIN-1D, a proprietary code at SNU, for electrochemistry computation. ANSYS-CFX, a commercial software package, will be used to deal with 3-D characteristics of molten salt flow and electric potential distribution. They chose a general scientific programming language, MatLab, as the development platform because of its flexibility in dealing with time-varying anode and cathode surface electrochemistry, molten salt electrolyte flow, and electric potential distribution in the molten salt electrolyte within a single application environment.

2-D Model Developments. Researchers have begun developing a 1-D model of the Mark-IV electrorefiner. The computations thus far have modeled the dissolution of the spent nuclear fuel at the anode, taking into account only uranium (U^{3+}), plutonium (Pu^{3+}), and zirconium (Zr^{4+}). Uranium and plutonium are the two elements of most importance in the system, while zirconium is the most

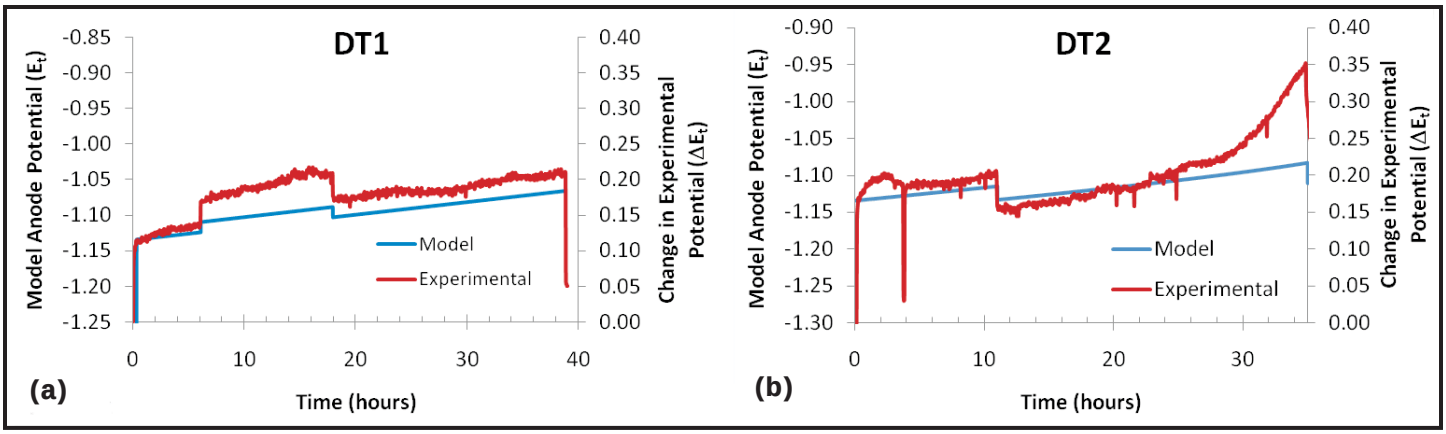


Figure 1. Anode potential comparison for two runs in repeatability tests.

active of the noble metals. Plutonium is quickly exhausted from the anode, followed by dissolution of primarily uranium, along with small amounts of zirconium. The total anode potential, as calculated by the model, has been compared to experimental data sets provided by INL (see Figure 1). The anode potential matches the experimental values quite well with root mean square values of 2.27 percent and 3.63 percent for two different data sets.

3-D Model Developments. The REFIN code for 1-D, time-dependent, electro-kinetic modeling was used to analyze the integral efficiency test data of the Mark-IV electrorefiner. The purpose of this analysis was to examine the capability of the REFIN model to describe the electrochemical process evolution. Resulting REFIN predictions, as seen in Figure 2, showed good agreement with the Mark-IV repeatability test data. Further improvements in predictions are expected with careful treatment of local species concentration changes on effective electrode surfaces.

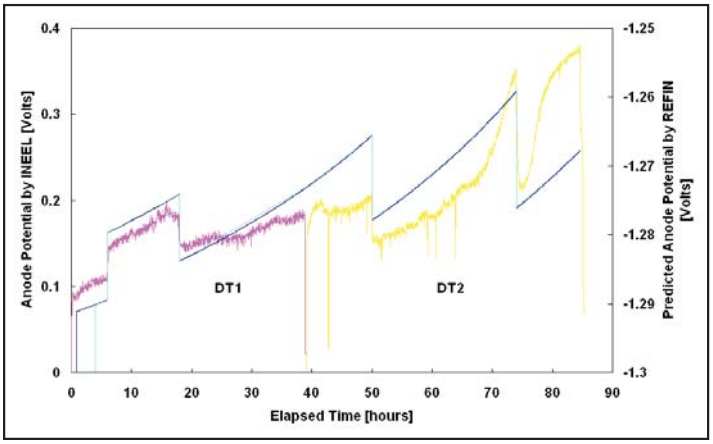


Figure 2. Anode potential comparison for two runs in repeatability tests by REFIN.

ANSYS-CFX addressed electrolyte fluid dynamics. In the simulation, the finite volume method solved the conservation equations of the species, mass, momentum, and energy. ICEM-CFD performed the meshing process in which a volume meshing was implemented using a tetra meshing with a set of curves and points. An analogy was assumed between the behavior of the heat and mass transfer in this approach. Thus, the concentration boundary layers near the electrode surfaces were effectively approximated with the temperature profiles calculated from the CFX solver by adjusting the heat fluxes through the anode and cathode surfaces. The prescribed computations effectively gave the diffusion layer thicknesses (see Figure 3) and found that heterogeneous diffusion boundary layers exist due to the flows incurred by various stirring devices such as the cadmium stirrer and the rotating cathode.

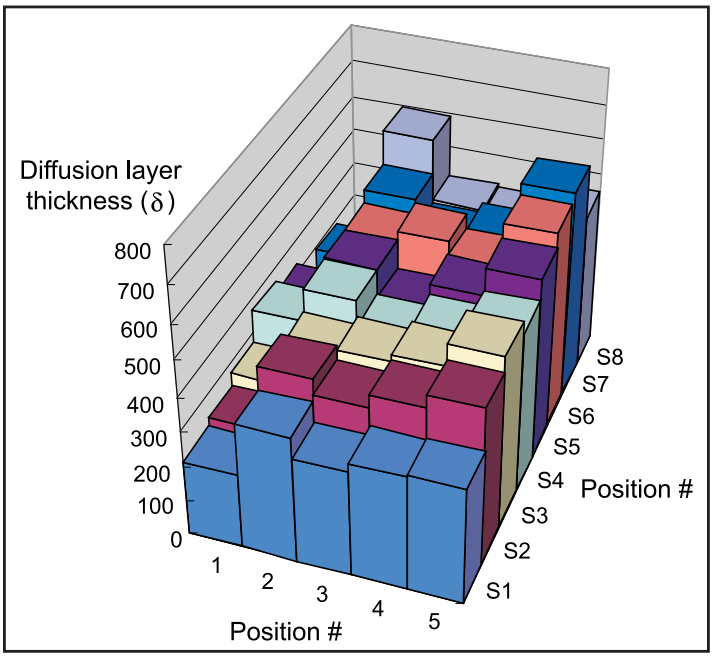


Figure 3. Calculated diffusion layer thicknesses on the cathode side wall.

Planned Activities

For the second year of the project, researchers will shift development focus to electrolyte fluid dynamics and electrochemical cathodic behaviors while increasing the number of species studied in order to give a more accurate model. They will also initiate the integration of the developed fluid dynamic model from ANSYS-CFX and the electrochemical models from REFIN codes in order to give a complete 3-D electro-fluid-dynamic model architecture. This integration and the associated computations may result in a computationally intensive model. A parallel computing system is being considered for effective computations.

Information relevant to the proposed computational modeling and validation tasks, including thermodynamic properties, physical properties, and electrochemical properties, will be continually updated and compiled, and researchers will initiate coordination efforts for model validation.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Sodium-Cooled Fast Reactor Structural Design for High Temperatures and Long Core Lifetimes/Refueling Intervals

PI (U.S.): James J. Sienicki, Argonne National Laboratory

PI (ROK): Gyeong-Hoi Koo, Korea Atomic Energy Research Institute (KAERI)

Collaborators: None

Project Number: 2007-007-K

Program Area: Generation IV

Start Date: November 2007

End Date: November 2010

Research Objectives

The objective of this project is to develop state-of-the-art analysis capabilities for the high-temperature structural design of sodium-cooled fast reactors (SFRs). Efforts will ultimately be focused on SFRs with higher core outlet temperatures (550°C or greater) than traditional SFRs. At high temperatures and long service lifetimes, the reactor structures may undergo creep and creep-fatigue, and accumulate inelastic strain as the reactor goes through operational cycles and transients. Structural stability of the reactor vessel and in-vessel components must also be assured during seismic events.

Research Progress

During the first year, researchers selected the advanced burner test reactor (ABTR) as the initial preconceptual reference design. The ABTR is a 250-MWt metallic-fueled pool-type SFR with core outlet and inlet temperatures of 510°C and 355°C, respectively, and a 60-year service lifetime. To assure the structural integrity of the ABTR preconceptual design, it is necessary to show that it can accommodate all of the specified duty cycle events corresponding to the Service Levels defined in the *ASME Boiler and Pressure Vessel Code Section III - Subsection-NH (ASME-NH)* with sufficient design margins. Researchers carried out a comprehensive review of the mechanical and thermophysical properties of materials available in the *ASME-NH* and identified suitable materials for use in ABTR components.

For the elevated temperature operating events, time-dependent structural damage such as creep rupture, creep ratchet strain, and creep-fatigue have to be evaluated. Researchers developed a set of duty cycle events for Service Levels A, B, C, and D for the ABTR with a supercritical carbon dioxide (S-CO_2) Brayton cycle power converter. They carried out structural evaluations for three groups of duty cycle events: 1) heat-up from hot standby to full power and reverse operation with hold at full power, 2) heat-up from refueling to full power and reverse operation with hold at full power, and 3) daily load following. The evaluations were carried out using the Structural Integrity Evaluation ASME-NH (SIE ASME-NH) computer code developed at KAERI that implements the rules of the *ASME-NH* and incorporates deterministic analyses of material inelastic behavior for use with results obtained from calculations assuming elastic behavior. Components for which structural evaluations were performed include the reactor vessel, reactor support, redan, reactor upper head, core support, and intermediate heat transport system piping. Elastic analyses were performed using the ANSYS finite element computer code (see Figure 1).

Researchers found that the original ABTR preconceptual design could not accommodate the creep-fatigue damage due to the repeated cycles of heat-up from hot standby to full power, hold at full power, and cool-down from full power to hot standby plus heat-up from refueling to full power, hold at full power, and cool-down from full power to refueling. Deadweight loadings on the core support structure were also found to exceed primary stress limits. Modifications were identified for the ABTR preconceptual design to enable the creep-fatigue limits to be met and for the core support to meet primary stress limits. Specific modifications entail increasing the reactor vessel height from the upper head to the sodium-free surface, decreasing

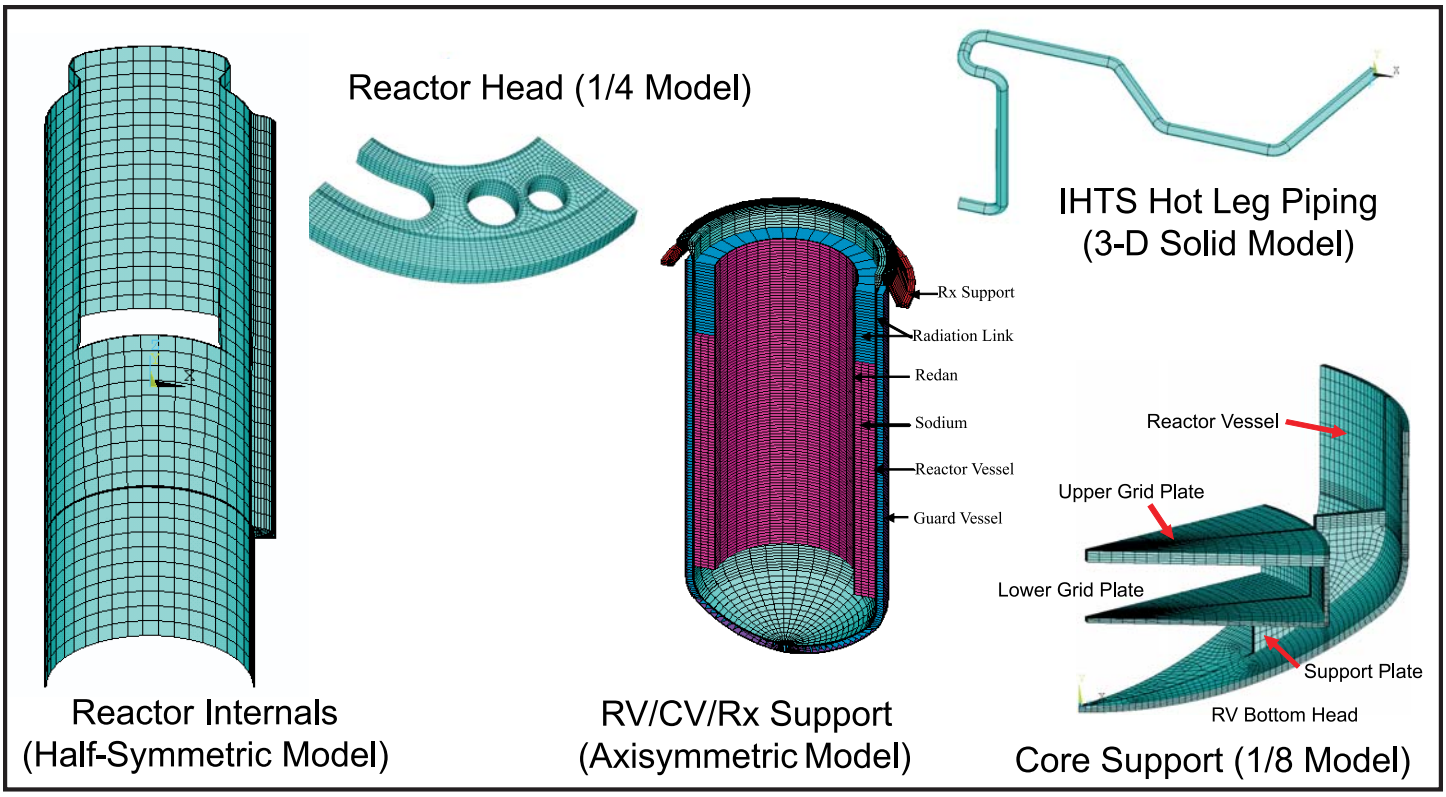


Figure 1. Finite element models for ABTR components for high-temperature design analysis.

the rate of temperature change in going between hot standby and full power from 100°C to 30°C per hour, thickening selected core support structural members, and increasing the gap between the reactor vessel and redan. Following these modifications, the creep-fatigue damage was reduced and satisfies the *ASME-NH* creep-fatigue rules. Load following operations affect the fatigue damage, but their effect is negligible; load following does not affect the creep damage. Figure 2 shows the results of the creep-fatigue evaluations for the modified ABTR preconceptual design and comparison with the *ASME-NH* limits. The modified core support with thicker members withstands the dead weight loads.

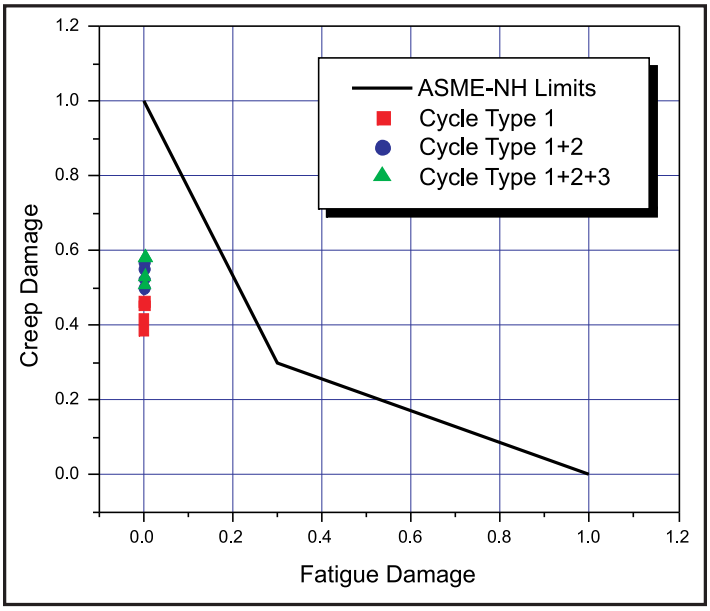


Figure 2. Results of creep-fatigue evaluation for selected ABTR duty cycle events for modified ABTR preconceptual design: Type 1 = Heat-up from hot standby to full power and cool-down to hot standby with hold at full power; Type 2 = Heat-up from refueling to full power and cool-down to refueling with hold at full power; Type 3 = Daily load following.

Planned Activities

In the second year of the project, researchers will develop an inelastic constitutive model for inelastic finite element analysis including the identification of material constants for and verification of the Chaboche model. Cyclic material tests will be carried out for Mod. 9Cr-1Mo steel. Structural integrity evaluations will be carried out with inelastic analyses using detailed finite element calculations. The results will be used to generate SIE ASME-NH input data and creep-fatigue damage will be evaluated. The new results will be compared with those from the first-year analyses providing a comparison of elastic and inelastic creep-fatigue damage evaluations. In the third year, seismic and buckling instability evaluations will be carried out.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Advanced Multi-Physics Simulation Capability for Very High Temperature Gas-cooled Reactors (VHTRs)

PI (U.S.): Changho Lee, Argonne National Laboratory

PI (ROK): Hyun Chul Lee, Korea Atomic Energy Research Institute

Collaborators: None

Project Number: 2008-001-K

Program Area: Generation IV

Project Start Date: November 2008

Project End Date: September 2011

Project Abstract

The objective of this project is to develop a suite of advanced multi-physics simulation methods and codes for high-fidelity, spatially detailed analysis of the coupled neutronic and thermo-fluid behavior of VHTRs. The focus is on reactors with prismatic block fuels. The research scope includes 1) the generation of neutronic and thermo-fluid models that accurately account for important physical phenomena in VHTRs and 2) coupling these models to obtain detailed, highly accurate multi-physics solutions.

The neutronics analysis capabilities to be developed include methods for generating fine-level multi-group cross sections that account for the double heterogeneity of coated particle fuels, plutonium resonances in the thermal energy range, and the thermal scattering kernel of graphite. Adaptive neutron transport solution methods will be developed for whole-core calculations ranging from homogenized assembly calculations with an in-line homogenization approach to detailed, 3-D whole-core transport calculations with explicit geometry modeling of fuel compacts and coolant holes.

The thermo-fluid analysis methods to be developed will accurately predict convective heat removal by the coolant flowing through the block and by multi-dimensional conduction within the block. Bypass coolant flow effects in the gap regions between blocks and radiative heat transfer will also be predicted. Various modeling strategies ranging from lower-order lumped-parameter models to higher-order large eddy simulations will be investigated, and a practical multi-resolution approach will be developed in which the local flow characteristics of "sub-domains" will be simulated using higher-order models; those results will be used to provide lower-order model calculations for the larger solution domain.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Experimental and Analytic Study on the Core Bypass Flow in a Very High Temperature Reactor

PI (U.S.): Richard Schultz, Idaho National Laboratory

PI (ROK): Min-Hwan Kim, Korea Atomic Energy Research Institute

Collaborators: None

Project Number: 2008-002-K

Program Area: Generation IV

Project Start Date: October 2008

Project End Date: September 2011

Project Abstract

Core bypass flow is a key issue in the very high-temperature reactor (VHTR) design for core thermal margins and target core exit temperature. In the prismatic modular reactor design, the core bypass flow is determined by the flow through the control element holes and through the radial and axial gaps between the graphite blocks for manufacturing and refueling tolerances. Bypass flow varies over core life because of irradiation swelling/shrinkage and thermal expansion of the graphite blocks. In a VHTR core consisting of stacked graphite blocks, the core bypass flow behavior occurs in a complicated, multi-dimensional way and it could be a significant proportion of the total core flow. Thus, an accurate prediction of the bypass flow and measures for its minimization are of major importance in assuring core thermal margins during both normal operation and accident conditions. Despite this importance, there has not been much effort in quantifying and accurately modeling the effect of the core bypass flow.

The objectives of this project are to 1) generate experimental data for bypass flow as a function of the inter-block gap size and block combination, 2) assess the thermo-fluid analysis tools for their accuracy and model improvements, and 3) identify and assess measures for reducing core bypass flow.

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

Nuclear Data Uncertainty Analyses to Support Advanced Fuel Cycle Development

PI (U.S.): Michael E. Dunn, Oak Ridge National Laboratory

PI (ROK): Choong-Sup Gil, Korea Atomic Energy Research Institute

Collaborators: None

Project Number: 2008-003-K

Program Area: AFCI

Project Start Date: November 2007

Project End Date: October 2010

Project Abstract

The main goal of this collaboration is to provide improved neutron cross-section data with uncertainty or covariance data for isotopes important to nuclear safeguards applications. Additional goals include 1) assessing uncertainties of the nuclear integral parameters due to the cross-section data, 2) improving safety validation, and 3) reducing capital cost through system design optimization for advanced fuel cycle (AFC) development.

To achieve the above objectives, researchers will first identify priority nuclides that need improved nuclear data with uncertainty information for AFC safeguards and design applications. Subsequently, they will produce improved nuclear data files with covariance data for these priority nuclides. The improved, evaluated cross-section data with covariance data will be demonstrated and tested in sensitivity/uncertainty calculations for select benchmark problems. The tests will include uncertainty assessments due to the nuclear cross-section data of the integral parameters for the AFC systems, such as $k_{eff.}$, reaction rates, reactivity worths, and spent fuel isotopic compositions, as well as validation of the cross-section data itself.

Potential benefits of this activity include capital cost reductions through system design optimization, improved technical basis for licensing, improved safety validation, increased design flexibility/optimization, and potentially significant time and cost savings associated with the design and conduct of experimental programs. The final product will be improved and tested nuclear data files with uncertainty information that will be submitted for inclusion in the ENDF/B file system.

Index of I-NERI Projects

FY 2004 Projects

FY 2005 Projects

FY 2006 Projects

FY 2007 Projects

FY 2008 Projects