International Nuclear Energy Research Initiative

Fiscal Year 2012 Annual Report

Cover

Image is of the Sizewell Nuclear Power Plant, UK. Photo courtesy of Public Domain

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

Foreword

Nuclear energy represents the single largest carbon-free baseload source of energy in the United States, accounting for nearly 20 percent of the electricity generated and over 60 percent of our low-carbon production. Worldwide, nuclear power generates 14 percent of global electricity. Continually increasing demand for clean energy both domestically and across the globe, combined with research designed to make nuclear power ever-safer and more cost-effective, will keep nuclear in the energy mix for the foreseeable future.

U.S. researchers are collaborating with nuclear scientists and engineers around the world to develop new technologies that will lower costs, maximize safety, minimize proliferation risk, and handle used fuel and radioactive waste. These are international concerns that the international nuclear community must address together. Just as all nations stand to benefit from nuclear energy, the risk of nuclear proliferation or the consequences of an accident know no national borders. Bilateral and multilateral collaborations build international consensus, capitalize on limited resources, and promote innovation far more effectively than any one nation can do alone.

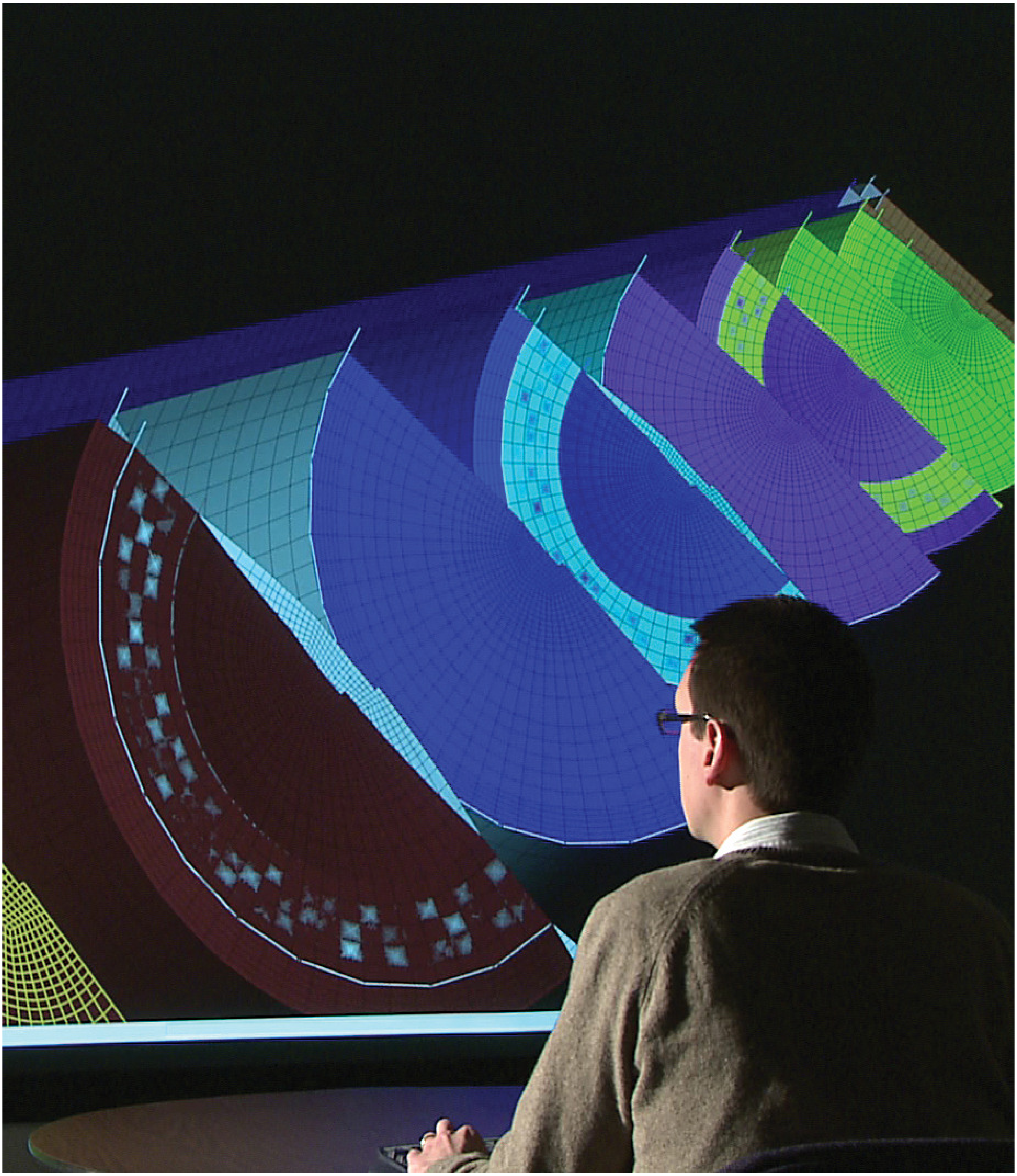
The International Nuclear Energy Research Initiative, or I-NERI, is perhaps even more relevant today than at its establishment. Designed to foster bilateral international partnerships, I-NERI crosses both geographical and institutional boundaries, forging teams from universities, industry, and government organizations including federal laboratories. I-NERI agreements have resulted in collaborative research and development that investigates next-generation nuclear systems and fuel cycles, helping to determine tomorrow's solutions to today's challenges. I-NERI research teams have made substantial contributions to the knowledge base that directs critical decisions about nuclear energy.

This annual report provides information on how these efforts are collectively helping to establish a solid foundation for advanced nuclear technologies. One project at a time, the global nuclear community is building tomorrow's nuclear energy systems and technologies.

Peter B. Lyons
Assistant Secretary, Office of Nuclear Energy
U.S. Department of Energy

Table of Contents

Sizewell Nuclear Power Plant, UK. Photo courtesy of Public Domain



Modeling and Simulation of Accident Tolerant Fuels,
Photo courtesy of Idaho National Laboratory

About This Document

The International Nuclear Energy Research Initiative (I-NERI) supports the advancement of nuclear science and technology through bilateral collaborative research between the United States and international partners. I-NERI is one of several mechanisms the U.S. Department of Energy's Office of Nuclear Energy (DOE-NE) utilizes to promote 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 teaming with international partners, U.S. researchers gain broader perspectives on issues of global importance and can potentially achieve faster results at reduced cost.

The I-NERI 2012 Annual Report provides an update on I-NERI accomplishments achieved during Fiscal Year (FY) 2012, including final activities and findings of completed projects and comprehensive progress summaries of ongoing projects.

- Section 1 provides an overview of the I-NERI program, including its purpose and processes. This section also provides updates on funding and program participants.
- Section 2 describes the scope of work supported by I-NERI collaborations.
- Section 3 provides a summary of FY 2012 program and collective project accomplishments.
- Section 4 presents the R&D work scope for current I-NERI collaborative projects between the United States and the two currently active partners: the European Union and the Republic of Korea. For these partnerships, the report provides summary information for individual projects that were initiated, ongoing, or completed in FY 2012.

Korean Ulchin Nuclear Power Plant
Photo courtesy of Public Domain

1 The I-NERI Program

1.1 Purpose

The United States is confronting two powerful imperatives that are driving today's resurgence of nuclear power: the escalation of energy demands parallels the increasing urgency to establish a clean energy portfolio to meet these demands. Nuclear power plants presently provide more than 60 percent of our low-emission energy supply. However, expanded use of nuclear energy as a power source faces obstacles that require further R&D for resolution. In April 2010, the U.S. Department of Energy, Office of Nuclear Energy (DOE-NE), published the Nuclear Energy Research and Development Roadmap[1] (henceforth "the Roadmap"), which defines four objectives to address existing challenges:

- Develop technologies and other solutions that can improve the reliability, sustain the safety, and extend the life of current reactors.
- Develop improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration's energy security and climate change goals.
- Develop sustainable nuclear fuel cycles.
- Understand and minimize risks of nuclear proliferation and terrorism.

In 2010, former Secretary Chu established the Blue Ribbon Commission on America's Nuclear Future to conduct a comprehensive review and develop recommendations for the backend of the fuel cycle. In January 2013, DOE issued the Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste. In addition to responding to recommendations of the Blue Ribbon Commission on America's Nuclear Future, that Strategy establishes the Administration's policy regarding the importance of addressing the disposition of used fuel and high-level waste.

Fortunately, the United States stands with a global nuclear community that shares our desire to resolve these challenges. Scientists worldwide are exploring ways to improve today's processes and to discover and implement next-generation methods that will ultimately promote safer and more cost-efficient use of nuclear energy. The nature of these partnerships, whether bilateral or multilateral, takes on two different but complementary roles: the United States advances the state of scientific knowledge and benefits from existing research by collaborating with countries that have mature nuclear energy programs, while providing useful assistance to those countries with developing technology and advancing global nuclear safety standards and non-proliferation frameworks. As noted in the Roadmap on page xi: "There is potential to leverage and amplify effective U.S. R&D through collaboration with other nations via multilateral and bilateral agreements."

1 http://energy.gov/sites/prod/files/NuclearEnergy_Roadmap_Final.pdf

In an increasingly global society, the importance of international cooperation has escalated beyond the advantages of information sharing. The Roadmap points out (on page v) that "international expansion of the use of nuclear energy raises concerns about the proliferation of nuclear weapons stemming from potential access to special nuclear materials and technologies." " Bilateral and multilateral R&D collaborations develop a common global understanding, and an integrated development approach incorporating nonproliferation, safeguards, and security technologies into the most basic elements of nuclear systems and fuel, thereby strengthening nonproliferation frameworks and protocols.

In response to the clear need for global cooperation in the nuclear arena, NE initiated the International Nuclear Energy Research Initiative (I-NERI). The I-NERI program, one of several international partnerships managed by the Office of International Nuclear Energy Policy and Cooperation (NE-6), promotes bilateral research to expand the contribution of nuclear power towards meeting U.S. energy goals. Bilateral agreements established by NE provide frameworks for innovative scientific and engineering R&D undertaken in cooperation with partnering international countries, foster closer collaboration among international researchers, improve communications, and promote sharing of nuclear research information.

Cooperative research projects funded through the I-NERI program are aligned with major NE R&D programs and ultimately contribute to the NE objectives noted in the Roadmap. Projects conducted under the I-NERI umbrella aim 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.

• Promote bilateral and multilateral collaboration with international agencies and research organizations to improve the development of nuclear energy.

• Promote a nuclear science and engineering infrastructure to meet future technical challenges.

Through the I-NERI program, NE has coordinated wide-ranging scientific discussions among governments, industry, academia, and the worldwide research community regarding the development of advanced reactor concepts and advanced fuel cycles. Figure 1 illustrates key features of the I-NERI program.

1.2 International Agreements

In order to initiate an international collaboration, a government-to-government agreement must first be in place. I-NERI bilateral agreements serve as vehicles to conduct joint R&D under various NE programs, enabling U.S. researchers to establish collaborative projects with their international colleagues that support development of next-generation nuclear energy systems and fuel cycle technologies.

To date, NE has implemented agreements with six countries and two international organizations, signed by DOE and the international partners noted in Table1. The table also presents the number of projects undertaken to date with each partner.

Figure 1. I-NERI program features.

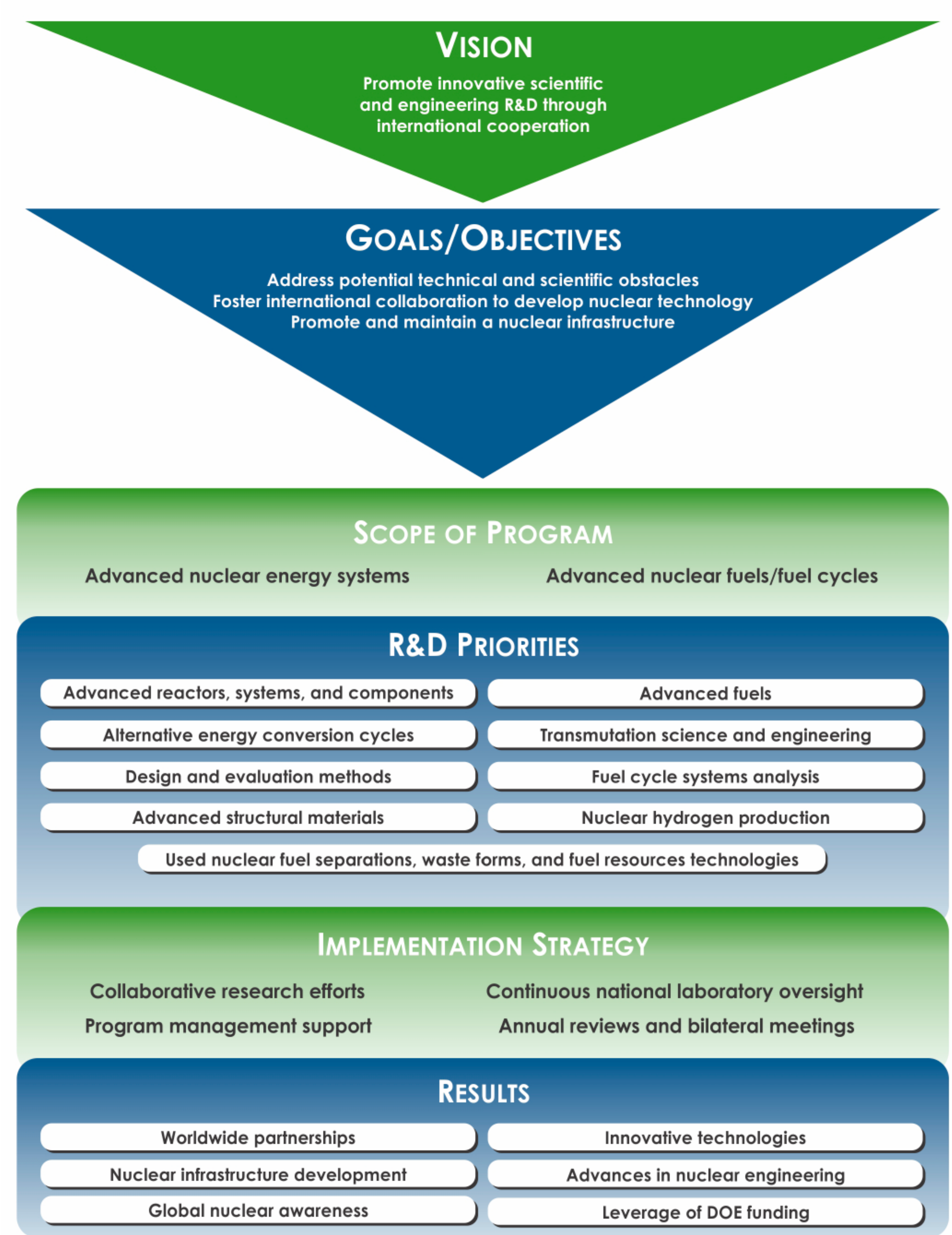


Table 1. I-NERI international partners and number of projects awarded.

Collaborator	Organization	Total
Brazil	Ministério da Ciência e Tecnologia (MST)	2
Canada	Department of Natural Resources Canada (NRCan) and Atomic Energy of Canada Limited (AECL)	12
European Union	European Atomic Energy Community (Euratom)	23
France	Commissariat à l'énergie atomique (CEA)	21
Japan	Agency of Natural Resources and Energy (ANRE) and the Ministry of Education, Culture, Sports, Science, and Technology (MEXT)	2
Republic of Korea	Ministry of Education, Science and Technology (MEST)[2]	46
Organisation for Economic Co-operation and Development (OECD)	The Nuclear Energy Agency (NEA) of OECD	1
Republic of South Africa	The Government of the Republic of South Africa	--
	Total	**107**

The I-NERI program is a spoke in a much larger wheel of U.S. participation in the international nuclear energy community. Outside the I-NERI program, the United States has negotiated bilateral action plans with Australia, China, India, Japan and Russia. There is also extensive multilateral collaboration with the international community via the Generation IV International Forum (GIF); the International Atomic Energy Agency (IAEA); the International Framework for Nuclear Energy Cooperation (IFNEC); the International Nuclear Cooperation (INC) framework, a cooperative effort with Eastern European countries and the former Soviet Union; and the Nuclear Energy Agency (NEA) within the Organisation for Economic Co-operation and Development (OECD). NE-6 coordinates U.S. involvement in each of these programs. Please visit their websites for more information.[3]

1.3 Funding

I-NERI is an important vehicle for enabling international R&D in nuclear technology on a leveraged, cost-shared basis. Each country in an I-NERI collaboration provides funding for its respective project participants; funding provided by the United States may be spent only by U.S. organizations. The United States funds I-NERI projects through its national laboratories, with the annual contribution based upon current-year budgets for NE R&D programs. I-NERI projects typically last three years, although budgeting protocols require the U.S. portion to be funded annually. While 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. This section provides approximate domestic and international funding numbers for FY 2012 and the I-NERI program to date.

2 Signatory agency was the Ministry of Science and Technology (MOST), superseded by MEST in March 2008.

3 http://www.gen-4.org, http://www.iaea.org/, http://www.ifnec.org/, http://insp.pnnl.gov/, and http://www.oecd-nea.org/.

In FY 2012, the United States provided $6.11 million to support I-NERI projects: $5.89 million toward ongoing projects and $225,000 to launch a new collaborative project with Euratom. International funding for FY 2012 was $5.62 million: $1.95 million from Euratom and $3.67 million from the Republic of Korea. The total pledge for the FY 2012 project (i.e., the three-year sum) is $1,237,500, with the United States committing $562,500 and Euratom providing $675,000.

To date, I-NERI sponsors have committed a total R&D investment of $261.8 million: $139.9 million contributed by the United States and $121.9 million by international collaborators (see Figure 2).

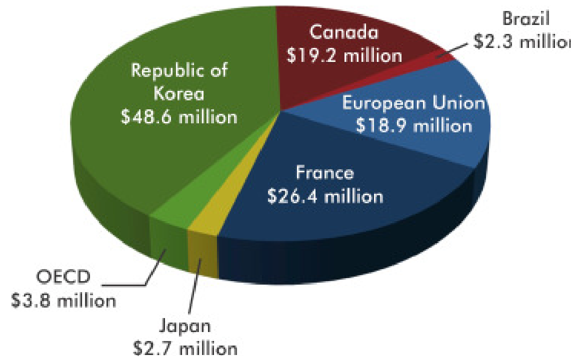


Figure 2. Breakdown of $121.9 million in foreign funding since I-NERI's inception, by international collaborator (numbers are approximate).

1.4 Program Participants

I-NERI encourages global sharing of resources. The program crosses both institutional and geographical boundaries, soliciting projects from international proposal teams that comprise participants from universities, industry, and government organizations, including federal laboratories. 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. I-NERI collaborative projects produce findings that would take the United States alone far more time and money to accomplish. The international infrastructure also brings multiple perspectives and priorities together to address shared obstacles. Figure 4 (pages 8 and 9) shows the broad spectrum of participants in the I-NERI program since its inception; asterisks indicate participants in ongoing projects.

Student Participation

One benefit of the I-NERI program is development of nuclear-related educational research opportunities. Encouraging young academics to participate in nuclear R&D promotes the nuclear science and engineering infrastructure, both in the United States and abroad. Support from the I-NERI program helps educational institutions remain at the forefront of science education and research, advance the important work of existing nuclear R&D programs, and create training for the next generation of nuclear scientists and engineers—those who will resolve future technical challenges. In FY 2012, ten U.S. and six foreign academic institutions participated in I-NERI research projects. As shown in Figure 3, approximately 33 U.S. students and 31 students from partner countries worked on active I-NERI research projects.

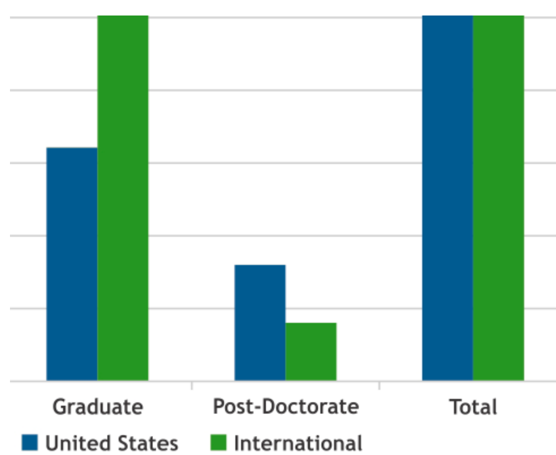


Figure 3. I-NERI student participation broken down by degree level (numbers are approximate).

Figure 4. Research institutions that have contributed to I-NERI collaborations.

KEY

*	FY 2012 participant
[G]	Government agency
[U]	University
[I]	Industry

Brazil

Eletronuclear [I]

Instituto de Pesquisas Energéticas e Nucleares (IPEN) [G]

Ministério da Ciência e Tecnologia (MST) [G]

Canada

Atomic Energy of Canada Limited (AECL) [I]

Chalk River Laboratories (CRL) [G]

CanmetENERGY [G]

Ecole Polytechnique de Montréal [U]

Gamma Engineering [I]

Université Bordeaux [U]

Université de Sherbrooke [U]

University of Manchester [U]

University of Manitoba [U]

Republic of Korea

Cheju National University [U]

Chosun University [U]

Chungnam National University *[U]

Hanyang University [U]

Korea Advanced Institute of Science and Technology (KAIST) *[G]

Korea Atomic Energy Research Institute (KAERI) *[G]

Korea Electric Power Research Institute (KEPRI) [G]

Korea Hydro and Nuclear Power Company (KHNP) [I]

Korea Maritime University [U]

Kyung-Hee University *[U]

Pusan National University [U]

Seoul National University *[U]

Ulsan National Institute of Science and Technology *[U]

Italy

Ente per le Nuove Tecnologie, l'Energia e l'Ambiente (ENEA) [G]

Società Informazioni ed Esperienze Termoidrauliche (SIET) [I]

Switzerland

Paul Scherrer Institute (PSI) *[G]

National Laboratories and Government Organizations

Argonne National Laboratory (ANL) *

Brookhaven National Laboratory (BNL)

Idaho National Laboratory (INL) *

Lawrence Livermore National Laboratory (LLNL)

Los Alamos National Laboratory (LANL) *

National Institute of Standards and Technology (NIST) *

Oak Ridge National Laboratory (ORNL) *

Pacific Northwest National Laboratory (PNNL) *

Sandia National Laboratories (SNL) *

Industry

Gas Technology Institute

General Atomics

Ultra Safe Nuclear Corporation, Inc. *

Westinghouse Electric

Universities

Colorado School of Mines *

Idaho State University *

Iowa State University

Massachusetts Institute of Technology

Ohio University *

The Ohio State University

Pennsylvania State University

Purdue University

Rensselaer Polytechnic Institute

Texas A&M University *

University of California–Berkeley *

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 North Texas *

University of Tennessee *

University of Wisconsin *

Utah State University

Belgium

Ghent University *[U]

Joint Research Centre–Institute for Energy (JRC-IE) *[G]

Joint Research Centre–Institute for Reference Materials and Measurements (JRC-IRMM) *[G]

Joint Research Centre–Institute for Transuranium Elements (JRC-ITU) *[G]

SCK•CEN (Belgian Nuclear Research Centre) *[G]

UK

University of Ontario Institute of Technology [U]

France

Commissariat à l'énergie atomique (CEA) *[G]

Electricité de France (EDF) [G]

Framatome ANP [I]

Laboratoire des Composites Thermostructuraux (LCTS) [G]

Organisation for Economic Co-operation and Development–Nuclear Energy Agency (OECD/NEA) [G]

Germany

Karlsruhe Institute of Technology *[U]

Japan

Hitachi, LTD [I]

Hitachi Works [I]

Japan Atomic Energy Agency (JAEA) [G]

Japan Atomic Energy Research Institute (JAERI) [G]

Tohoku University [U]

Toshiba Corporation [I]

University of Tokyo [U]

Netherlands

Nuclear Research and Consulting Group *[I]

German Nuclear Power Plant
Photo courtesy of Public Domain

2 Research Work Scope

I-NERI project work scopes are jointly developed by the United States and the collaborating country based on terms of the bilateral agreement and current common R&D needs. Potential topics for collaboration in the coming year are selected at

each annual joint planning and project review meeting. For the United States, the work scope of I-NERI projects must be directly linked to the scientific and engineering needs of the principal NE research programs, currently Reactor Concepts Research, Development and Demonstration (Reactor Concepts RD&D) and Fuel Cycle Research and Development (FCR&D).

Past I-NERI project teams have contributed to the extensive and growing knowledge base in both of these R&D programs. In support of reactor concepts, I-NERI projects have investigated such topics as next-generation materials (e.g., nano-composited and oxide dispersion-strengthened steels, silicon carbide composites, and zirconium alloys); improved energy conversion through the Brayton cycle; advanced sensors, instrumentation, and controls; hydrogen production through thermochemical reactions and high-temperature electrolysis; production viability and efficiency; safety issues and proliferation risk reduction; and analysis of distribution and storage methods. In addition, the program has contributed to our knowledge of modeling and simulation; some of the first I-NERI project teams utilized these tools, and roughly half the current projects explicitly call for the use, qualification, and/or improvement of numerical techniques within their objectives. Sample new nuclear fuels being investigated include advanced transmutation fuels and inert matrix fuels. Fuel cycle research includes waste forms, separation of fission products from nuclear waste, and advanced head-end processes that condition spent fuel.

This section provides an overview of the current work scopes for these NE R&D programs.

2.1 Reactor Concepts Research, Development and Demonstration

The mission of the Reactor Concepts RD&D program is to develop new and advanced reactor designs and technologies with broader applicability and improved affordability and competitiveness. RD&D activities address technical, cost, safety, and security challenges associated with the program elements described below.

Advanced Reactor Concepts (ARC)

ARC is an expanded version of the Generation IV (Gen IV) research program. The program sponsors RD&D that explores advanced reactor technologies and methods that offer the potential for reduced capital and operating costs, better performance, enhanced safety, and reduced proliferation risk. Focus areas include advanced thermal and fast neutron spectrum systems.

Next Generation Nuclear Plant (NGNP)

The NGNP will demonstrate the technical viability of high-temperature gas-cooled reactor (HTGR) technology for the production of electricity and high- temperature process heat for industrial use. DOE provides support through R&D ranging from understanding fundamental nuclear phenomena to developing advanced fuels that could improve the economic and safety performance of these advanced reactors.

Small Modular Reactor (SMR) Advanced Concepts R&D

SMRs provide simplicity of design, enhanced safety features, the economics and quality afforded by factory production, and more flexibility (financing, siting, sizing, and end-use applications) than larger nuclear power plants. SMRs can provide power for applications where sites lack the infrastructure to support a large unit, such as smaller electrical markets, isolated areas, smaller grids, sites with limited water and acreage, or unique industrial applications. Thus this concept is of benefit not only to countries with highly developed economies and mature nuclear programs but also to those with limited resources. The program supports RD&D activities that advance the understanding and demonstration of these innovative reactor technologies and concepts.

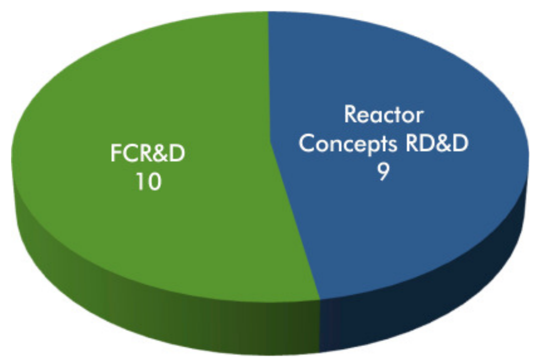


Figure 5. Ongoing projects by program area.

Light Water Reactor Sustainability

Over the next three decades, most currently operating nuclear power plants will reach the ends of their operating licenses. To help meet the nation's expanding electricity requirements, however, existing reactors must continue operating beyond 60 years. This program supports acquiring the scientific understanding needed to develop and demonstrate technologies that support safe and economical long-term operation of existing reactors, as well as new technologies that enhance performance. Extending the life of the current fleet requires fundamental science to predict and measure changes in materials, systems, structures, and components as they age in environments found within operating reactors.

Advanced Modeling and Simulation

Two major initiatives are developing the advanced modeling and simulation that will provide the validated tools necessary to enable fundamental change in how the United States designs and manages both existing and future nuclear facilities. The Consortium for Advanced Simulation of Light Water Reactors (CASL) is a DOE-sponsored energy innovation hub that brings together industry, universities, and the national laboratories to create a virtual model with advanced predictive and simulation capabilities for the current generation of reactors. The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program is a longer-term initiative that is developing a suite of integrated codes and cross-cutting numerical tools to support the entire range of NE's nuclear energy R&D.

2.2 Fuel Cycle Research and Development

One of the four NE R&D objectives delineated in the 2010 Nuclear Energy Research and Development Roadmap is achievement of sustainable fuel cycle options, defined as those that improve uranium resource utilization, maximize energy generation, minimize waste generation, improve safety, and limit proliferation risk. Through a long-term science-based approach, the FCR&D program is developing a suite of technology options that will enable informed decisions about the management of nuclear waste and used reactor fuel.

Today's commercial nuclear power plants run on a once-through fuel cycle, which utilizes only about five percent of the fuel's energy potential before the used fuel is placed in storage for future disposal. Advanced fuel cycles and techniques are being developed to process this fuel for re-use in reactors. This may allow substantially more energy to be drawn from the same amount of nuclear material, while reducing the quantity of long-lived radioactive elements in the spent fuel. Fuels R&D aims to increase the efficient use of uranium resources, reduce the amount of used fuel requiring disposal, and evaluate the inclusion of non- uranium materials to reduce the long-lived radioactive elements in used fuel. In a full recycle system, only waste products require disposal, not used fuel. The mission of the FCR&D program is to develop used nuclear fuel management strategies and technologies to support meeting federal government responsibility to manage and dispose of the nation's commercial used nuclear fuel and high-level waste; develop sustainable fuel cycle technologies and options that improve resource utilization and energy generation, reduce waste generation, enhance safety, and limit proliferation risk. The program is applying a methodical systems engineering analysis approach to identify suitable technologies.

FCR&D has five campaigns addressing different program elements that support the overall objectives of sustainable fuel cycle options.

Fuel Cycle Options

This campaign is developing processes to evaluate alternative fuel cycles in order to guide the selection of one or more sustainable options and prioritize associated R&D needs. Researchers perform integrated fuel cycle analyses and technical assessments and provide information that can be used to inform NE's fuel cycle R&D activities and strategies. The campaign is studying options for once-through, modified recycle and full recycle systems, looking at all stages from mining to disposal. Research under this initiative also supports the sodium-cooled fast reactor (SFR), whose design is geared toward improved management of high-level wastes, specifically actinides.

Advanced Fuels

NE is supporting R&D for various fuel forms, including cladding, needed to implement those fuel cycle options. For any given fuel type, fuel qualification requires engineering-scale demonstration of the fabrication processes and irradiation of lead-test assemblies to demonstrate in-pile performance. NE is working toward developing a state-of-the art R&D infrastructure to support a goal-oriented science-based approach. The campaign is investigating both transmutation fuels, which have potential for enhanced resource utilization and proliferation resistance, and next-generation light water reactor (LWR) accident-tolerant fuels that exhibit enhanced performance and safety and reduced waste generation.

Separations, Waste Forms, and Fuel Resources

Features of a sustainable fuel cycle include reduced processing, waste generation, and proliferation risk. NE is working towards next-generation fuel cycle separation and waste management technologies that enable these improvements. Research scope includes developing advanced waste forms and developing and demonstrating technologies that separate transuranic elements and fission products from used nuclear fuel, as well as investigating cutting-edge alternatives that could widen fuel cycle options.

Used Fuel Disposition

This program aims to provide a sound technical basis for a new national policy to manage the back end of the nuclear fuel cycle, including identification and evaluation of safe and secure options for storage, transportation, and permanent disposal of radioactive wastes resulting from existing and future fuel cycles. Objectives range from identifying and addressing gaps in existing data and methodologies to using advanced modeling tools for studies of potential disposal system concepts and environments; research scope ranges from interim storage to final disposal. International activities are a cornerstone of these efforts and include participation in international working groups addressing relevant challenges, support for bilateral interactions between the United States and the Republic of Korea and the United States and Japan, and planning for U.S. involvement in disposal R&D in European underground research laboratories (URLs).

Material Protection, Accounting, and Control Technologies

As nuclear systems grow more complex and their use becomes more widespread, NE aims to enhance safety and security while minimizing proliferation risk. This program supports efforts to develop innovative technologies and analysis tools to enable next-generation nuclear materials management for future U.S. nuclear fuel cycles, significantly advancing the state of the art in accounting and control. NE is also supporting the necessary research to integrate safeguards and security into the earliest stages of the design cycle.

3 Summary of FY 2012 Accomplishments

3.1 Research Activities

Since the program's inception, 107 projects have been awarded. There are currently 18 active projects, plus one collaboration completed in FY 2012. Both the number of awards and the consistency of project achievement demonstrate I-NERI's success in fostering international collaboration. The following sections provide an overview of ongoing and recently completed research in each of the two program areas. A fuller summary of project activities and findings is presented in Section 4 of this annual report.

Reactor Concepts RD&D

Of the nine ongoing Reactor Concepts RD&D projects, three are investigating advanced materials for nuclear systems. Five are improving tools for nuclear studies, generating, validating and/or upgrading existing data/databases and modeling/ simulation codes. The ninth is dedicated to nuclear plant safety.

One materials-related study is developing micro- and nanoscale testing techniques that researchers can use to assess mechanical property changes due to irradiation on the macro scale, thus assessing a material's viability to endure the harsh conditions inside a nuclear reactor. The research team has developed a range of test samples and begun testing, comparing effectiveness and accuracy of different methods.

The other two projects are investigating ways of lowering susceptibility to stress corrosion cracking (SCC). One project is studying thermal treatments of Alloy 690 to affect atomic re-ordering. Researchers have subjected the alloy to various thermomechanical treatments and are testing the samples. Results thus far show that some treatments—especially cold working—change the material's behavior significantly. Neutron diffraction indicates that lattice contraction occurs during aging at 400°C, presumably resulting from the development of short-range order (SRO) in the alloy during low- temperature annealing—a hypothesis which further studies have so far confirmed.

The second SCC project is investigating the effects of aging on the microstructure of a dissimilar metal weld—specifically, the chromium dilution effect at the weld interface formed when joining low-alloy steel (LAS) to Alloy 690 with an Alloy 152 weld filler. The researchers have produced and characterized a representative dissimilar weld mock-up and found that thermal aging concentrates chromium and nickel in the weld root region. The concentrated chromium contributes to growth and formation of chromium and nickel in the weld root region. The concentrated chromium contributes to growth and formation of chromium precipitates in the fusion boundary region. Analysis results for specimens tested in PbO + 0.1 M NaOH solutions at 315°C show that PbO content in the solution degrades the passivity of oxide formed on Alloy 600 and leads to oxide change in chemical composition, ultimately increasing SCC susceptibility.

While these projects are adding to the knowledge of materials, a fourth team is ensuring the resulting data can be fully leveraged for future progress by addressing the interoperability of materials databases, thus facilitating data exchange among research partners. In FY 2012, team members generated upgraded schemas for the U.S. Gen IV Materials Handbook database and for MatDB, the database housed in the European Commission's Joint Research Centre-Institute for Energy and Transport. As the two sets of schemas must be sufficiently compatible to enable data exchange, the research teams have mapped specific attributes between schemas and identified commonalities. Results show promise for future interoperability.

Another project is developing an instrument for high-resolution measurements of fission fragment velocity, energy, and charge. Resulting data will build upon existing nuclear data files for several key isotopes crucial to future fast reactor systems. The advanced instrument will detect and measure both fission fragments simultaneously, unlike current instruments that only detect a single fragment. Work during the first year focused on testing individual parts of an array that measures time-of-flight with high resolution based on thin conversion foils, electrostatic mirrors and ultra-fast microchannel plates. The final design will also incorporate energy measurements; a prototype ionization chamber has been constructed, and work is under way on a second device with improved resolution.

Another data project team is archiving loading records from the ZPPR-15 benchmark test program to generate high-fidelity as-built Monte Carlo models from these valuable metallic fuel measurement data. The team has reviewed all drawer master identifications and logbooks from ZPPR-15 Phases A, B, C and D, ensuring future availability of data, and has analyzed 12 experimental configurations and provided various modeling approximations for the drawer masters. Project researchers have also generated 35 Monte Carlo models for the BFS-76-1A critical experiment, a transuranic (TRU) burner physics experiment.

Another project is verifying and validating a suite of high-fidelity multi-physics simulation methods for advanced nuclear reactors (developed by a recently completed I-NERI project) by comparing results against MCNP (a Monte Carlo N-Particle code) solutions and experimental measurements for reactor cores. The main research focus for FY 2012 was to refine the DeCART neutronics code, provide appropriate benchmark problems for reactor types of interest, use the problems to perform verification tests, and update cross-section libraries. The researchers seek novel methods to reduce errors. For example, they proposed a new azimuthal angle discretization scheme adopting Gaussian quadrature, which reduces the error caused by the azimuthal angle discretization by a factor of 2 to 5. For accurate and reliable DeCART solutions, the research team developed a subgroup cross-section library generation tool and procedure, which together produce the reference resonance integrals using MCNP5. Preliminary test results indicate that the estimated resonance cross sections for major actinides were accurate.

The new project initiated in FY 2012 will contribute to modeling and simulation advances, with cross-code verification of the high-fidelity computational fluid dynamics (CFD) models applied to nuclear reactor core flows. Four partner institutions will systematically cross-verify models of nuclear reactor core geometries characterized by either rod bundles or pebble beds, using common data shared among team members.

The final Reactor Concepts RD&D project is developing and demonstrating advanced plant monitoring, diagnostics, and prognostics methods for beyond design-basis accidents. Self-powered sensors and sensor networks will provide plant information during extreme conditions. Both domestic and foreign teams reviewed the respective safety critical functions and components of typical LWRs in their countries, developing suggestions for technology to monitor beyond design-basis accidents. Researchers are identifying requirements, methodologies and resources for the development of a system for safety index monitoring during severe accidents (SIMSA). Team members conducted proof-of-concept testing for time distribution mapping and signal transformation to a non-uniform bin space, successfully demonstrating a transient prognostic technique. Researchers have also developed process equipment monitoring and prognostic tools for diagnostic application: a MATLAB process equipment monitoring (PEM) toolbox and a process and equipment prognostics (PEP) tool. The team has also begun developing and demonstrating wireless technologies for condition monitoring during a station blackout and has narrowed findings to produce a recommended network.

FCR&D

Of the ten ongoing FCR&D projects, three are investigating alloys that will withstand high temperatures and high radiation levels, two are contributing to fundamental understanding of actinides, four are investigating advanced fuels for LWRs or fast reactors, and the tenth project contributes directly to the back end of the fuel cycle by researching separations to improve waste management of spent nuclear fuel.

This year marked the completion of a collaborative research effort with the Republic of Korea to refine the microstructure of austenitic stainless steels for enhanced radiation resistance and evaluate the resulting properties. The research team applied equal channel angular processing (ECAP)—a promising technique for thermomechanical treatment of a variety of steels and alloys—to change the chemistry and refine the size and distribution of oxide nanoparticles. Compared with their coarse-grained counterparts, the nanograined steels have shown extraordinary tolerance against helium, krypton and iron ion irradiation while maintaining high strength and reasonable ductility.

A second project team is also researching nanoparticle-strengthened steel, combining two advanced materials processing technologies used to produce nanostructured ferritic alloys (NFAs) and ferritic/martensitic (F/M) dual-phase steels. In FY 2012, they applied thermal treatments to two oxide dispersion-strengthened (ODS) 9Cr alloys and characterized the treated materials, determining that higher carbon content in a NFA alloy does not improve fracture toughness. Partial phase transformation heat treatment increased one alloy's strength and uniform elongations, while thermal annealing improved fracture toughness for the alloys, and controlled hot-rolling resulted in even more significant improvements.

A third collaboration is developing and characterizing advanced ODS and F/M steels, investigating corrosion effects of these steels in sodium, and examining the ODS steels' irradiation behavior. They are utilizing several irradiation programs to prepare materials for post-irradiation analysis and to gather data for modeling. Team members have produced heats of three ODS alloys, are sharing testing data, and are conducting long-term corrosion testing on ODS alloys in lead and lead–bismuth.

To improve fundamental understanding of actinide elements, one project team is producing new and essential experimental data on fission fragment mass distributions and prompt neutron emission as a function of incident neutron energy for major and minor actinides. In FY 2012, researchers measured the prompt neutron emission spectra from neutron-induced fission of 235U and 239Pu for incident neutron energies from 1 to 200 MeV. They analyzed the experimental data with the Los Alamos model for incident neutron energies of 1 to 8 MeV and carried out experiments to measure fission fragment mass yields from 238U as a function of incident neutron energy. The latter task employed a double Frisch-gridded ion chamber developed for this purpose.

Another project team is conducting detailed investigation of the properties of two different neutron detectors to measure prompt fission neutron spectra: a traditional NE213 detector and a p-terphenyl scintillator. During FY 2012, the Euratom team investigated properties of p-terphenyl scintillator detectors, preparing their application to investigate fission neutron spectra in the energy range up to 10 MeV. U.S. researchers investigated different symmetric reactions and targets using NE213 detectors for measurements in the energy range up to 20 MeV. The primary challenge is proper preparation of a lithium target for a reaction to test detector efficiency. Two targets have been identified, tested and dismissed; current efforts are focused on producing a lithium target on a nickel backing with nickel covering. Two other reactions provided more promising results.

Two projects are investigating advanced fuels for LWRs. One aims to advance the state of understanding of particle fuel technology by addressing technical questions related to fuel/target fabrication and performance. In FY 2012, the project team completed initial pressing and sintering studies and characterized porous uranium dioxide (UO_2) particles for open and interconnected porosity. The low-density particles have a weak microstructure and thus are well suited as a near-dustless substitute for UO_2 powder for LWR fuel pellet fabrication.

The second project supports the introduction of fully ceramic micro-encapsulated (FCM) fuel, which is believed to be robust and accident-tolerant. The project team has established the fuel assembly concept and its feasibility through compatibility analyses. Enhanced accident tolerance has been demonstrated through analyses of design-basis and beyond design-basis accidents. Researchers have manufactured and irradiated fuel samples and initiated testing and analysis.

The other two fuel-based projects are examining fuels for fast reactors. One team has cast several kilogram-scale batches of alloys to develop methods of minimizing losses and reducing waste streams during fabrication of metallic fuel pins for SFRs. The most promising samples will be compared with ternary alloys. Team members have also begun fabricating coated samples, which will be tested in a bench-scale casting system designed and fabricated in FY 2012. This system is now being installed in a minor- actinide capable glovebox.

The second fast reactor fuel project aims for improved understanding of the behavioral characteristics of minor actinide transmutation fuel types such as advanced mixed oxides, advanced metallic alloys, inert matrix fuels, and other ceramic fuels for fast neutron spectrum conditions. In FY 2012, researchers worked towards developing high spatial resolution instrumentation that will help determine thermal conductivity and mechanical properties. They used a focused ion beam (FIB) to prepare and perform electron backscatter diffraction (EBSD) studies on irradiated fuel samples in conjunction with computational modeling and simulation efforts. U.S. team members overcame significant challenges to transport FIB-prepared samples to the Institute for Transuranium Elements in Germany, supporting the planned joint characterization of irradiated silicon carbide.

The final FCR&D collaboration is designing electrochemical technology methods to recover zirconium from used nuclear fuel rods for more effective waste management. In FY 2012, team members generated experimental measurements of molten salts to provide key model parameters, which will be used to develop a 3D kinetic model for uranium–zirconium electrorefining. The project team also developed a codeposition computational model to study an electrolytic zirconium recovery system based on the process of competitive electrodeposition. To validate the computational model, the researchers conducted copper and nickel deposition experiments in a static cell and a rotating cylinder hull cell.

3.2 Program Activities

I-NERI programmatic accomplishments include successful completion of one U.S.–Republic of Korea (ROK) collaborative research project (2009-002-K, Enhanced Radiation Resistance through Interface Modification of Nano-Structured Steels for Gen IV In-Core Applications) and the initiation of a new project with Euratom (2012-001-E, High-Fidelity Thermal Hydraulic Fuel Assembly Simulations for Nuclear Reactors).

Key management of each I-NERI agreement takes place during an annual project performance review and bilateral program planning meeting (BINERIC). These annual meetings facilitate information exchange and provide opportunities for the parties to review ongoing projects, investigate new opportunities for collaboration, and update research scopes to support current R&D priorities.

The BINERIC meetings serve as a forum to discuss areas of mutual interest for future research. At each meeting, participants from the two collaborating countries establish general topics for joint research for the coming year, along with a schedule for requesting and evaluating proposals. Each respective I-NERI country coordinator sends out a request for proposals following the agreed guidance, and the submitted proposals are jointly evaluated for funding.

In FY 2012, DOE held a formal meeting with their Euratom colleagues in Brussels, Belgium, and hosted two meetings with the Republic of Korea in the United States. All ongoing projects were approved for continued funding at the most recent BINERIC meetings, as the research efforts are proceeding well and continue to address NE objectives and national priorities. NE requested a slight shift in scope for one U.S.–ROK project (2011-003-K, Verification and Validation of High-Fidelity Multi-Physics Simulation Codes for Advanced Nuclear Reactors). For the upcoming FY 2013 I-NERI solicitations for proposed joint research projects, both bilateral partnerships agreed to retain existing joint research topic areas, as summarized in the introductions to Sections 4.1 and 4.2.

In FY 2013, the United States will host the annual U.S.–Euratom BINERIC and project review.

Two U.S.–ROK meetings will take place: one BINERIC will occur via Video-conference in late spring, and the Republic of Korea will host a joint BINERIC/ project review in the fall.

DOE has also held recent discussions with Canada to identify priority areas for collaboration and plans to issue a solicitation for new U.S.–Canada projects.

As appropriate, DOE will continue to pursue new cooperative agreements with prospective partner countries.

These international collaborations have forged lasting ties that will continue promoting the strong infrastructure necessary to overcome future challenges to the expanded use of nuclear energy. The resulting technological and scientific advances are responding to the need for economical and environmentally conscious sources of energy. I-NERI's goals and objectives continue to be satisfied.

4 Project Summaries and Abstracts

Glovebox
Photo courtesy of Idaho National Laboratory

4.1 United States–Euratom Collaborative Projects

DOE and Euratom, an international organization composed of the members of the European Union, 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.

Research areas of interest to both the United States and Euratom include advanced reactor concepts and associated fundamental nuclear science; existing plant life extension, integrity of components, and performance optimization; waste transmutation and management; reactor safety; severe accident management; advanced fuels; nuclear medicine and radioprotection; and uranium programs. In addition, both countries have modeling and simulation programs that can be expanded and employed in pursuit of research objectives.

In FY 2012, efforts continued on seven ongoing projects, and a new project was initiated. One project team is conducting detailed investigation of the properties of two different neutron detectors to measure prompt fission neutron spectra, while two other projects are producing data on prompt fission neutron multiplicity and spectra, which will help address reactor safety and waste management issues. Two are investigating advanced nuclear fuels; in one case, the project team is characterizing minor actinide-bearing transmutation fuels, and in the other, the team is investigating fabrication and performance of particle fuel and targets. Two project teams are contributing to materials studies. One collaboration is researching irradiation and corrosion effects in materials used for innovative reactor systems. The second is investigating the viability of using standards-compliant schemas and ontologies to address interoperability of materials test databases, as materials test data currently differ in format and associated semantics, impeding information exchange between partners.

Below is a listing of current I-NERI U.S.–Euratom projects, followed by summaries of their FY 2012 accomplishments and an abstract of the new project.

2010-001-E	Measurements of Fission Fragment Mass Distributions and Prompt Neutron Emission as a Function of Incident Neutron Energy for Major and Minor Actinides
2010-002-E	Spherical Particle Technology Research for Advanced Nuclear Fuel/Target Applications
2010-003-E	Irradiation and Testing of Advanced Oxide Dispersion-Strengthened and Ferritic–Martensitic Steels
2010-004-E	Development of a Standard Neutron Detector for the Energy Range up to 20 MeV and Its Application
2010-005-E	Interoperability of Material Databases
2010-006-E	State-of-the-Art Post-Irradiation Examination of Advanced Nuclear Fuels
2011-001-E	Development of a 2E-2V Instrument for Fission Fragment Research
2012-001-E	High-Fidelity Thermal Hydraulic Fuel Assembly Simulations for Nuclear Reactors

Measurements of Fission Fragment Mass Distributions and Prompt Neutron Emission as a Function of Incident Neutron Energy for Major and Minor Actinides

Project Number: 2010-001-E

PI (U.S.): Robert Haight, Los Alamos National Laboratory

PI (Euratom): Franz-Josef Hambsch, Joint Research Centre–Institute for Reference Materials and Measurements

Collaborators: Commissariat à l'énergie atomique et aux énergies alternatives, Centre DAM Bruyeres-le-Chatel

Program Area: FCR&D

Project Start Date: November 2009

Project End Date: October 2012

Research Objectives

The project is producing new and essential experimental data on fission fragment mass distributions and prompt neutron emission as a function of incident neutron energy for major and minor actinides. Improved fission fragment mass distributions, prompt fission neutron multiplicity, and spectral data are needed to optimize the design and safety assessment of fast reactors, accelerator-driven systems, and waste management scenarios. Researchers are measuring fission fragment mass distributions and producing accurate data files for the relevant isotopes as a function of incident neutron energy using neutron facilities at the Los Alamos National Laboratory (LANL) and the Institute for Reference Materials and Measurements (IRMM), together with advanced detectors being developed in these laboratories and at the Commissariat à l'énergie atomique (CEA), Centre DAM Bruyères-le-Châtel laboratory. Results will contribute to the ENDF and JEFF libraries.

Research Progress

Fission Neutron Emission

The research team measured the prompt neutron emission spectra from neutron-induced fission of 235U and 239Pu for incident neutron energies from 1 to 200 MeV at the Los Alamos Neutron Science Center (LANSCE) Weapons Neutron Research (WNR) facility. They analyzed the experimental data with the Los Alamos model for incident neutron energies of 1 to 8 MeV, using a CEA multiple-foil fission chamber containing deposits of 100 mg 235U and 90 mg 239Pu to detect fission events. Outgoing neutrons were detected by the FIGARO array of 20 liquid organic scintillators. The researchers used a double time-of-flight technique to deduce the incident neutron energies from the spallation target and the outgoing energies from the fission chamber. These data were used to test the Los Alamos model, and the total kinetic energy (TKE) parameters were optimized to obtain a best fit to the data. The prompt fission neutron spectra were also compared with the evaluated data in ENDF/B-VII.0. The researchers calculated average energies from both experimental and calculated fission neutron spectra. Figure 1 provides sample data.

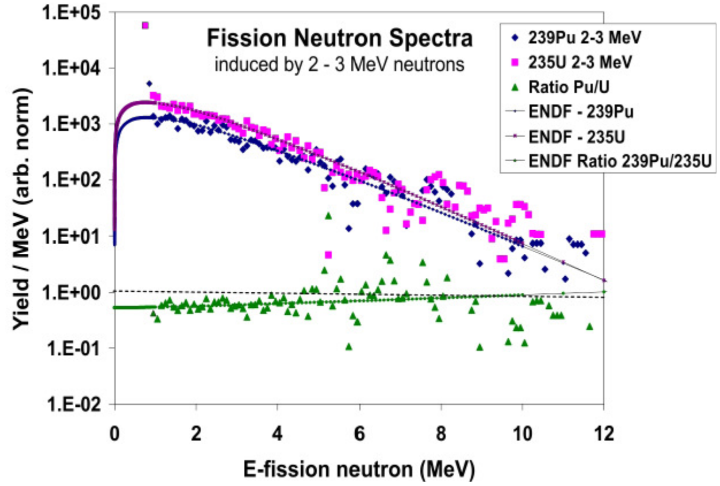


Figure 1. Prompt fission neutron spectra for fission induced by neutrons of 2–3 MeV on ^{235}U (red squares) and ^{239}Pu (blue diamonds), both arbitrarily normalized. The ratio is shown as green triangles. The data are compared with the ENDF/B-VII.0 evaluated data.

Fission Fragment Mass Yields

Also at LANSCE, the project team carried out experiments to measure fission fragment mass yields from 238U as a function of incident neutron energy. The IRMM team developed a double Frisch-gridded ion chamber for this purpose. The ionization chamber cathode is a thin metalized polyimide film onto which the actinide has been uniformly deposited with a low areal density (< 200 µg/cm2). The ionization chamber is used to measure in coincidence the kinetic energies and angles of emergence of both fission fragments. Researchers determined fragment mass distribution through mass and momentum conservation, taking into account corrections for prompt neutron emission, energy loss in sample/ substrate and pulse height defect.

The project team conducted experiments on fission yields in FY 2009 and FY 2010, but the event rates were too low in those experiments to finalize the data analysis. Preliminary results from those experiments are shown in Figure 2. For FY 2012, researchers altered the experiment, optimizing to obtain statistics to improve the data analysis. The revisions entailed changing the neutron beam collimation from 1 to 2 centimeters in diameter and moving the chamber closer to the neutron source. Researchers also removed material that scatters neutrons away from the detector, reducing the background event rate and thus further improving the analysis. The project team collected five weeks of data in late 2012; the experiments continue in 2013.

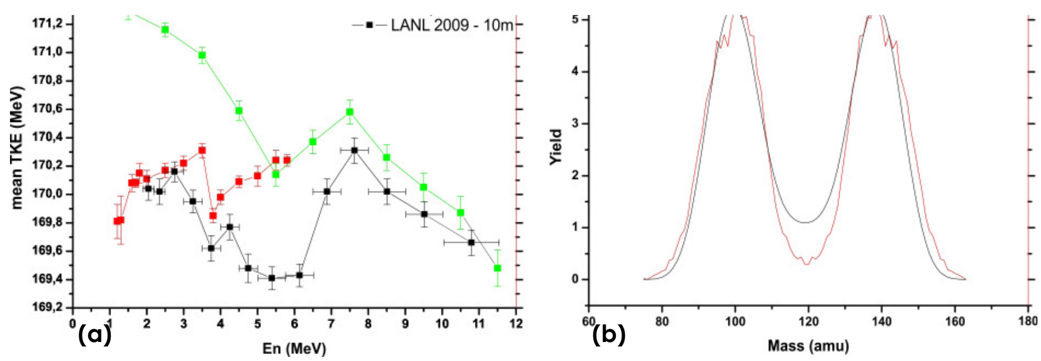


Figure 2. (a) Preliminary mean TKE distribution in $^{238}U(n,f)$ reaction as a function of incident neutron energy compared to the results obtained by Zoller and Vives; (b) pre-neutron mass distribution at 10.79 MeV compared to TALYS (Euratom software) code result.

Planned Activities

Although this I-NERI project has concluded, scientists plan to continue further measurements of prompt neutron emission from neutron-induced fission with a new flight path at WNR, parallel-plate avalanche fission counters made at Lawrence Livermore National Laboratory, and new detector arrays for high-energy neutrons (> 0.6 MeV) and lower-energy neutrons (50 keV–1 MeV). By mid-2013, the research team will commission a new data acquisition system based on waveform digitizers, and production measurements will begin shortly thereafter.

The analysis from the 238U(n,f) experiments is ongoing. The higher statistics data collected in FY 2012 are expected to significantly improve the data analysis process, which in turn should allow for a detailed study of the change in fission mass yields as a function of incident neutron energy. Researchers foresee taking new measurements with different actinides, such as 235U or 239Pu, and the preliminary plan is to start these measurements in the second half of FY 2013.

Photo courtesy of Public Domain

Spherical Particle Technology Research for Advanced Nuclear Fuel/ Target Applications

Project Number: 2010-002-E

PI (U.S.): Stewart Voit, Oak Ridge National Laboratory

PI (Euratom): Joseph Somers, Joint Research Center–Institute for Transuranium Elements

Collaborators: Los Alamos National Laboratory

Program Area: FCR&D

Project Start Date: February 2010

Project End Date: January 2013

Research Objectives

The goal of this work is to advance the state of understanding of particle fuel technology by addressing technical questions related to fuel/target fabrication and performance. A key objective is the evaluation of spherical particles for use as (1) a potential near-dustless feed for ceramic fuel fabrication and (2) a host matrix for the infiltration of a second actinide phase. The actinide-bearing kernels may be dispersed in either a ceramic or metal matrix, forming a ceramic–ceramic or ceramic–metallic composite.

In FY 2012, the project team completed initial pressing and sintering studies and characterized porous UO_2 particles for open and interconnected porosity. The low-density particles have a weak microstructure and thus are well-suited as a near dustless substitute for UO_2 powder for light water reactor (LWR fuel pellet fabrication.

Research Progress

The process by which UO_3 is reduced to UO_2 affects the development of pore structure in the spherical particles. This relationship is illustrated by the mercury porosimetry intrusion curves displayed in Figure 1. This graph shows a semi-logarithmic plot of the normalized volume of mercury which has penetrated into the particle pore network as a function of applied pressure. Results from three UO_2 samples reduced under different conditions are shown for comparison.

Almost no mercury is intruded into the particles treated in the highest temperature range (700°C air followed by 700°C Ar-H_2). The particle density from this batch is very low; thus the porosity at the exterior surface has closed, and the mercury forms an envelope around the particles but does not penetrate. The other samples—those processed in air at 550°C followed by Ar-H_2 at 475°C and those processed in Ar-H_2 at 475°C—have an open network of pores, and mercury intrudes into over 50 percent of the particle volume. Although these two batches treated at lower temperatures have approximately the same amount

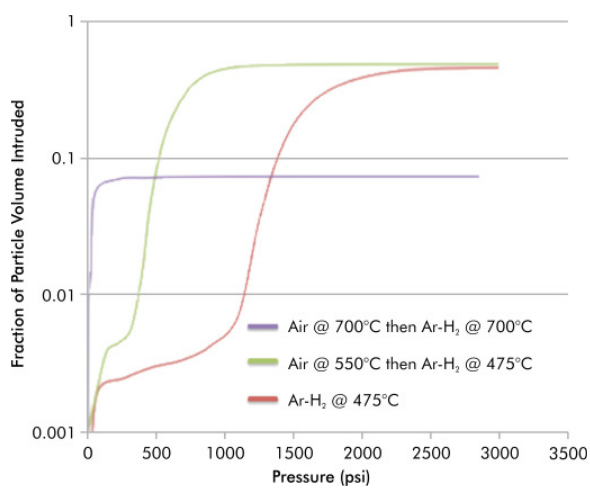


Figure 1. UO_2 spherical particle porosity via mercury porosimetry.

of open porosity, there is a clear difference in the pore size, as indicated by the pressure required to intrude mercury into the same volume.

The research team conducted a pressing study to compare the compressibility of low-density UO_2 particles with differing pore characteristics. Two sol-gel batches were chosen for comparison, as shown in the stress–strain curves for particles and powder pressed at 160 MPa (see Figure 2). The results were contrasted with a reference UO_2 powder from Areva with demonstrated good powder processing behavior.

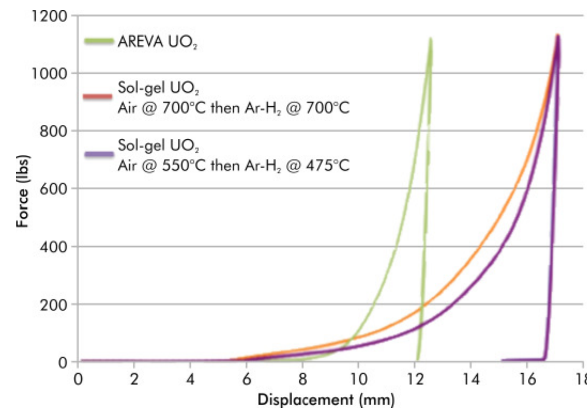


Figure 2. Force versus displacement curves from UO_2 spherical particle pressing study.

The initial packing density in the die for low-density spherical sol-gel particles with diameters of ~550 μm is much less than for the reference powder that has a mean agglomerate size of < 100 μm. As a result, during the compression stage for the spherical particles, the packing density increases dramatically as the weak particle microstructure begins to fracture and fill intra- and interparticle void space. Researchers also observed a difference in the stress–strain curve for the low-fired and moderate-fired spherical particles. Particles reduced to UO_2 at higher temperatures have less porosity (as seen in Figure 1) and require a greater pressing force for the punch to be displaced an equivalent distance.

Select green (unsintered) pellets from the pressing study were fired at high temperature in a dilatometer to investigate the densification behavior of spherical particles versus reference powder. All pellets were sintered in gettered argon gas at a heating rate of 5°C per minute up to 1550°C with an isothermal hold time of 4 hours. Although the spherical particles are nominally 550 μm in diameter, the fundamental grain size within the particles is less than 100 nm, as the project team determined in FY 2011 using scanning electron microscopy. The smaller size is due to the low temperatures at which transformation from amorphous UO_3 and crystallization of UO_2 grains occurs, resulting in very slow kinetics for grain coarsening and minimal growth during the short heat treatment time. The high reactivity of nanosized grains is derived from the tendency to reduce excessively large surface area by decreasing the radius of curvature through the formation of "necks" at the interfaces between grains. This leads to a lower temperature for onset of densification as compared to microsized grains, as can be seen in Figure 3. The low-fired particles have the smallest grain size and thus the lowest onset sintering temperature. The rate of densification is also greater for spherical particles with nanosized grains as compared to microsized UO_2 powder. The fractional change in length shown in Figure 3 is greater for the spherical particles; this is a reflection of the high reactivity combined with the low starting density of the green compacts.

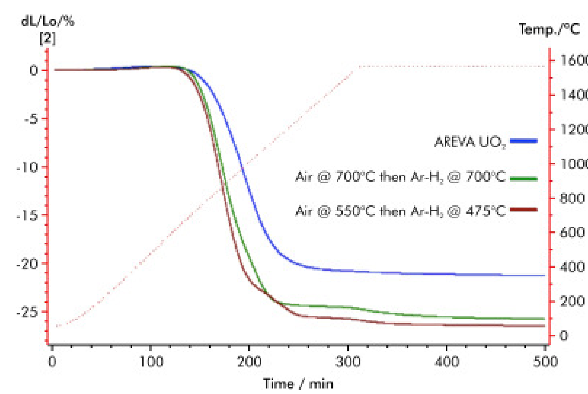


Figure 3. Dilatometer response curves for comparison of different UO_2 materials.

Planned Activities

Sol-gel-derived UO_3 particles were reduced to UO_2 under varied conditions. Results showed that it is possible to fabricate low-density spherical particles with open and interconnected porosity. Compression and densification studies have been performed, and although the processing parameters have not been optimized, the pressing and sintering behavior of porous UO_2 particles compares reasonably well with a reference UO_2 powder. The next step is to modify the particle microstructure to improve compaction properties and assess the degree to which a representative pellet microstructure can be achieved.

The porous particles may also be used as a host matrix for infiltration of a second oxide phase. This concept has potential application as a transmutation target form and, like a uranium–americium target, the process is particularly well-suited for remote fabrication with high-activity materials. The research team will also perform infiltration studies to achieve a fundamental understanding of the sol-gel microsphere calcination and infiltration process and to assess the range of second-phase loading.

Irradiation and Testing of Advanced Oxide Dispersion-Strengthened and Ferritic–Martensitic Steels

Project Number: 2010-003-E

PI (U.S.): Stuart Maloy, Los Alamos National Laboratory

PI (Germany): Concetta Fazio, Karlsruhe Institut für Technologie (formerly Forschungzentrum Karlsruhe)

Collaborators: Idaho National Laboratory, Lawrence Livermore National Laboratory, Oak Ridge National Laboratory, Pacific Northwest National Laboratory, Paul Scherrer Institute, SCK•CEN (Belgian Nuclear Research Center)

Program Area: FCR&D

Project Start Date: December 2009

Project End Date: December 2013

Research Objectives

This collaboration is investigating irradiation and corrosion effects in materials used for innovative reactor systems, including fast reactors cooled with sodium, heavy liquid metal (HLM), and gas. The project team is developing and characterizing advanced oxide dispersion-strengthened (ODS) and ferritic–martensitic (F/M) steels, investigating corrosion effects of these steels in sodium, and examining the ODS steels' irradiation behavior. They are utilizing several irradiation programs, such as those conducted at MEGAPIE and the Fast Flux Test Facility (FFTF), to prepare materials for post-irradiation analysis and gather data for modeling.

Sharing collaborative research results among scientists at U.S. national laboratories, the U.S. Department of Energy, and Euratom's GETMAT (Generation IV and Transmutation Materials) will improve understanding of these innovative systems' potential viability and competitiveness, support development of improved safety features, and provide information about transmutation systems using HLM coolant and a fast-spectrum spallation neutron flux.

Research Progress

Development of Advanced ODS Steels (including welds)

The U.S. team produced a 50-kilogram heat of the mechanically alloyed (MA) ODS ferritic alloy14YWT, and recently completed 40-hour milling on this heat of material through a subcontract with Zoz GmbH in Germany. An elemental scan across the particle is shown in Figure 1. The U.S. team shared these data with GETMAT research scientists, making similar progress on development of 12Cr and 9Cr ODS alloys.

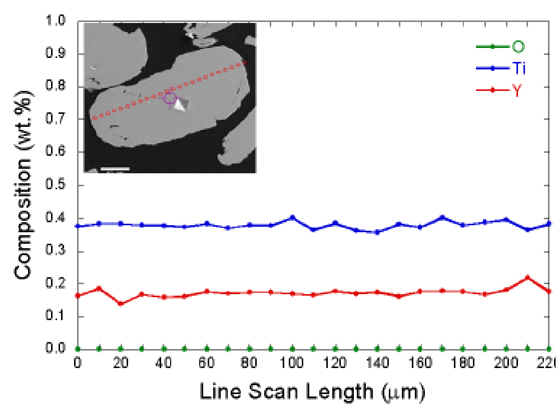


Figure 1. Graph showing results of a scan of oxygen, titanium and yttrium concentration across a 40-hour-milled powder of 14YWT; indicates uniform elemental distribution after milling.

Advanced Materials Coolant Compatibility

Los Alamos National Laboratory (LANL) has restarted its DELTA lead–bismuth loop test facility and is planning a 3,000-hour corrosion test campaign, which will start in FY 2013. Simultaneously, GETMAT researchers are conducting long-term (up to 10,000 hours) corrosion testing on ODS alloys in lead and lead–bismuth.

Irradiation Behavior of Advanced ODS Steels

U.S. researchers are conducting more detailed transmission electron microscope (TEM) and atom probe analyses of ODS steel alloy MA-957 after irradiation to 100 dpa in the FFTF. In addition, the research team is testing ODS steels irradiated through the Swiss spallation neutron source target irradiation program, STIP IV. Euratom researchers will obtain high-dose irradiation data from materials irradiated in the MATRIX program, which reproduces the fast spectrum conditions for core components. However, this activity is presently delayed because of hot cell repairs.

Information Sharing

Significant data sharing progress has been made through collaborative meetings. In particular, the U.S. principal investigator (PI) presented highlights of the initial results of MA-957 testing and progress in the large scale 14YWT powder production project at the GETMAT technical meeting in Gent, Belgium, January 2012.

Multiscale Modeling

U.S. researchers are developing creep models of F/M iron–chromium alloys and validating those models through experimental results. Although this activity was initially sponsored through the NEAMS program, it will no longer be pursued under the scope of I-NERI, as funding is discontinued in FY 2013.

Planned Activities

The U.S. and Euratom research teams will exchange specimens of ODS alloys for further study and will continue attending technical meetings to share results. The final GETMAT meeting is planned for September 2013, during which the teams will exchange data and discuss future collaborations.

Development of a Standard Neutron Detector for the Energy Range up to 20 MeV and Its Application

Project Number: 2010-004-E

PI (U.S.): Nikolay Kornilov, Ohio University

PI (Euratom): Franz-Josef Hambsch, Joint Research Centre–Institute for Reference Materials and Measurements

Collaborators: Idaho State University, Los Alamos National Laboratory, National Institute of Standards and Technology

Program Area: FCR&D

Project Start Date: November 2010

Project End Date: October 2013

Research Objectives

The objective of this project is to develop a standard neutron detector (SND) with an accurately measured and modeled response function to improve the accuracy of both prompt fission neutron spectra and (n,p) angular distribution in the neutron energy range up to 20 million electron volts (MeV). To begin this effort, the project team is conducting detailed investigation of the properties of two different neutron detectors to measure prompt fission neutron spectra—a p-terphenyl scintillator and symmetric reactions/targets with a "traditional" NE213 detector—that are believed have high accuracy in the neutron energy ranges to 10 MeV and 20 MeV, respectively.

Based on their initial results, as noted in the FY 2011 report, the research team modified their initial plan to base the SND design around a p-terphenyl scintillator after detailed investigation revealed excessive variation in energy dependences of the light output, a critical scintillator characteristic. Additional research has uncovered that the properties of available p-terphenyl scintillator detectors are not greatly superior to those of traditional NE213 detectors. Therefore, the U.S. research team is conducting further studies using an NE213, while the Euratom team will continue investigating p-terphenyl.

Research Progress

During 2012, the Euratom team investigated properties of

p-terphenyl scintillator detectors, preparing their application to investigate fission neutron spectra in the energy range up to 10 MeV. U.S. researchers investigated different symmetric reactions and targets using NE213 detectors for measurements in the energy range up to 20 MeV.

Joint Research Centre

The research team generated light output curves to compare detectors made of different materials and as inputs into Monte Carlo simulations. The light output function for protons is constructed by determining the position of the most energetic protons for a given neutron energy. The position of the most energetic protons can be determined by identifying the high-energy minimum of the response function's first derivative (see Figure 1). In an ideal detector, this position is indicated by a hard edge; as detectors have finite resolution, the edge is smeared.

Detector efficiency is determined by comparing the experimental neutron energy spectrum to the Mannhart evaluation.[1] The efficiency for 2 MeV neutrons, with a 100 keV$_{33}$ (electron equivalent) threshold, is shown in Figure 2. While the liquid scintillator detectors (LS301), shown as blue points, indicate remarkably similar efficiencies for the same cell thickness, the p-terphenyl detectors display a wide range of efficiencies. This discrepancy might be due to a deficient manufacturing process.

Researchers have started measuring the prompt fission neutron spectrum (PFNS) for 252Cf(SF) using an ionization chamber to detect fission fragments and several of their own neutron detectors to measure the PFNS, with data acquisition based on a digital system. The team is also using segmented anodes to determine the Phi angle of the fission fragment emissions. Team members are now analyzing the data from several million events. Figure 3 shows a comparison of typical mass distributions from an early analysis versus those in the literature. Figure 4 shows an early PFNS analysis.

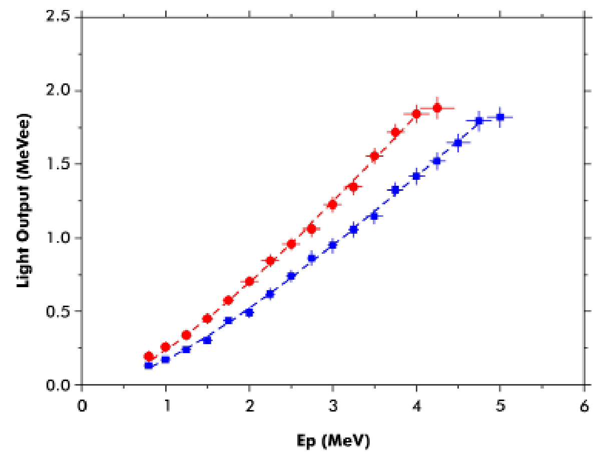


Figure 1. Light output function for the p-terphenyl detector pth185 (full circles) and the SCIONIX LS301 det2 (full squares).

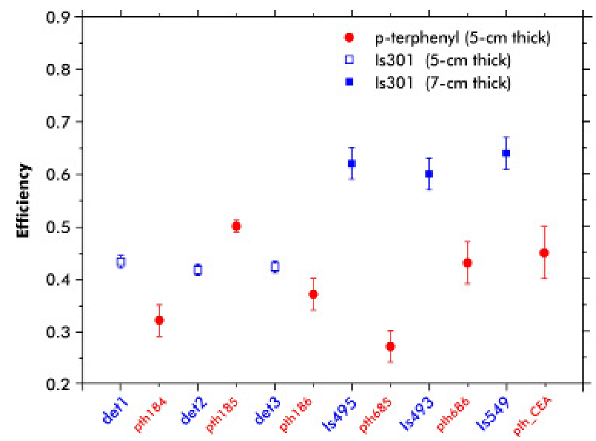


Figure 2. Relative efficiency for p-terphenyl and liquid scintillator detectors.

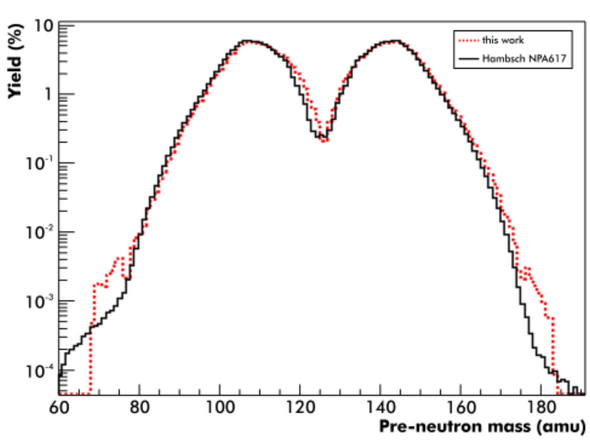


Figure 3. Pre-neutron fragment mass yield obtained in this work and by Hambsch et al.

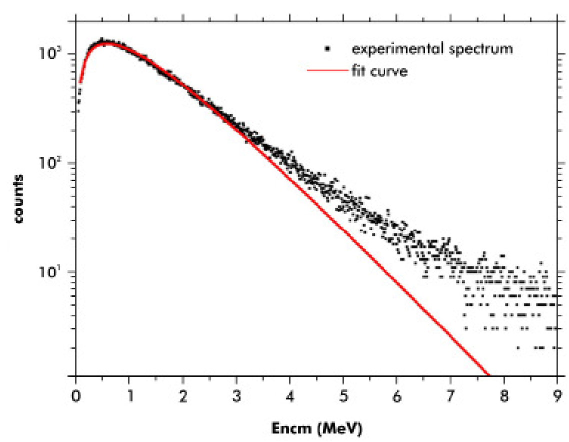


Figure 4. Neutron evaporation in the center-of-mass system.

1 F.-J. Hambsch, S. Oberstedt, *Nucl. Phys. A,* 617, 347 (1997).

The final goal is to set up a SCINTIA neutron detector array at the GELINA time-of-flight (TOF) white neutron source facility. The array would use all available neutron detectors to measure prompt neutron emissions in resonance-induced fission of ^{235}U and ^{239}Pu

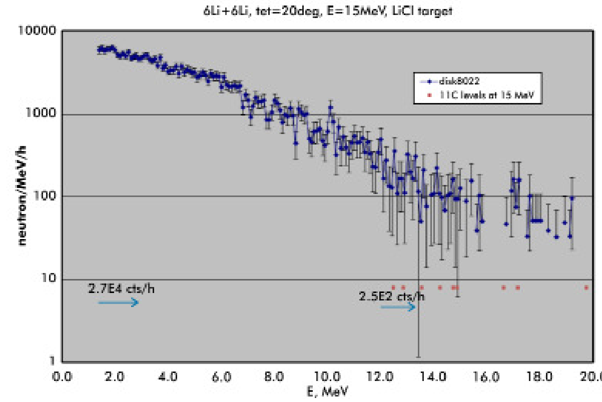


Figure 5. Neutron spectrum from a LiCl target at a 20° angle. The count rate is given for different energy intervals. Red points indicate levels from the $^6Li(^6Li,n)$ reaction.

Ohio University

Because of the symmetry of Y(Y,n) neutron production reactions, the neutron yield in the center-of-mass system is the same for symmetric forward and backward angles. The laboratory system, however, produces different neutron energies, which allows researchers to estimate detector efficiency at high energy if the low-energy efficiency is known. The best candidate reaction for this task is $^6Li(^6Li,n)$. The primary challenge is proper preparation of the lithium target, which must be protected from oxygen and other atoms that may induce neutrons from the target. Having investigated several metal combinations, the research team identified two targets as the most stable: ^6LiF (thickness d = 710 μg/cm^2) and ^6LiCl (thickness d = 10,000 Å on gold backing with 1000 Å gold covering).

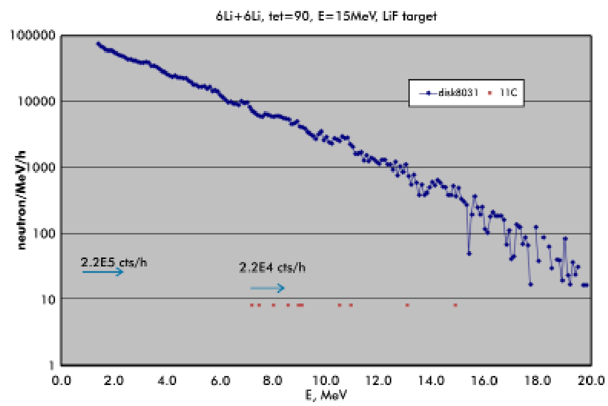


Figure 6. Neutron spectrum from a LiF target at 20°.

Both targets gave negative results (see Figures 5 and 6), as the neutron yield from separate levels was much lower than the neutron background. These reactions cannot be used to measure detector efficiency.

Current efforts are focused on producing a lithium target on a nickel backing with nickel covering. U.S. researchers also investigated the reaction of $^{12}C(^{12}C,n)$ (Q = -2.6 MeV), which exhibited good separated levels in the TOF spectrum. Due to the negative Q value, however, the highest available neutron energy is only ~8 MeV, while the minimum energy at the university accelerator is about 18 MeV with ^{12}C ions. Because of this positive result, the researchers plan to conduct future experiments on the reaction of $^{13}C(^{13}C,n)$ (Q = 11.37 MeV). The next step is to prepare a ^{13}C target.

The most promising result was reached with a deuterium polyethylene (CD_2) target and D(d,n) reaction with a target thickness of 1 mg/cm^2. The CD_2 target has obvious advantages over a metal TiD target: smaller energy thickness with the same amount of deuterium, and smaller energy and angle straggling.

Figures 7 and 8 show the TOF spectra for two different angles measured with an NE213 detector on flight path 4.08 m, channel width 0.21 ns, for deuteron energy 8.01 MeV. Neutron energy from the D(d,n) reaction is 11.16 MeV at 0° angle

and 2.77 MeV at a 120° angle. Researchers concluded that this reaction would enable measuring the NE213 detector efficiency in the energy range of 2.6–12 MeV. This energy interval is large enough that scientists could investigate the (n,p) scattering angular distribution by detecting input neutron energy of 14.5 MeV.

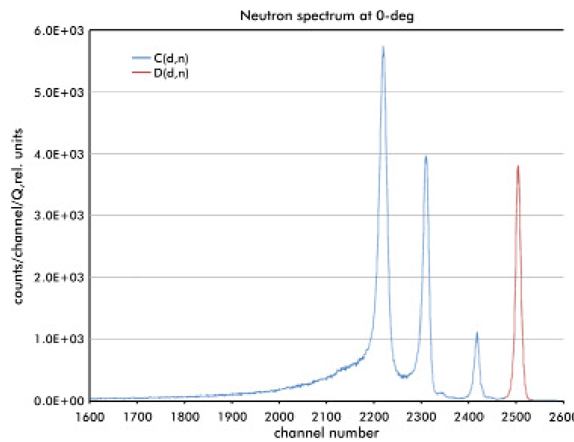


Figure 7. Neutron spectrum from CD_2 target at a 0° angle. Blue lines indicate levels from the 12C(d,n) reaction, and red lines indicate levels from the D(d,n) reaction.

Planned Activities

The Euratom research team will continue measurements and analyses of the 252Cf(SF) PFNS and fission fragment correlations. The digital data acquisition system will be further developed to accommodate about 15 to 20 neutron detectors, and the team will start building the SCINTIA neutron detector array at GELINA.

The U.S. team will use a CD_2 target and D(d,n) reaction to measure NE213 detector efficiency. This task includes development of a data analysis method. Team members will also investigate the symmetric reaction of 13C(13C,n) (Q = 11.37 MeV) to measure neutron detector efficiency up to 20MeV energy on the Ohio University tandem accelerator. Researchers will initiate experiments to study (n,p) scattering.

As a follow-on activity to this I-NERI project, the researchers plan to participate in an IAEA coordinated research project (CRP) that will develop a new type of detector to investigate standard neutron fields on the basis of Al(d,n), Be(d,n), B(d,n) reactions for neutron energies < 20 MeV.

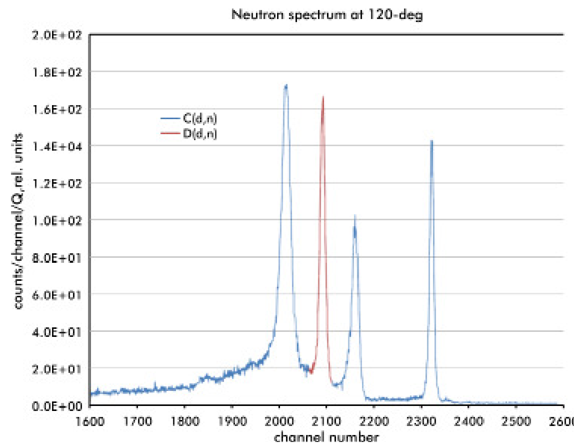


Figure 8. Neutron spectrum from CD_2 target at a 120° angle.

Interoperability of Material Databases

Project Number: 2010-005-E

PI (U.S.): Lianshan Lin and Weiju Ren, Oak Ridge National Laboratory

PI (Euratom): Timothy Austin, Peter Hähner, Hans-Helmut Over, Joint Research Centre–Institute for Energy and Transport

Collaborators: None

Program Area: Reactor Concepts RD&D

Project Start Date: January 2011

Project End Date: December 2013

Research Objectives

This project is investigating the viability of using standards-compliant schemas and ontologies to address interoperability of materials databases, facilitating data exchange between research partners. As numerous components in Gen IV systems are exposed to high temperatures, neutron fluences, and corrosive environments, safe and economic system operations necessitate extended materials qualification testing. However, existing materials test databases currently present compatibility challenges: they differ in format and associated semantics; they are stored in both heterogeneous and distributed database repositories; these repositories have a variety of working environments and software tools to access the data; and these environments and tools are in a constant state of change.

The main goal of the project is the implementation of web-service software for data import and export using an agreed standardized schema at Oak Ridge National Laboratory (ORNL) and the Joint Research Centre (JRC). Database interoperability will reduce costs associated with redundant materials testing programs, promote long-term data preservation, enable improved auditing traceability, and support the reuse of data.

The work is undertaken within the broader scope of international efforts to develop technologies that enable interoperability of materials databases. The present project builds on the results of a recently completed European Committee for Standardization (CEN) workshop on the viability of standards-compliant data formats, starting with a tensile test schema derived from documentary technology standards and a guide for using and developing data formats for engineering materials test data.1 The introduction and adoption of standards-compliant data formats (schemas and ontologies that are a faithful representation of procedural standards for mechanical testing) will allow researchers to leverage established and emerging web-based technologies for data storage and retrieval.

1 CWA 16200:2010 (2010), "A Guide to the Development and Use of Standards-Compliant Data Formats for Engineering Materials Test Data," retrieved February 27, 2012, from *ftp://ftp.cen.eu/CEN/Sectors/List/ICT/CWAs/CWA16200_2010_ELSSI.pdf.2 Ibid*.

Research Progress

In the first project year, the collaborators thoroughly discussed and verified materials database interoperability between the Gen IV Materials Handbook (U.S. database) and MatDB (EC JRC database). In a preliminary study, researchers extracted and compared the tensile test schemas from both databases, identifying numerous technical details of interest. Both database architectures were also studied. Despite extreme differences between these two databases, the preliminary comparison and analysis suggests that massive data exchange between the two databases is possible.

In FY 2012, U.S. team members updated the tensile test schema in the Gen IV Materials Handbook database (see Figure 1). Based on the updated schema, they generated new database schemas for materials, tensile, creep test, and low-cycle fatigue (LCF) tests (see Figure 2).

C1-Tensile.xsd

```
<?xml version="1.0" encoding="utf-8"?>
<xs:schema elementFormDefault="qualified" xmlns:xs="http://www.w3.org/2001/XMLSchema">
<xs:element name="C1Tensile">
<xs:complexType>

  <xs:element name="GeneralInformation">

  <xs:element name="Specimen">

  <xs:element name="PretestSpecimenConditioning">

  <xs:element name="TestingConditions">

  <xs:element name="RawData">

  <xs:element name="CalculatedParameters">

  <xs:element name="FailureInformation">

  <xs:element name="TestDataNotes">

  <xs:element name="RelatedRecords">

  <xs:element name="RecordManagementInformation">

  <!-- for RawData -->
    <xs:element name="Time">

    <xs:element name="Load">

    <xs:element name="Displacement">

    <xs:element name="Strain">

    <xs:element name="Stress">

</xs:complexType>
</xs:element>
</xs:schema>
```

Figure 1. Tensile test schema in the Gen IV Materials Handbook database.

A-Materials.xsd

```
<?xml version="1.0" encoding="utf-8"?>
<xs:schema elementFormDefault="qualified" xmlns:xs
<xs:element name="AMaterials">
<xs:complexType>
    <xs:element name="GeneralInformation">

    <xs:element name="ChemicalComposition">

    <xs:element name="PhysicalProperties">

    <xs:element name="ThermalProperties">

    <xs:element name="ElectricalProperties">

    <xs:element name="QualitativeEnvironmentalResi

    <xs:element name="MaterialBiography">

    <xs:element name="AvailableProductForms">

    <xs:element name="ApplicableSpecifications">

    <xs:element name="ApplicableCodesandStandards"

    <xs:element name="MaterialRecordNotes">
</xs:complexType>
</xs:element>
</xs:schema>
```

Figure 2. Materials schema that is representative of the tensile, creep test, and LCF test schemas.

The EC JRC team is also generating updated schemas for MatDB. As the two sets of schemas must be sufficiently compatible to enable data exchange, the research teams have mapped specific attributes between schemas and studied the results. The teams have identified, one by one, the common attributes between the two databases for the four schemas noted above.

These schema files only define the data structure to be extracted from the database; they do not actually contain any data. As part of the data exchange between the Handbook and MatDB databases, three actual XML data samples for each schema have been extracted from the Handbook database. One of the XML data sample files is illustrated in Figure 3.

Due to structural changes in both the Handbook and MatDB, ORNL and JRC development teams are examining the utility of an existing standard test schema2 derived from the ISO-6892 standard for ambient temperature tensile testing. This schema provides a more consistent framework that would support data exchange between the two databases, enabling the two teams to independently map their local data formats to a common structure/messaging format. Figure 4 provides a comparison of ISO-6892-1 and Handbook tensile test schemas.

C1-Tensile-Inconel617_0023_0001.xml


```xml
<?xml version="1.0" encoding="utf-8"?>
<C1Tensile xmlns:xsi="http://www.w3.org/2001/XMLSchema" xsi:schemaLocation="C1-Tensile.xsd">
    <GeneralInformation>
        <DataProjectNameOrID>US High Temperature Gas-Cooled Reactor Program</DataProjectNameOrID>
        <TestingOrganization>Oak Ridge National Laboratory</TestingOrganization>
        <HowDidTestEnd>Ruptured</HowDidTestEnd>
        <DataProcessedBy>Oak Ridge National Laboratory</DataProcessedBy>
    </GeneralInformation>

    <SpecimenInformation>
        <MaterialTradeName>Inconel 617</MaterialTradeName>
        <BatchOrHeatNumber>XX01A3US</BatchOrHeatNumber>
        <ProductForm>Plate</ProductForm>
        <Product>
            <ProductThickness unit="in">0.512</ProductThickness>
        </Product>
        <BaseorWeldMaterial>Base Material</BaseorWeldMaterial>
        <SpecimenCuttingOrientation>Longitudinal</SpecimenCuttingOrientation>
        <GageLength unit="in">1.25</GageLength>
    </SpecimenInformation>

    <PretestSpecimenConditioning>
        <PedigreeMaterialHeatTreatment>Solution annealed</PedigreeMaterialHeatTreatment>
    </PretestSpecimenConditioning>

    <TestingConditions>
        <LoadingDirection>Tension</LoadingDirection>
        <TestControlMode>Rate of Separation of Heads</TestControlMode>
        <TestTemperature unit="°F">73.4</TestTemperature>
        <TestTemperatureNotes>The test room temperature was reported as 22°C in the source docume
        <TestEnvironment>Air</TestEnvironment>
    </TestingConditions>
```


Figure 3. Tensile test data extracted from the Handbook database.

1	ISO-6892			ORNL	
2	uniaxial			GeneralInformation	
3		*testing_standard*			TestStandards
4		*test_category*			
5		*material_category*			
6					
7	conditions			specimen	
8		extensometer_gauge_length			Gagelength
9		nominal_temperature		TestingConditions	
10		test_phase			TestTemperature
11		placement			
12		preliminary_force			
13		measurement_systems			
14		remarks			
15					
16	series				
17		point			
18			time		time
19			gauge_length		
20			elongation		displacement

Figure 4. Tensile test schema comparison between ISO-6892-1 and the Handbook.

Planned Activities

The development teams will use the comparisons of schemas to build up those with common elements between the two databases. The exchange interface will first be applied to the tensile test schema and then migrated to other schemas. As a successful data exchange requires a certain number of common elements and attributes, the researchers may have to revise the original data structures of one or both databases. Researchers will do the least revision necessary for successful exchange.

The Handbook system provides a convenient and efficient tool to export existing data records into external XML files. U.S. researchers will utilize this tool and the standard schema interface between MatDB and the Handbook to extract massive data records. The difficulties in this procedure involve reorganizing the data into the appropriate format according to the interface schemas.

2 Ibid.

The ORNL and JRC teams will cooperate to exchange the massive data records between MatDB and the Handbook. The data exchange entails a two-way communication process, with an intermediate step of exporting data into XML files before importing to the other database, and vice versa. The initial exchange will be data from the "Generation IV International Forum Very High-Temperature Reactor System Materials Project Plan." Further exchanges will be discussed and approved by DOE and JRC management.

CEN recently commissioned another workshop on standards for representing and reporting engineering materials data electronically. The project researchers plan to contribute to this workshop by mapping local data structures to provide the CEN workshop participants with valuable insight into the requirements for a standard representation of engineering material.

State-of-the-Art Post-Irradiation Examination of Advanced Nuclear Fuels

Project Number: 2010-006-E

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

PI (Euratom): V. V. Rondinella, Joint Research Centre–Institute for Transuranium Elements

Collaborators: Colorado School of Mines, Los Alamos National Laboratory, Massachusetts Institute of Technology, University of Central Florida

Program Area: FCR&D

Start Date: January 2011

End Date: December 2014

Research Objectives

Safe and effective implementation of new closed nuclear fuel cycle concepts employing fast reactors with advanced fuel forms necessitates improved understanding of the behavioral characteristics of the proposed advanced fuels. The fuel systems under consideration include minor actinide (MA) transmutation fuel types such as advanced mixed oxides (MOX), advanced metallic alloys, inert matrix fuels (IMFs), and other ceramic fuels (nitrides, carbides, etc.) for fast neutron spectrum conditions. Most of these advanced fuel compounds have already been the object of past examination programs, which included irradiations in research reactors. The knowledge derived from previous experience constitutes a significant, albeit incomplete, body of data. Today's new or upgraded experimental tools can extend the scientific and technological knowledge associated with the new generation of nuclear reactors and fuels. More rapid progress can be achieved through effective synergy with advanced (multiscale) modeling efforts.

This project is undertaking further investigations using some of the state-of-the-art experimental techniques now available. The objectives are three-fold:

- Extend the available knowledge on properties and irradiation behavior of high burnup and minor actinide-bearing advanced fuel systems.
- Upgrade/develop advanced modeling tools using experimental data and expertise on the irradiation behavior of nuclear fuels, establishing a synergy between experimentation and multiscale modeling and code development.
- Promote the effective use of international resources—i.e., information exchange among leading experimental facilities—to characterize irradiated fuel.

Research Progress

In FY 2012, Idaho National Laboratory (INL) researchers worked towards developing high spatial resolution instrumentation that will help determine thermal conductivity and mechanical properties. They used a focused ion beam (FIB) to prepare and perform electron backscatter diffraction (EBSD) studies on irradiated fuel samples in conjunction with computational modeling and simulation efforts. Their experimental findings will be further coupled with the computational modeling and simulation efforts. Team members were successful in their efforts to transport FIB-prepared samples to the Institute for Transuranium Elements (ITU), supporting the planned joint characterization of irradiated silicon carbide (SiC).

Properties of Advanced Fuels and Synergy between Methodologies

Advanced characterization techniques such as transmission electron microscopy (TEM) and EBSD have been widely utilized for non-radiological materials; however, the extreme difficulty of working with radioactive samples has inhibited their application to nuclear investigations. These advanced techniques provide the ability to obtain critical experimental data on the evolution of microstructure, dislocation density, grain orientation, and composition in irradiated materials from the atomic to the mesoscale. Using FIB, the INL team has taken high-burnup fuel irradiated in the Fast Flux Test Facility (FFTF) reactor and produced site-specific samples for TEM and EBSD examination from this highly radioactive and friable material, obtaining first-of-a-kind images. In parallel, the team has developed the MARMOT code, INL's flagship mesoscale simulation code1 that predicts the microstructural evolution in fuel under a variety of reactor and experimental conditions.

Both the experimental methods and the modeling and simulation techniques improve understanding of the evolution of the fuel's microstructure and defining of its impact on important material properties. By combining the two methods, the INL team has developed a coordinated approach that greatly reduces the limitations of each method in isolation. Experimental microstructures, characterized both before and after irradiation, are fully reconstructed in the MARMOT code. Figure 1 shows experimental high-burnup mixed oxide fuel microstructures that have been computationally reconstructed and used as the simulation initial conditions. Using the resulting simulation, researchers can determine the effective thermal conductivity of various areas within the fuel pellet and predict grain boundary migration and fission gas segregation. The reconstruction includes the topology, grain orientation, crystal structure, and chemical composition.

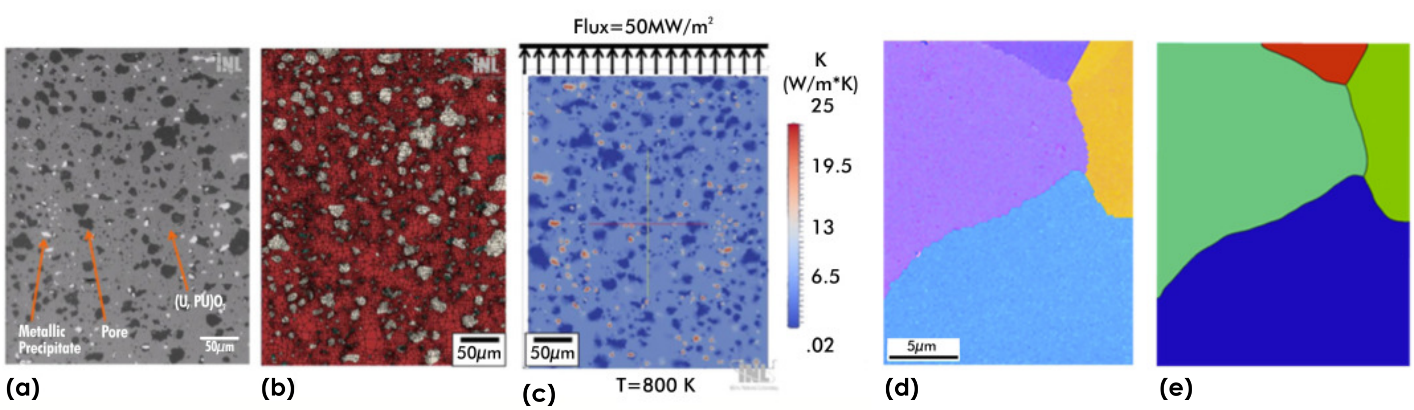


Figure 1. Examples of reconstructed microstructures: (a) optical micrograph of high-burnup ACO-3 metal oxide fuel, (b) reconstructed simulation mesh, (c) heat conduction simulation setup, with initial values for thermal conductivity overlaid on the mesh, (d) EBSD scan of ACO-3 fuel, and (e) reconstruction of the microstructure in the MARMOT code.

1. M. Tonks, D. Gaston, P. Millett, D. Andrs, and P. Talbot, "An object-oriented finite element framework for multiphysics phase field simulations," *Comp. Mat. Sci.*, 51 (2012) 20–29.

A postdoctoral researcher from ITU worked on the INL team and implemented a physics-based fission gas behavior model that he had previously developed into the BISON fuel performance code. This model was developed for analyzing the coupled phenomena of fission gas swelling and release in UO fuel during irradiation with an emphasis on modeling grain-face gas bubble development and the related dependence of the fission gas swelling and release on the local hydrostatic stress. This is of special importance when analyzing fuel behavior during power ramps and pellet- cladding mechanical interaction conditions.

ITU will provide INL (and EPRI) researchers with the final version of the Power_Condense computer program, including a new graphical interface. The program reduces the number of data points in a reactor power history, which can greatly reduce computational time when this data is imported into modeling codes. Data produced during reactor operation, such as linear heat rate, coolant temperature, or reactor state, are routinely stored in short time intervals, often 15 minutes or less. Because irradiation times can last several years, a huge amount of data is produced.

Advanced Instrumentation

The INL team is developing two advanced laser-based techniques to study the mechanical and thermal properties of nuclear fuel: the Thermal Conductivity Microscope and the Mechanical Properties Microscope.

Thermal Conductivity Microscope (TCM). This laser-based instrument measures thermal diffusivity and thermal effusivity directly, allowing users to determine thermal conductivity. This method alleviates the need to measure the specific heat separately, which is rather difficult under remote-handling conditions. The instrument is being designed so that it can be remotely operated and maintained, will function in a radiation environment, and can make measurements on nuclear fuel and other radioactive samples. Once operational, the TCM will provide micron-level thermal property information commensurate with the microstructure heterogeneity in nuclear fuel.

The project team achieved the following key accomplishments pertaining to the design, construction, and testing of the TCM:

- Developed an experimental protocol appropriate for measuring thermal conductivity of high-burnup nuclear fuel and fresh nuclear fuel.
- Completed a study on the suitability using a fiber laser for sample heating.
- Identified a figure of merit that can be used to judge data reliability.

The researchers used the TCM to obtain values for the effusivity, thermal diffusivity, and thermal conductivity of metallic systems; results compare very well with those known from the literature. Of particular note is the rather good correspondence between literature values and TCM-determined values for ceramic materials, as the data in Table 1 show (next page).

Table 1. TCM-determined and literature values for SiO_2 and CaF_2 effusivity, thermal diffusivity, and thermal conductivity.

	Sample A (SiO_2)	Sample B (CaF_2)
Phase lag (deg)	60.4	46.3
Effusivity ($J/m^2\ s^{1/2}\ K$), measured	1490	4570
Effusivity (J/m^2, $s^{1/2}\ K$), literature	1436	4989
Effusivity error	<4%	~8%
Diffusivity (m^2/s), measured	9.80×10^{-7}	3.25×10^{-6}
Diffusivity (m^2/s), literature	9.5×10^{-7}	3.4×10^{-6}
Diffusivity error	3%	4%
Conductivity (W/(m K), measured	1.47	8.24
Conductivity (W/(m K), literature	1.4	9.2
Conductivity error	5%	10%
Effusivity ($J/m^2s^{1/2}K$)	1460	4180
Measured including R_{th}=5 x 10^{-9} m^2 K/W		

Mechanical Properties Microscope (MPM). This unique laser ultrasound instrument is designed to operate in a hot cell environment via remote control manipulation. In FY 2012, researchers initiated mockup testing of the MPM to measure mechanical properties of nuclear fuel. The MPM will provide micron-level mechanical property information commensurate with microstructure heterogeneity. The MPM's spatial resolution enables results to be tied directly to other electron/photon microstructural imaging technologies. This aspect is essential to understanding microstructure's role in determining mechanical properties of nuclear fuel.

Researchers completed a number of important steps on the path to implementing remote MPM operation: obtaining final approval of the remote equipment qualification plan for Phase I testing, finalizing construction of the prototype instrument, and completing initial mockup testing. In addition, the team took first-time measurements of the elastic constants of a cast uranium–molybdenum fuel ingot and compared the data to a textured uranium–molybdenum fuel plate imposed by rolling (see Table 2).

Table 2. Comparison of elastic constants of a cast uranium–molybdenum fuel alloy and a rolled textured sample.

	Isotropic ingot	Anisotropic rolled foil	Percent change
Young's Modulus (GPa)	102	71	30
Shear Modulus	36	25	30

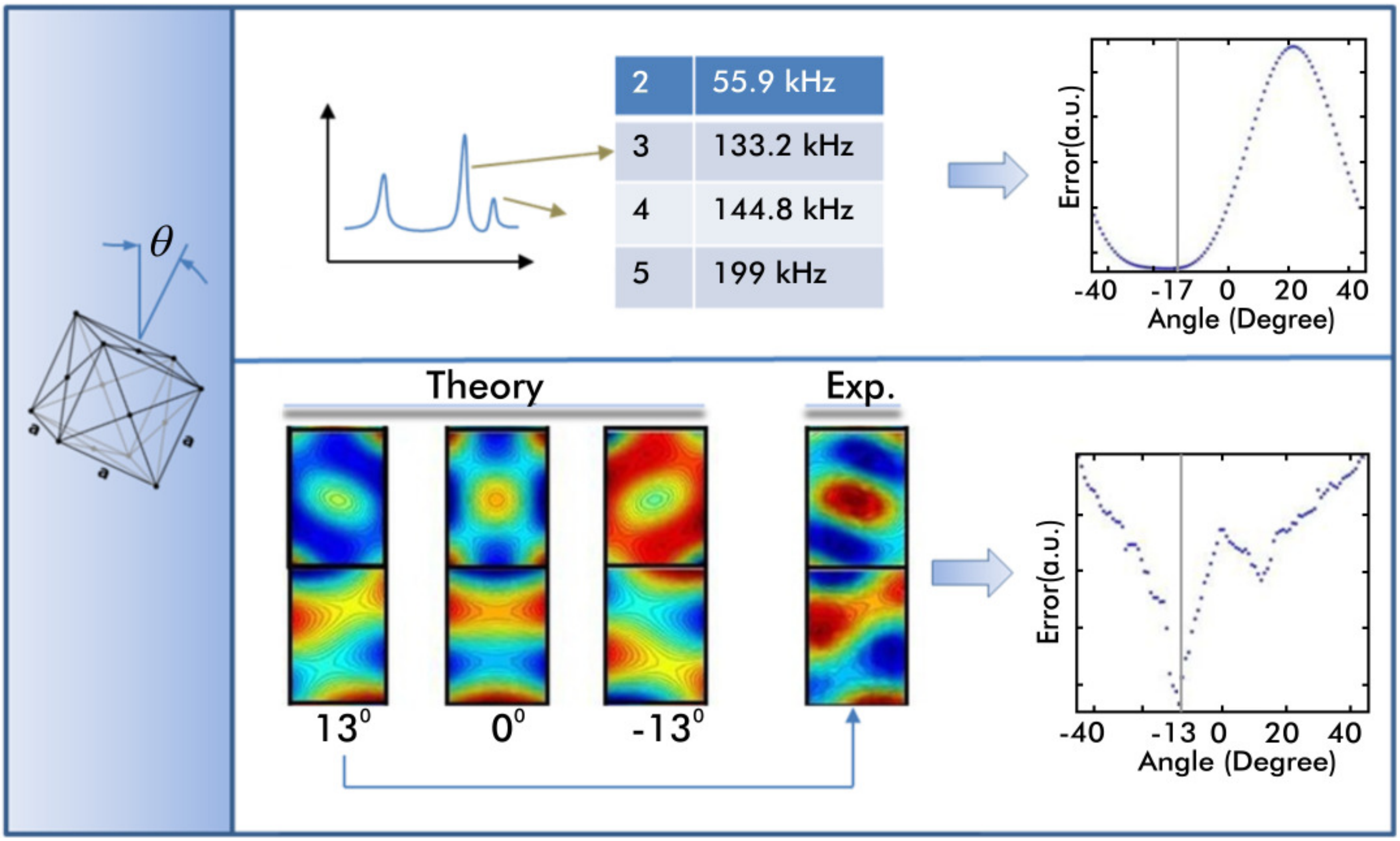


Figure 2. Cover illustration for the November 2012 issue of *IEEE-UFFC*. The MPM was used to measure crystallographic orientation of a single crystal sample. Laser resonant ultrasonic spectroscopy (LRUS) uses lasers to excite and detect resonant modes. The crystallographic orientation of a copper sample determined using the eigenmode method (bottom) is compared with the traditional eigenfrequency method (top).

Joint Studies

One of the more logistically difficult project tasks is a combined characterization effort of irradiated fuel samples. The collaborators have discussed several irradiated and non-irradiated fuel materials and are exploring the viability of sharing Euratom's METAPHIX irradiation test samples with U.S. researchers. The transport of samples between INL and ITU remains a difficult issue.

Nonetheless, the project team has made progress with the cooperative study of irradiated SiC. Four β-SiC samples have been investigated, three of which were irradiated in the Massachusetts Institute of Technology reactor, under the umbrella of the Advanced Test Reactor National Scientific User Facility. Colorado School of Mines performed Hall coefficient measurements in 2011 to determine electrical carrier densities and impedance. In CY 2012, INL employed FIB techniques to prepare atom probe tomography, transmission electron microscopy, and nano- and micro- indentation measurement samples, as well as samples for shipment to ITU for micro- indentation and TEM measurements. These samples have been successfully shipped and are currently under examination at ITU.

Along with materials, the two research teams continue to exchange information and knowledge. Drawing on ITU experience, INL is working to implement an electron probe microanalyzer (EPMA) for highly radioactive fuels and materials. The two teams reviewed the coating device for preparation of EPMA samples, how INL could adapt the design, and how to use software for microprobe measurement and data processing. In addition, INL remains involved in the ITU-initiated informal Cameca shielded SX 100 group of users, who share experiences and problem-solving ideas.

Planned Activities

In FY 2013, researchers plan to extend the EPMA information exchange to the nuclearization of a FIB apparatus. A FIB will be delivered to ITU in early 2013, and their researchers will visit INL to discuss details of the device's installation and nuclearization of the device.

Researchers will maintain interactions related to post-irradiation examinations, including operation of an EPMA. They will also continue to develop advanced techniques such as micro X-ray diffraction, FIB, thermal conductivity measurements, and mechanical property measurements.

The project team will move forward with joint studies on irradiated SiC. Team members will also continue their efforts to resolve issues associated with transport of high-level radioactive materials, including fuels. For the project to deploy its full scope—including joint characterization of materials—a viable means of transport to facilitate sample exchange is essential. Until a solution is found, the scope of the joint work will remain forcibly limited.

Development of a 2E-2V Instrument for Fission Fragment Research

Project Number: 2011-001-E

PI (U.S.): Fredrik Tovesson, Los Alamos National Laboratory

PI (Euratom): Stephan Oberstedt, Joint Research Center–Institute for Reference Materials and Measurements

Collaborators: Idaho National Laboratory

Program Area: Reactor Concepts RD&D

Project Start Date: October 2011

Project End Date: September 2014

Research Objectives

The objective of the project is to produce accurate data files of fission measurements and evaluations for several key isotopes over the incident neutron energy range relevant to present and future nuclear applications. Design, optimization, and safety assessment of future fast reactor systems require improved fission fragment nuclear data for major and minor actinides. The project will contribute to the ENDF/B-VII and JEFF 3.1 nuclear data libraries.

The research team is developing instrumentation for high-resolution measurements of fission product velocity, energy, and nuclear charge. The resulting data provides fission product mass and the corresponding yield curves. This work supports ongoing developments at two spectrometer facilities: the VERDI (VElocity foR Direct particle Identification) spectrometer at the Institute for Reference Materials and Measurements, and a similar dual-arm spectrometer at Los Alamos National Laboratory (LANL). The current VERDI spectrometer measures energy and velocity from only one of the two fission products emitted in binary fission (hence, 1E-1V), while the advanced instruments will simultaneously measure both fission fragments (2E-2V). In addition, the LANL dual-arm instrument will also use Bragg curve spectroscopy to measure nuclear charge.

The collaboration consists of exchange of expertise and technologies, joint experimental efforts, sharing of detector designs, and communication of technical and scientific information. The two instruments are being developed in parallel, and the teams share technologies that benefit the joint program. Experimental activities are also being carried out with other facilities at both laboratories, including the Geel Linear Accelerator (GELINA), the 7-MV Van de Graaff neutron source (MONNET), and the Los Alamos Neutron Science Center (LANSCE). Key project tasks are as follows:

- Complete the design study.
- Prepare prototype design/technical report.
- Conduct initial measurements.

Research Progress

Work during the first year focused on testing the individual parts of the detector array to develop a design for the full system. In order to build a 2E-2V instrument with high resolution, it is necessary to design detectors that measure time of flight with high timing resolution (100–200 picoseconds [ps]) and an instrument that provides high- resolution (< 1%) energy measurements of fission fragments.

Researchers completed testing of a time-of-flight instrument based on thin conversion foils, electrostatic mirrors and ultra- fast micro channel plates. The system's timing resolution was tested with fission fragments from a Cf-252 spontaneous fission source and with alpha particles from a Th-229 calibration source. As shown in Figure 1, the measurement using the Th-229 source indicates five peaks in the time-of-flight histogram— one for each of the five main alpha-particle energies. The time resolution extracted from this measurement is 190 ps, corresponding to 135 ps per detector, which meets the design requirement. Based on these results, the project team decided to use this basic design for the time-of-flight section of the full instrument.

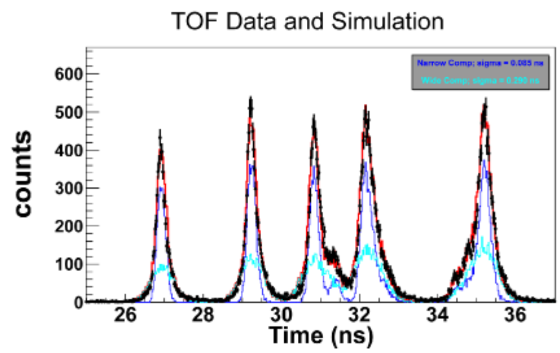


Figure 1. Time-of-flight measurement of alpha-particles from a Th-229 calibration source (black points) overlap the simulation (red line). The time resolution was determined to be 190 ps (full width at half maximum [FWHM]) based on this measurement.

Detection efficiency of the time-of-flight detectors is important to ensure high counting statistics in fission yield measurements. Using a calibration source, the research team evaluated efficiency by registering coincidences between the timing detector and a silicon detector. The number of coincidences was measured for different accelerating potentials on the timing detector's electrostatic mirror, and the result is shown in Figure 2. The efficiency for alpha-particles was shown to be about 70% and fairly insensitive to the acceleration potential. The next step will be to check the efficiency for fission fragments, which is expected to be significantly higher because of their higher specific ionization in the conversion foil.

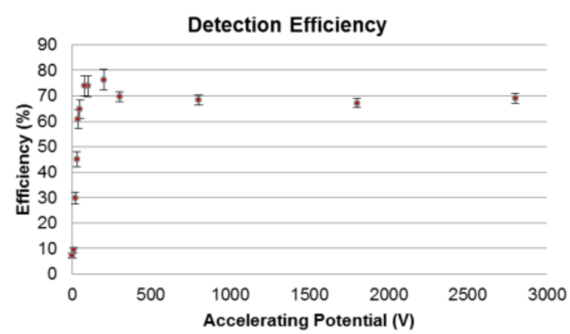
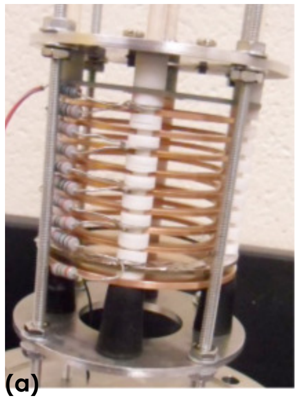


Figure 2. Efficiency of the timing detectors for alpha-particles as a function of accelerating potential on the electrostatic mirror.

The project team also designed a first ionization chamber prototype for fission fragment energy measurements, shown in Figure 3a. In this case, a tri-isotope alpha-source was used to investigate the resolution, which was determined to be 1.1% (see Figure 3b). This still needs to be improved slightly, and there is work in progress to build a second prototype that will be run with different counting gases, at lower pressures, and with more sophisticated signal processing to further improve the resolution.

Planned Activities

The work on the 2E-2V instrument in FY 2013 will focus on completing ionization chamber performance testing and designing a fully integrated prototype instrument. The ionization chamber testing will study the different variables that affect resolution, such as gas pressures, gas type, analog signal processing, and output digitization.



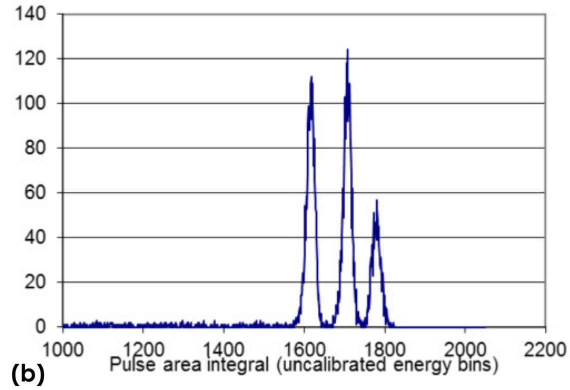


Figure 3. (a) The first ionization chamber prototype; (b) an energy spectrum collected with a tri-nuclide alpha-source.

The prototype design will consist of two identical collinear spectrometers, each with a time-of-flight and energy detector section. The time-of-flight section needs to be under vacuum ($<10^{-6}$ torr), so one of the challenges in system integration is to design a thin window that isolates the vacuum side from the ionization chamber that will be operated at around 150 torr.

High-Fidelity Thermal Hydraulic Fuel Assembly Simulations for Nuclear Reactors

Project Number: 2012-001-E

PI (U.S.): Elia Merzari, Argonne National Laboratory

PI (Euratom): Ferry Roelofs, Nuclear Research and Consultancy Group

Collaborators: Ghent University, SCK•CEN (Belgian Nuclear Research Centre)

Program Area: Reactor Concepts RD&D

Project Start Date: October 2012

Project End Date: September 2015

This project will engage in cross-code verification of the high-fidelity computational fluid dynamics (CFD) models applied to nuclear reactor core flows. The aim is to systematically cross-verify the models of the four partner institutions in nuclear reactor core geometries characterized by either rod bundles or pebble beds, using common data shared among team members. A more reliable methodology can reduce arbitrary safety margins and the need for testing new designs and lead to improved design efficiency. However, proposed methodologies must be thoroughly verified and validated. There is currently a scarcity of useful experimental data to validate high-fidelity computations in complex geometries such as those in reactor cores.

At Argonne National Laboratory, the Simulation-based High-efficiency Advanced Reactor Prototyping (SHARP) project is a multi-divisional collaborative effort to develop a modern set of design and analysis tools for advanced nuclear reactors. With the SHARP suite, users construct highly detailed component models using high-fidelity methods or complex virtual reactor models that accurately reproduce the multi-physics behavior of nuclear reactor cores. The Nuclear Research and Consultancy Group employs primarily the commercial STAR-CCM+ finite volume CFD code, while Ghent University and SCK•CEN both use the ANSYS suite of codes. These codes will be compared using commonly agreed benchmarks.

The project objectives are to 1) systematically verify high-fidelity computational tools in the two geometries under examination (rod bundles and pebble beds) and 2) identify potential experiments to be used in validation exercises. By exchanging information and benchmark results, the researchers will be able to identify discrepancies or concerns with current predicting technologies. With these concerns identified, the team will then propose an experimental plan for validation.

The project involves three major tasks:

- **Verification of large eddy simulation (LES) or direct numerical simulation (DNS) CFD codes for rod bundle flows and related analysis.** Rod bundles in nuclear reactors are typically spaced by rigid mechanical devices such as wires or grids. The presence of such devices complicates the flow considerably, and the lack of detailed experiments makes CFD validation problematic. The participants will work toward verifying numerical simulations pertaining to rod bundle flows.
- **Verification of DNS/LES CFD code for pebble bed flows and related analysis.** Participants will simulate pebble bed core flows and compare results. Future work will focus on finding efficient ways to accurately compute the flow in a realistic random pebble bed.
- **Experimental plan for validation of LES/DNS codes for rod bundles and pebble beds.** Based on the results of the two tasks above, the research team will devise a separate effects experimental plan to validate high-fidelity CFD codes.

4.2 United States–Republic of Korea Collaborative Projects

Director William D. Magwood IV of DOE-NE signed the first bilateral I-NERI agreement on May 16, 2001, with Director General Chung Won Cho of the Republic of Korea (ROK) Atomic Energy Bureau, signing for the ROK Ministry of Science and Technology. The first U.S.–ROK collaborative research projects were awarded in FY 2002, with a total of 46 projects awarded to date.

Areas of mutual interest between the two countries cover next-generation reactor and fuel cycle technology concepts that increase efficiency, safety and proliferation resistance; innovative nuclear plant design, manufacturing, construction, operation, maintenance and decommissioning activities; advanced nuclear fuels; reactor safety; and fundamental nuclear sciences.

FY 2012 marked the completion of one U.S.–ROK collaboration, while efforts continue on ten ongoing projects. One U.S.–ROK project team is focused on plant safety during beyond design-basis accidents. Five collaborations contributed to the materials knowledge base. Two of these are developing nanostructured metal alloys through innovative treatment processes, while two are looking specifically at cause and prevention of stress corrosion cracking. The fifth is developing microcharacterization techniques. Two projects are supporting the fuel cycle for existing LWR fleets, one through introduction of an accident-tolerant fuel, and the other through electrochemical technology methods to recover zirconium from used nuclear fuel rods. Three others are studying advanced fuel cycles, including the physics of transuranic burner reactors, fuel fabrication for SFRs, and high-fidelity multi-physics simulation methods and codes for very high-temperature reactors (VHTRs).

This section provides a listing of current I-NERI U.S.–ROK projects, along with summaries of FY 2012 accomplishments.

2009-001-K	ZPPR-15 and BFS Critical Experiments Analysis for Generation of Physics Validation Database of Metallic-Fueled Fast Reactor Systems
2009-002-K *	Enhanced Radiation Resistance Through Interface Modification of Nanostructured Steels for Gen IV In-Core Applications
2010-001-K	Investigation of Electrochemical Recovery of Zirconium from Spent Nuclear Fuels
2010-002-K	Science-Based Approach to Nickel Alloy Aging and Its Effect on Cracking in Pressurized Water Reactors
2010-003-K	Low-Loss Advanced Metallic Fuel Casting Evaluation
2010-004-K	Development and Characterization of Nanoparticle-Strengthened Dual-Phase Alloys for High-Temperature Nuclear Reactor Applications
2011-001-K	Atomic Ordering in Alloy 690 and Its Effect on Long-Term Structural Stability and Stress Corrosion Cracking Susceptibility
2011-002-K	Development of Microcharacterization Techniques for Nuclear Materials
2011-003-K	Verification and Validation of High-Fidelity Multi-Physics Simulation Codes for Advanced Nuclear Reactors
2011-004-K	Development of Diagnostics and Prognostics Methods for Sustainability of Nuclear Power Plant Safety Critical Functions
2011-005-K	Fully Ceramic Microencapsulated Replacement Fuel for Light Water Reactor Sustainability

* Completed in FY 2012

ZPPR-15 and BFS Critical Experiments Analysis for Generation of Physics Validation Database of Metallic-Fueled Fast Reactor Systems

Project Number: 2009-001-K

PI (U.S.): Richard McKnight, Argonne National Laboratory

PI (ROK): Sang Ji Kim, Korea Atomic Energy Research Institute

Collaborators: Korea Advanced Institute of Science and Technology

Program Area: Reactor Concepts RD&D

Project Start Date: November 2009

Project End Date: April 2013

Research Objectives

This collaborative research program comprises two related research areas. The project team is archiving the ZPPR-15 loading records for Phases A, B, C, and D and using these valuable metallic fuel measurement data to generate high- fidelity as-built Monte Carlo models. These models will be used to validate advanced computational design analysis methods, thereby reducing uncertainties.

In parallel, researchers from Argonne National Laboratory (ANL) and the Korea Atomic Energy Research Institute (KAERI) are collaborating to develop both an experimental plan and models to study the physics of a transuranic (TRU) burner reactor. The last 20 years have seen a considerable evolution in the fast reactor core concept. These experiments, which the KAERI team will conduct, will investigate design features specific to a typical TRU burner core, covering physics areas that the ZPPR-15 physics experiment did not, while considering licensing requirements for a commercial reactor. Results will be used to produce valuable high-fidelity models. Both teams are collaborating in developing the high-fidelity models and will share the final results.

Research Progress

Analyses of ZPPR-15 Experiments and Generation of the Experimental Database

Work on ZPPR-15 Phases B (plutonium core), C (plutonium and uranium mixed core), and D (uranium core) is progressing well. Phase A was completed in the first project year. For each configuration, ANL researchers have captured and verified complete loading records (see Figure 1) and entered them into a database for generating the as-built models. To date, they have generated detailed as-built MCNP models for six ZPPR-15B experimental configurations representing control rod worth measurement, four ZPPR-15C experimental configurations representing sodium void reactivity worth measurement, and two of the proposed nine ZPPR-15D configurations. The team has also reviewed all ZPPR-15 Phase B, C, and D drawer master identifications and logbooks, effectively assuring that this valuable integral data will not be lost and the models will be available for future validation activities, such as those undertaken by the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program.

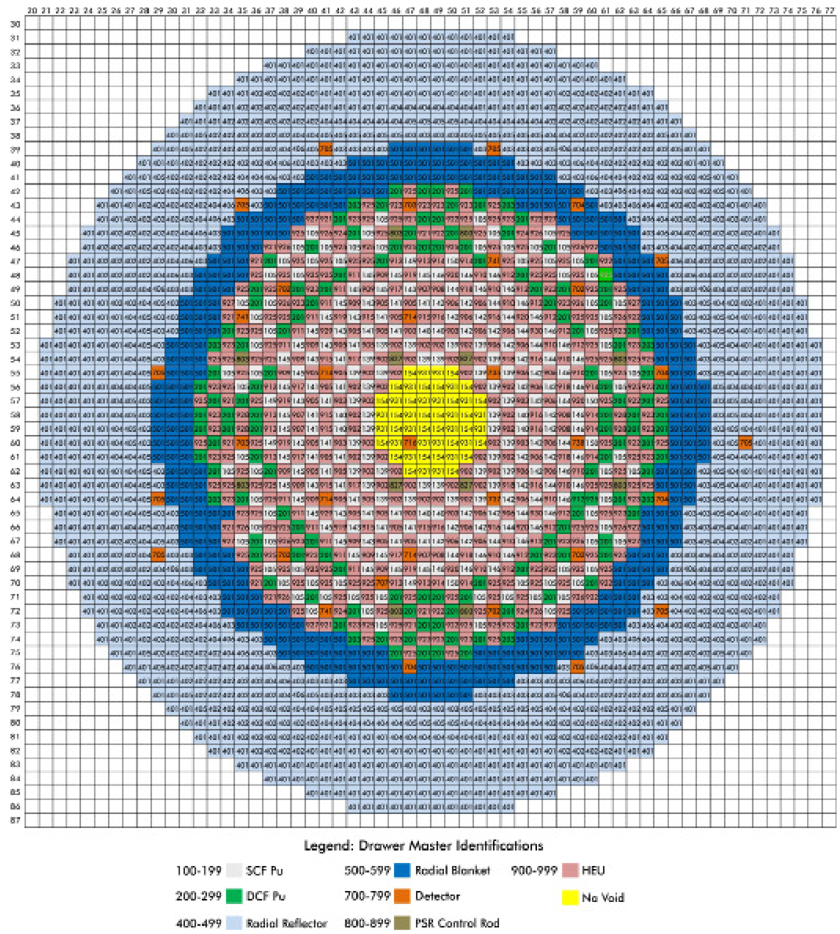


Figure 1. Core map of ZPPR-15C loading 168.

These experiments have been analyzed with MCNP using ENDF/B-VII.0 data. The resulting k values are good, although they exhibit a slight underprediction for ZPPR-15C (-32 pcm) and a slight overprediction for ZPPR-15D (+120 pcm). Although these biases are only marginally larger than the measurement uncertainties, similar results for ZPPR-15B (-200 pcm) indicate a consistent trend. However, sodium void worth results with ENDF/B-VII.0 data are generally within ~5% to 10%. The positive worths of voiding the central zones are in excellent agreement with the measured values—within 1%.

The KAERI team also analyzed the same 12 experimental configurations and provided various modeling approximations for the drawer masters. In addition to the analyses of whole core models, they analyzed eigenvalues of unit drawer models for uranium- fueled drawer masters and compared with the as-built drawer models. Results indicated that the proposed one-dimensional drawer master model reproduced k in uranium-fueled drawer masters with the same degree of accuracy as in plutonium- fueled drawer masters.

The team analyzed the benchmark configurations using the TRANSX/CRX-1D/ THREEDANT deterministic code system. They calculated plate-by-plate neutron flux for spatial homogenization using the CRX-1D solver, developed in the first year of this project and incorporated into the TRANSX code, to apply equivalence theory for treatment of heterogeneous self-shielding.

The average k discrepancies in homogeneous models between THREEDANT and the MCNP calculations were about 77 pcm Δρ for the Phase B Pu core, 62 pcm Δρ for the Phase C mixed Pu-U core, and 41pcm Δρ for the all-U Phase D core. When one-dimensional drawer models are used, the discrepancies between THREEDANT and MCNP calculations increased from -897 to +78.6 pcm Δρ on average, in accordance with the increased percentage of uranium. These relatively large discrepancies seem to originate from the Dancoff correction approximation as applied to the broad energy group width of a 150-group structure. Hence, hyper-fine energy group calculations should improve the results of deterministic one-dimensional drawer models.

The KAERI team also investigated anisotropic scattering treatment in the Monte Carlo and deterministic codes for representative multi-group ZPPR-15 benchmark problems. Limited ability to address negative scattering cross sections (those originating from higher-order Legendre moment) currently leads Monte Carlo calculations to overestimate eigenvalues by about 200 pcm Δk. However, for isotropic scattering, the multigroup Monte Carlo calculations resulted in a discrepancy of only 1 pcm when compared to THREEDANT deterministic calculations. These results are valuable for future validation of ZPPR-15 experiments based on multi-group cross-section structures.

Compilation of Physics Experiments at the BFS Facility

The KAERI team has generated 35 Monte Carlo models for the BFS-76-1A critical experiment, based on the information in the original experimental report from Russia's Institute of Physics and Power Engineering (IPPE), where the BFS facility is located. These models were transmitted to ANL along with a reference document. The MCNP model at a critical configuration of BFS-76-1A underestimated the k compared with the measured value. In collaboration with IPPE researchers, the KAERI team is investigating the reason for this underestimation. Researchers have also calculated sodium void reactivities and control rod worths and compared findings with measured values from the ENDF/B-VII.0 library, revealing agreement with 10%~15% accuracy. The KAERI team delivered a final BFS-76-1A report to the ANL team, including measurement uncertainties and detector count rates of control rod drop experiments.

Planned Activities

The ANL team will continue verifying and uploading ZPPR-15 experimental data into the database and will complete the remainder of the ZPPR-15D as-built models. Once these models are available, KAERI will conduct analyses similar to those described above. ANL expects these activities to continue past the official project completion date. KAERI will also further examine the reference as-built Monte Carlo model of the BFS-76-1A critical experiment, aiming to resolve the discrepancy between predicted

and measured eigenvalues. Once the KAERI researchers have successfully modified the model, they will deliver it to the ANL team.

Enhanced Radiation Resistance through Interface Modification of Nanostructured Steels for Gen IV In-Core Applications

Project Number: 2009-002-K

PI (U.S.): Todd R. Allen, University of Wisconsin–Madison

PI (ROK): Jinsung Jang, Korea Atomic Energy Research Institute

Collaborators: Argonne National Laboratory, Los Alamos National Laboratory, Seoul National University, Texas A&M University, University of Florida

Program Area: FCR&D

Project Start Date: October 2009

Project End Date: September 2012

Research Objectives

The objective of this project, under an area of interest referred to as "nanonuclear," was to increase radiation tolerance in candidate alloys for Gen IV fuel cladding through optimization of grain size and grain boundary characteristics. The focus has been on nanocrystalline metal alloys with a face-centered cubic (fcc) crystal structure. Through a combination of grain refinement and grain boundary engineering, the scientists tailored material strength, ductility, and resistance to swelling by 1) changing the sink strength for point defects, 2) increasing the nucleation barriers for bubble formation at grain boundaries, and 3) changing the precipitate distributions at boundaries. The long-term goal is to design and develop bulk nanostructured austenitic steels with enhanced void swelling resistance and substantial ductility, and to enhance their creep resistance at elevated temperatures.

Compared to ferritic/martensitic steels, austenitic steels possess good creep and fatigue properties at elevated temperatures and improved toughness at low temperatures. However, austenitic steels have a major disadvantage in their vulnerability to significant void swelling in nuclear reactors, especially at the temperatures and radiation doses anticipated in the Gen IV reactors. This lack of resistance to void swelling led the research team to adopt ferritic/martensitic steels as the preferred material for the fast reactor cladding application. Recently, however, scientists at Oak Ridge National Laboratory developed a new type of austenitic stainless steel, HT-UPS (high-temperature ultrafine precipitate-strengthened), which is expected to show enhanced void swelling resistance through the trapping of point defects at nanometer-sized carbides.

Researchers demonstrated three effects of reducing the grain size and increasing the fraction of low-energy grain boundaries: 1) reducing the available radiation-induced point defects (by increasing the sink area of the grain boundaries), 2) making bubble nucleation at the boundaries less likely (by reducing the fraction of high-energy boundaries), and 3) improving material strength and ductility under irradiation (by producing a higher density of nanometer-sized carbides on the boundaries). Although this project focused on void swelling, advances in processing of austenitic steels are also likely to improve the radiation response of the mechanical properties.

Research Progress

Both U.S. and Korean researchers performed a significant amount of work to refine the microstructure of austenitic stainless steels and evaluate the resulting properties. Compared with their coarse-grained (CG) counterparts, the nanograined steels have shown extraordinary tolerance against helium, krypton and iron ion irradiation, while maintaining high strength and reasonable ductility. Nanocrystalline 304 and 316L stainless steels are thermally stable up to 600°C. In Phase 2 of the project, researchers evaluated these nanograined steels for neutron radiation tolerance and radiation resistance up to 300 dpa.

The research team applied equal channel angular processing (ECAP) to change the chemistry and refine the size and distribution of oxide nanoparticles. ECAP has been found to be an efficient and promising technique for thermomechanical treatment of a variety of steels and alloys. This final report briefly summarizes some of the highlights of the project.

Researchers examined the deformation mechanisms of Fe-14Cr-16Ni alloys subjected to ECAP. The average grain size has been refined from 700 µm down to about 400 nm, while yield strength is five to six times greater than CG alloys. Ductility remained very high, at approximately 15 percent uniform elongation.

Researchers also evaluated the helium ion irradiation resistance of ultrafine-grained (UFG) Fe-14Cr-16Ni alloys. At a peak fluence level of 5.5 dpa, both CG and UFG alloys had helium bubbles of 0.5–2 nm in diameter. However, the density of helium bubbles, dislocation loops, and radiation hardening were significantly reduced in the UFG Fe-Cr- Ni alloy compared to those in its CG counterpart. The results imply that refinement of microstructures can effectively enhance radiation tolerance in bulk metals.

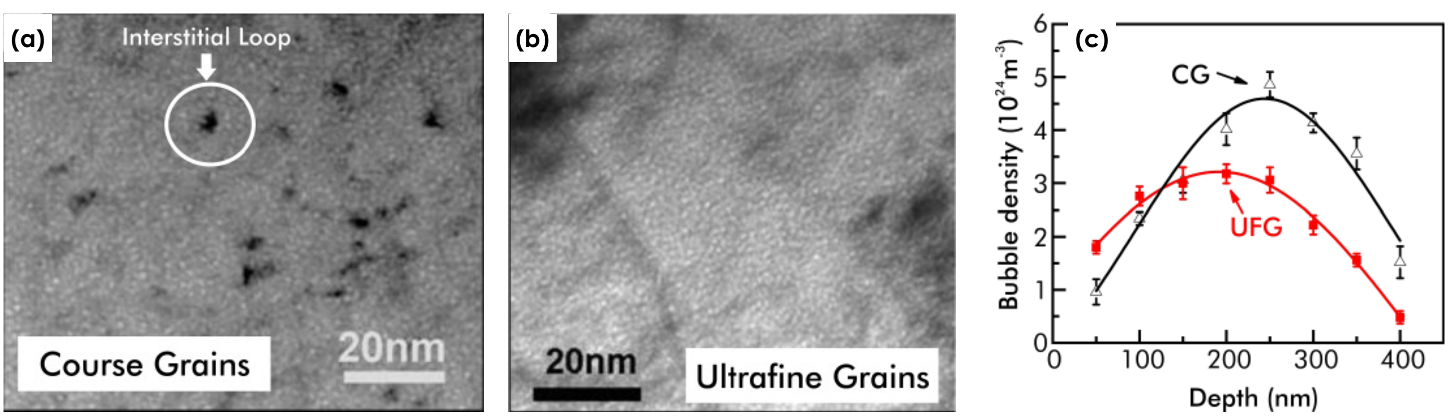


Figure 1. Cross-sectional transmission electron microscope (TEM) micrograph of Fe-Cr-Ni alloy irradiated with helium ions at 150 keV and a fluence of 6 × 1016 cm-2: (a) CG alloys with grain sizes of ~700 µm show a high density of helium bubbles and interstitial loops (black dots); (b) UFG alloys with grain sizes of ~400 nm have a lower density of helium bubbles and no dislocation loops; (c) helium bubble density is evidently lower in UFG alloys than in CG alloys.

In situ krypton irradiation was performed on Fe-14Cr-16Ni alloys after ECAP. Studies show that dislocation loop density in the ECAP alloys is much lower than in the as-received CG alloys (see Figure 2). Furthermore,

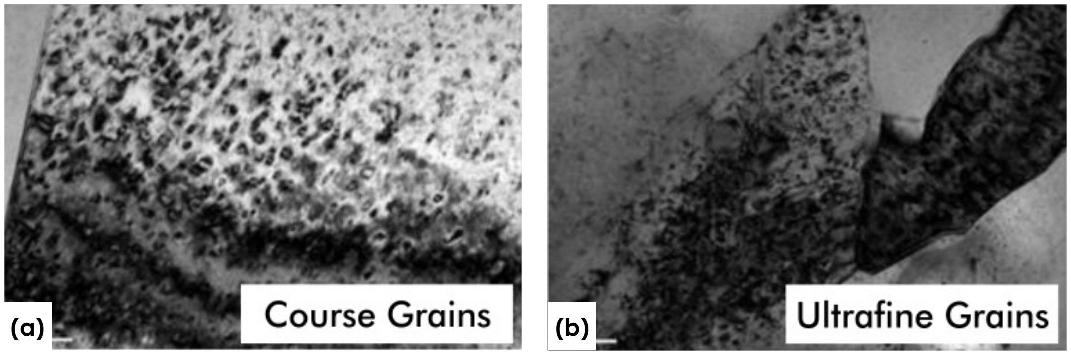


Figure 2. Microstructural evolutions in (a) CG and (b) ECAPed UFG alloys during *in situ* krypton radiation at 400°C up to 5 dpa (experiments performed at the Argonne-IVEM [intermediate voltage TEM] center). UFG alloys clearly had much lower dislocation loop density and smaller dislocation loop size.

no defect denuded zone was observed along grain boundaries in UFG specimens. This may be related to the fact that grain size (~400 nm) is greater than TEM foil thickness (~100nm), and thus foil surfaces act as defect sinks.

The research team also processed 304L and 316L stainless using ECAP. The final products have an average grain size of ~100 nm, and grains are mostly equiaxed. The team then evaluated the microstructure and tensile behavior and concluded that ECAP induces significant strengthening in these alloys.

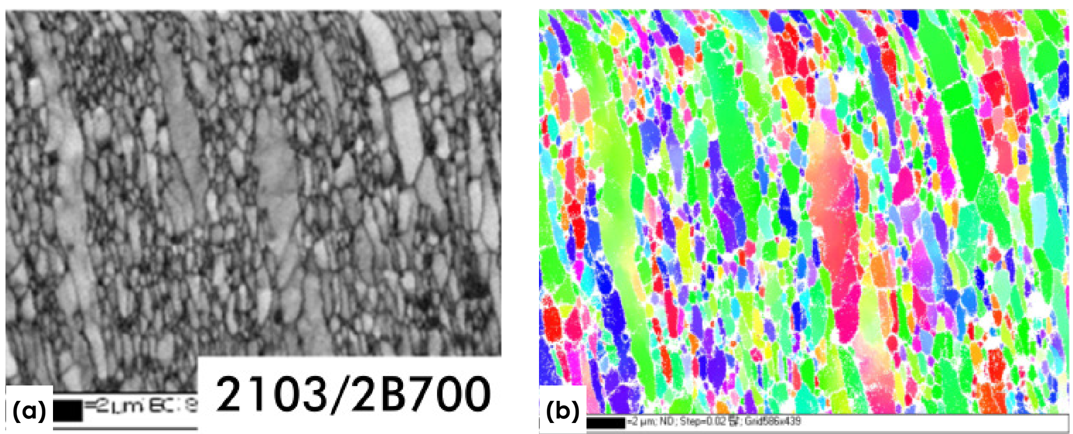


Figure 3. Examination of ECAPed 2103 ODS alloy microstructure shows (a) grain refinement and (b) the formation of predominantly high-angle grain boundaries.

The South Korea research team prepared oxide dispersion-strengthened (ODS) alloys by consolidating mechanically alloyed powders, then processed the as- consolidated bars using ECAP. Preliminary microscopy studies show moderate grain refinement and uniform dispersion of nanometer-scale oxide particles. South Korea researchers also investigated the microstructure and hardness of ECAP-processed ODS alloys. Figure 3 shows that ECAP can refine the microstructure of ODS alloys and introduce high-angle grain boundaries. The researchers also examined PM2000 alloys and performed torsion studies of 316L stainless steels.

The research team conducted in situ krypton ion irradiation under a transmission electron microscope. Significant microstructural damage, in the form of defect clusters, typically occurs in metals subjected to heavy ion irradiation. High-angle grain boundaries have long been postulated as sinks for defect clusters, such as dislocation loops. The results of this project activity provide direct evidence that high- angle grain boundaries in nanocrystalline nickel, with an average grain size of ~55 nm, can significantly reduce the density and size of radiation-induced dislocation loops compared to their bulk counterparts, thus significantly enhancing the nickel's radiation tolerance.

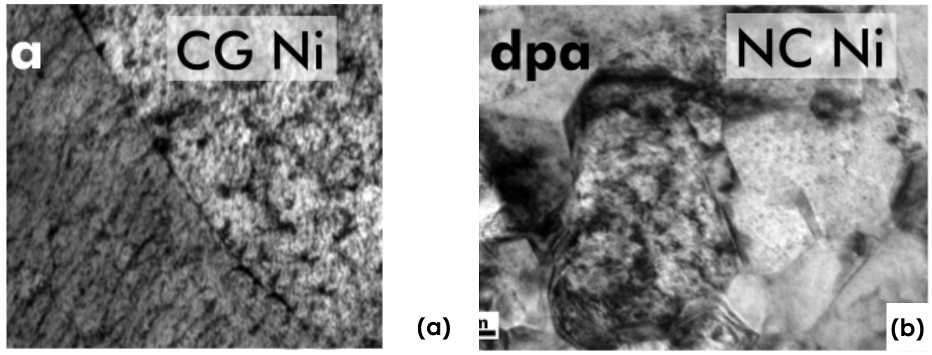


Figure 4. *In situ* krypton irradiation shows significantly higher defect density in (a) CG nickel than in (b) nanocrystalline nickel.

The research team used numerous microscopy techniques to investigate a 12Cr-ODS steel that had undergone multiscale microstructure refinement through equal channel angular extrusion (ECAE) using a hot isostatic press (HIP). The material processed by HIP showed a bimodal microstructure. Researchers observed three types of second-phase particles, including Y_2O_3, $(Cr, Fe)_{23}C_6$ and Cr oxide. The post-ECAE alloy showed refined grains and a homogeneous grain size. The chromium-rich phases (chromium oxide and chromium carbide) were effectively refined, while yttrium oxide particles around the prior UFG boundaries were redistributed. Large yttrium oxide particles underwent a geometry change from round or faceted to triangular or elliptic. The steel hardened by approximately 35%, a change related to grain refinement and precipitation hardening.

The team used high-resolution electron backscattered diffraction with a newly designed deformation device to observe the microstructure and texture evolution of ODS ferritic steel during stepwise uniaxial tensile deformation. Both the rotation behavior of individual ferrite grains and overall preferred orientation were traced and analyzed. Findings indicate the grains' tendency to rotate towards the stable orientation of ‹1 0 0› //NDand ‹1 1 1›// ND, although the degree of rotation was dependent on the grains' initial orientation. Vickers hardness tests and TEM analyses before and after uniaxial tensile deformation correlated the microstructure with its mechanical properties.

Iron ion irradiation of ECAPed Fe-17Cr-12Ni-2.5Mo alloys indicated that the crystal structure of the irradiated sample surface changed from fcc to a body-centered cubic (bcc) phase. The newly developed bcc grains had sizes consistent with the ion irradiation depth, which could be estimated using molecular dynamics simulations.

Planned Activities

This was the final year of the I-NERI project. The project team has already published partial findings and will document the remaining experimental results in a few more papers. The team is now actively seeking potential funding opportunities to continue collaborations in order to examine proton, heavy ion, and neutron radiation-induced damage in nanocrystalline steels and ODS steels.

Investigation of Electrochemical Recovery of Zirconium from Spent Nuclear Fuels

Project Number: 2010-001-K

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

PI (ROK): Il-Soon Hwang, Seoul National University

Collaborators: Korea Atomic Energy Research Institute, University of Idaho

Program Area: FCR&D

Project Start Date: December 2010

Project End Date: November 2013

Research Objectives

This project uses both modeling and experimental studies to design optimal electrochemical technology methods for recovery of zirconium from used nuclear fuel rods for more effective waste management. The objectives are to provide a means of efficiently separating zirconium into metallic high-level waste forms and to support development of a process for decontamination of zircaloy hulls. Modeling work includes extension of a 3D model previously developed by Seoul National University for uranium electrorefining by adding the ability to predict zirconium behavior. Experimental validation activities include tests for recovery of zirconium from molten salt solutions and aqueous tests using surrogate materials.

Research Progress

In the second year of this project, the University of Idaho focused on experimental measurements of molten salts to provide key model parameters to Seoul National University in support of their development of a 3D kinetic model for U-Zr electrorefining. Cyclic voltammetry and chronopotentiometry were performed on $ZrCl_4$ and UCl_3 mixtures in the eutectic LiCl-KCl at 500°C. From initial analysis of the data, it is clear that zirconium has a complex behavior in the molten salt, with the presence of $ZrCl_4$, $ZrCl_2$, and ZrCl. The research team calculated possible diffusion coefficients for zirconium in the eutectic salt; results range from 2.88×10^{-7} to 1.99×10^{-5} cm²/s, which compares well with previously reported data. The team also calculated an effective apparent standard reduction potential for zirconium, finding a value of -0.849 V for Ag/AgCl, which also compares well with the value reported by Argonne National Laboratory. Initial analysis of the uranium cyclic voltammetry data reveals quasi-reversible behavior with the presence of UCl_4 and UCl_3. The calculated apparent standard reduction potentials of Ag/AgCl are 1.15 V and -0.33 V for the U(III)/U and U(IV)/U(III) redox couples, respectively. These values compare well with many reduction potentials reported in the literature. The cyclic voltammogram for 1.0 wt% UCl_3 in LiCl-KCl is shown in Figure 1.

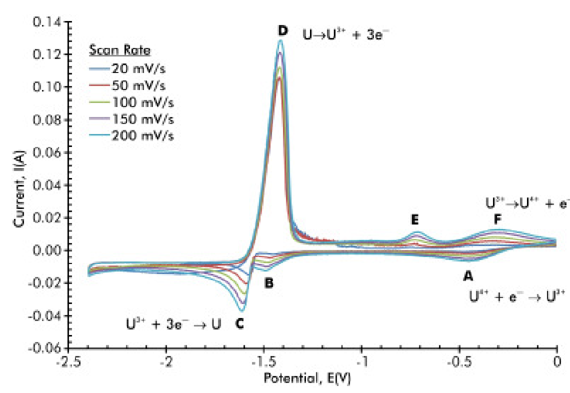


Figure 1. Cyclic voltammogram for 1.0 wt% UCl_3 in LiCl-KCl.

The project team developed a codeposition computational model to study an electrolytic zirconium recovery system based on the process of competitive electrodeposition. They chose copper and nickel electrolysis in an aqueous solution as the surrogate metals for developing a dynamic codeposition model. Using the rotating cylinder hull (RCH) cell, which provides well-controlled mass transport, researchers simulated binary (copper–nickel) electrochemical codeposition behaviors to examine multispecies electrodeposition. Based on the results, the team formulated a new mathematical model to simulate the dynamic behavior of binary electrochemical deposition on a rotating cylinder electrode. In addition, binary electrodeposition analyses were carried out in the surrogate system consisting of copper–nickel in an aqueous electrolyte. The team found that reasonable results could be obtained for the comprehensive understanding of partial current density of species deposition in association with local electrode overpotential. Figure 2 provides an image of the RCH cell and a schematic of its internal structure.

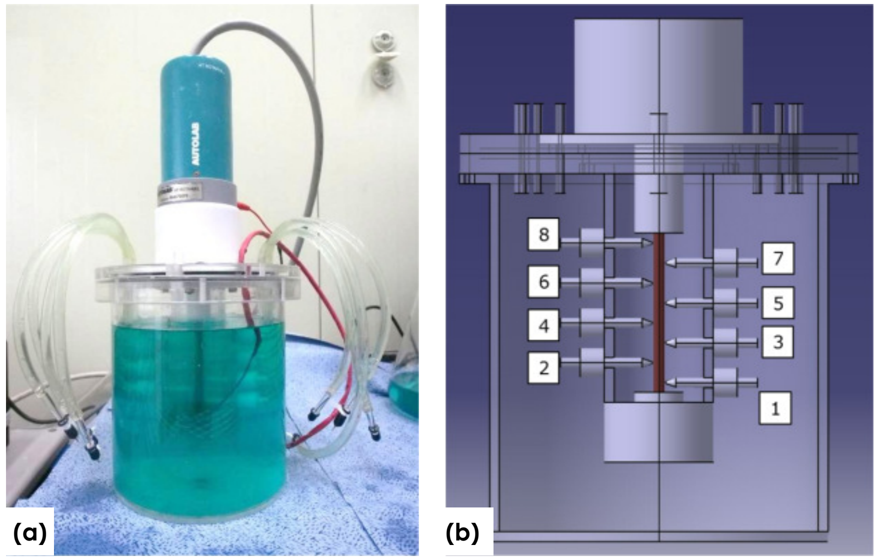


Figure 2. (a) The RCH cell used to validate the codeposition computational model; (b) schematic of cell's internal structure.

To validate the computational model, the researchers conducted copper and nickel deposition experiments in a static cell and an RCH cell. They used cathodic polarization curve fitting on copper and nickel deposition to estimate the transfer coefficient, exchange current density, and diffusion boundary layer thickness. These curves were compared against results of the computational model to benchmark the model; there were not significant discrepancies between the experimental polarization curves and computational results. For the RCH cell, researchers benchmarked overpotential distribution along the cathode for copper deposition. In addition, from the comparison of cathodic polarization curves on copper deposition, nickel deposition, and copper–nickel codeposition, the project team confirmed that the RCH cell can provide useful experimental data for codeposition.

Planned Activities

In the final year of the project, the electrochemical reaction model will be extended from two to three dimensions and validated via comparison with both molten salt and surrogate aqueous system experimental data. Improvements will be incorporated back into the model.

For the experiments with molten salts, researchers will further analyze data from uranium and zirconium systems to more accurately determine diffusion coefficients. Also, the team will compare calculated reduction potentials with thermodynamic data in the super-cooled state, using the findings to calculate activity coefficients in the LiCl-KCl molten salt. This will be followed by examination of the behavior of zirconium and uranium together in a $ZrCl_4$ -UCl_3 -LiCl-KCl molten salt and collection of pure zirconium at the electrode surface from the salt mixture.

For the surrogate aqueous system experiments, the team will perform codeposition experiments using an RCH cell with a mixture solution of CuSO4 and NiSO4. The overpotential distribution along the cathode will be measured, and the composition of deposited alloy will be analyzed to investigate local partial current density.

With the 3D model fully developed, researchers will use it to develop an innovative process for zirconium recovery—including both decontamination of zircaloy cladding hulls and removal of zirconium contamination from an electrorefiner used to process U-Zr fuel.

Advanced Instrumental Science-Based Approach to Nickel Alloy Aging and Its Effect on Cracking in Pressurized Water Reactors

Project Number: 2010-002-K

PI (U.S.): Chi Bum Bahn and Ken Natesan, Argonne National Laboratory

PI (ROK): Ji Hyun Kim, Ulsan National Institute of Science and Technology

Collaborators: Korea Atomic Energy Research Institute, Seoul National University

Program Area: Reactor Concepts RD&D

Project Start Date: December 2010

Project End Date: September 2013

Research Objectives

This project is investigating the effects of aging on the microstructure and stress corrosion cracking (SCC) behavior of a dissimilar metal weld—specifically, the chromium dilution effect at the weld interface formed when joining low-alloy steel (LAS) to Alloy 690 with an Alloy 152 weld filler. To characterize the weld interface, researchers are applying a variety of advanced analytical techniques, including energy dispersive X-ray spectroscopy (EDS), secondary ion mass spectroscopy (SIMS), and 3D atom probe tomography (3D APT). Investigators are analyzing the oxide film that forms on weld interface samples exposed to simulated pressurized water reactor (PWR) primary water conditions using in situ Raman spectroscopy.

The project is also conducting fundamental studies of the effect of lead (Pb) on Alloy 600 steam generator (SG) tube corrosion/cracking. Using synchrotron X-ray reflectivity, the team will obtain molecular-scale details of lead distribution and chemical states at the metal oxide–liquid interface. These details will be compared with results from in situ electrochemical spectroscopy and typical ex situ oxide film analysis. In situ impedance spectroscopy will be used to study oxide film properties at high temperatures and how the presence of dissolved lead changes those properties. Conventional ex situ surface oxide film analysis and EDS techniques will determine the chemical composition profile.

Research Progress

The project team has produced a representative dissimilar weld mock-up of Alloy 690/Alloy 533 Gr. B with Alloy 152 weld metal filler. This material has been aged in furnaces under accelerated temperature conditions, and two weld samples have been sent to Ulsan National Institute of Science and Technology (UNIST) for analysis: a 35 mm thick sample aged at 450°C for the equivalent of 60 years, and a 17 mm thick sample aged at 400°C for the equivalent of 15 years. One more sample is in the furnace at 400°C to achieve a 30- year equivalent aging time.

The UNIST team characterized the as-welded and aged dissimilar metal weld samples using EDS, SIMS, and 3D APT, and obtained micrographs using a scanning electron microscope (SEM) and transmission electron microscope (TEM). They found that the thermal aging concentrates chromium and nickel in the weld root region. The concentrated chromium contributes to growth and formation of chromium precipitates in the fusion boundary region (see Figure 1).

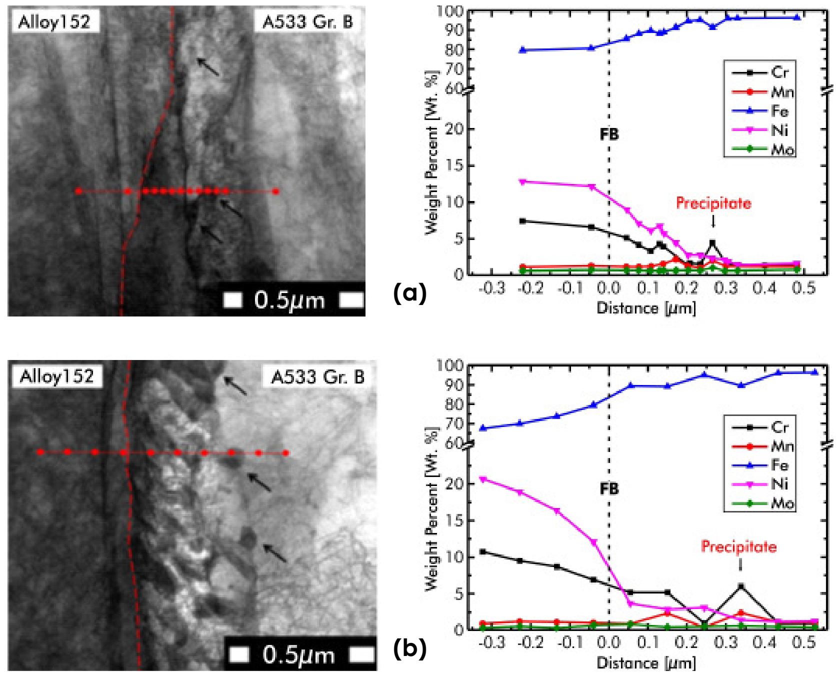


Figure 1. Dark field TEM micrographs and EDS in the weld root region of (a) the as-welded dissimilar metal weld and (b) the weld after aging an equivalent of 30 years.

The team has completed SCC crack growth rate (CGR) testing on the non-aged Alloy 152 weld. Researchers measured a crack growth rate of 1.8E-11 m/s at K = 28 MPa√m. CGR testing is in progress on the Alloy 690 heat-affected zone and the weld–LAS interface. So far, researchers have observed small SCC crack growth rates in the range of 1E-12 m/s in the Alloy 690 heat- affected zone specimen. SCC testing on the weld–LAS interface is investigating the effect of chromium dilution on Alloy 152's SCC behavior. Using the weld mock-up, the team machined a specimen aligned along the Alloy 152 butter–Alloy 533 LAS interface. SCC crack growth rate measurements to date are 1.1E-11 m/s at K = 26 MPa√m.

The project team performed the *in situ* Raman spectroscopic analysis of the film oxide on as-welded and aged dissimilar metal weld interfaces in a primary water environment, increasing the test temperature from 25°C to 300°C. Figure 2 shows the *in situ* Raman spectra for as- welded and aged interfaces heated to 300°C. Using SEM–EDS, researchers also characterized the specimen *ex situ* and compared to the *in situ* results to enhance understanding. The chromium oxide peaks were observed in both the as-welded and aged samples. The team observed the spinel peak of $NiCr_2O_4$ in the as-welded sample, while $Fe_xCr_{3x}O_4$ and $NiFe_2O_4$ were observed in the aged sample.

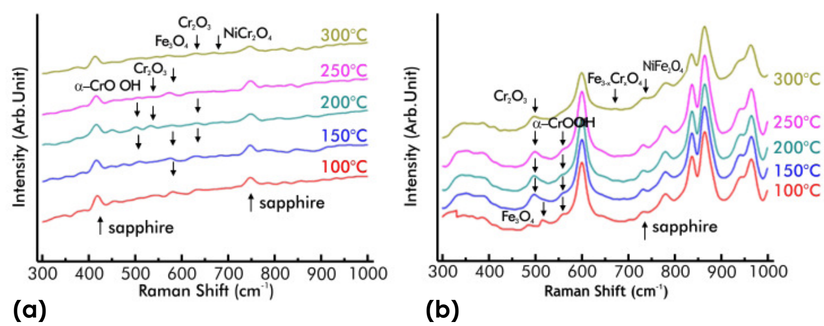


Figure 2. *In situ* Raman spectra for (a) as-welded and (b) aged dissimilar metal weld interfaces heated up to 300°C.

Researchers at Argonne National Laboratory (ANL) sought to improve the surface crystallinity of Ni(110) as measured via X-ray reflectivity. Using the 33ID-E beamline in ANL's Advanced Photon Source (APS) facility, the project team performed surface treatment by electropolishing and argon sputtering/ annealing in an ultra-high vacuum chamber. The surface condition was examined using reflection high-energy electron diffraction (RHEED) before and after each sputtering/annealing cycle. Figure 3a provides a RHEED image after the sputtering/annealing cycles. The image shows multiple parallel streaks, which are typically observed with a practically flat metal surface. After the successful surface pre-treatment, the team took X-ray measurements at room temperature using the APS 5ID-D beamline. Figure 3b shows the specular crystal truncation rod (CTR) data on the Ni(110)–water interface at room temperature, demonstrating that this treatment procedure allows scientists to measure CTR data with small errors even in lower-intensity regions.

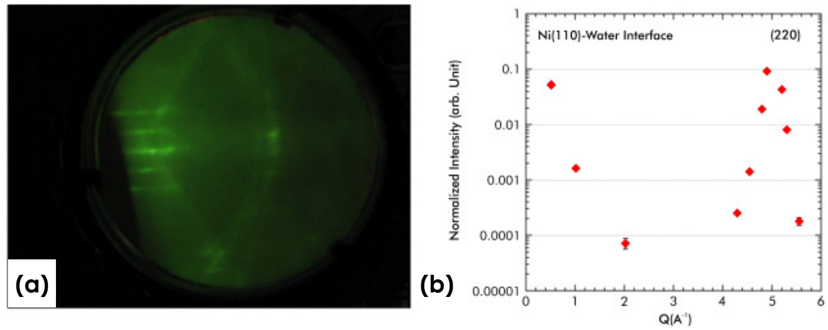


Figure 3. (a) RHEED image of Ni(110) surface after sputtering/annealing cycles and (b) specular crystal truncation rod data of Ni(110)–water interface at room temperature.

Figure 4 presents the results of the TEM– EDS analyses for the specimens tested in (a) 10 and (b) 1000 parts per million (ppm) PbO + 0.1 M NaOH solutions at 315°C. Nickel-rich oxide in the outer layer and relatively chromium-rich oxide in the inner layer formed in the unleaded 0.1 M NaOH solution. In the 0.1 M NaOH solution containing only 10 ppm PbO, researchers observed a duplex oxide layer similar to the oxide layer in the unleaded solution. Lead in the oxide was below 1 wt%, while the nickel was slightly depleted in the outer oxide layer compared with the oxide composition formed in unleaded solution. When the PbO increased to 1000 ppm, researchers observed nickel depletion in the outer nickel-rich oxide, and the inner chromium-rich oxide no longer appeared. In a 1.1 M NaOH + 5000 ppm PbO solution, nickel depletion in the oxide was more severe, and the amount of lead incorporated into the oxide greatly increased. The project team conducted electrochemical impedance and found that increased PbO content in the solution degraded the passivity of oxide formed on Alloy 600, as well as led to oxide change in chemical composition. Oxide is regarded as an imperfect capacitor analogous to the parallel electrical circuit of an ideal capacitor and resistance.

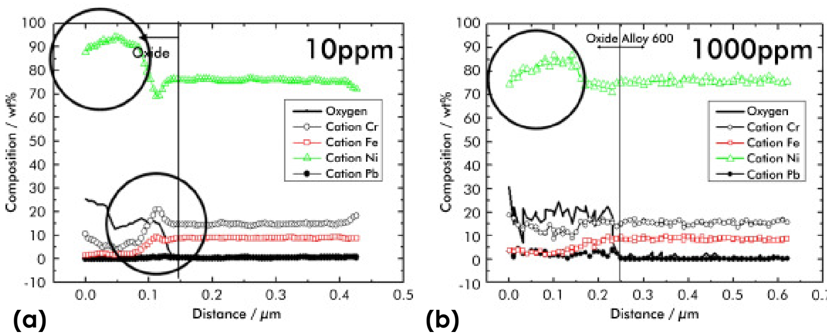


Figure 4. TEM–EDS analyses of the surface oxide layer formed on the thermally treated Alloy 600 specimens in aqueous solutions at 315°C as a function of PbO content: (a) 10 and (b) 1000 ppm in 0.1 M NaOH.

During oxide formation and growth, a charge transfer and ionic transport is involved. When PbO is introduced into a solution, oxide becomes more defective, increasing the charge carriers and allowing easy ionic transport. Consequently, the passivity is degraded with the PbO content, which leads to SCC susceptibility.

Planned Activities

The project team will continue aging dissimilar weld mock-up samples. By April 2013, the U.S. team will provide UNIST with a 17 mm thick weld sample aged at 400°C for the equivalent of 30 years. The team will characterize this specimen and conduct in situ oxide film analysis. Researchers will complete SCC CGR testing on the heat-affected zone of Alloy 690 and the weld–LAS interface.

Researchers will conduct synchrotron X-ray experiments to characterize the surface interface between Ni(110) and pure water at room temperature. Similar X-ray testing is planned to identify the interface between Ni(110) and solution containing lead.

The team will also perform high-temperature/high-pressure testing in water as a function of immersion time. After the high-temperature experiment, scientists will analyze the oxide microstructure using TEM–EDS. Combining information about the oxide structure and chemical composition will provide insight into oxidation kinetics in leaded solution.

Low-Loss Advanced Metallic Fuel Casting Evaluation

Project Number: 2010-003-K

PI (U.S.): Randall Fielding, Idaho National Laboratory

PI (ROK): Dr. Ki-Hwan Kim, Korea Atomic Energy Research Institute

Collaborators: None

Program Area: FCR&D

Project Start Date: December 2010

Project End Date: December 2013

Research Objectives

The objective of this project is to develop methods of minimizing fuel losses and reducing waste streams during fabrication of metallic fuel pins for sodium-cooled fast reactors (SFRs). This work will enhance the technical readiness level of the metallic fuel fabrication process. This process has three phases: (1) fuel pin casting, (2) loading and fabrication of the fuel rods, and (3) fabrication of the final fuel assemblies. Most fuel losses and waste/recycle streams occur during fuel pin casting. Recycle streams include fuel pin rework, returned scraps, and fuel casting heels, which are of special concern in the counter-gravity injection casting process because of the large masses involved. Large recycle and waste streams lower the productivity and economic efficiency of the fabrication process.

The project team will evaluate fuel losses that occur during casting of U-Zr/U-TRU-Zr (uranium–transuranic–zirconium) fuel pins for an SFR. This will be accomplished by casting a considerable amount of fuel alloy in a furnace, and then quantitatively evaluating losses in the melting chamber, crucible, and mold. After identifying and quantifying materials at key process stages, the team will develop means of reducing losses, including permanent crucible coatings, permanent reusable molds, and advanced techniques such as continuous casting. The goal is to develop a coating technology for re- usable crucibles with good thermal cycling characteristics and excellent compatibility between fuel melt and coating layer. The scope of work consists of three major tasks:

- Casting development and mold loss comparison
- Coating development and characterization
- Continuous casting technology

Research Progress

Casting Development and Mold Loss Comparison

The research team at Korea Atomic Energy Research Institute (KAERI) cast a batch of U-10Zr-5Mn and one of U-10Zr-5Ce (weight basis) on approximately a one-kilogram scale.

Chemical composition and mass balance of both batches were measured and used to determine fuel loss levels. Both batches cast well. For the U-10Zr-5Mn batch, mass balance determined that approximately 92.4% of the charge flowed into the mold assembly, while 6.1% stayed in the crucible as residue and dross, and 1.5% was lost. A greater amount of cerium-containing material was retained in the crucible as residue and dross: 88.6% of the material flowed into the mold assembly and 11.3% remained in the crucible, leaving a fuel loss of only 0.1%. Both castings were done using an yttrium oxide (Y_2O_3)-coated crucible; however, the manganese crucible was slurry-coated, and the cerium crucible was plasma-spray-coated. The difference in coating deposition technique contributed to the differences in fuel losses seen in the two casting batches. The Y_2O_3 coating applied through slurry coating was porous and not as even throughout the crucible, thus easily penetrated by the melt, leading to higher losses. In contrast, the plasma-sprayed coating used in the cerium-containing melt was uniform and dense. The increased density precluded any melt infiltration, lessening losses.

The Idaho National Laboratory (INL) team continues efforts to design a new furnace to prepare fuel pins for analysis. The furnace is sized for charges as large as 500 grams, although expected charges will be approximately 200 grams. The current mold is designed to cast three pins, each 4.3 mm in diameter and 250 mm in length. Both the mold and crucible are independently induction-heated with 10 kW power supplies and digital furnace controls. This furnace was thoroughly tested using a copper–nickel alloy surrogate for the U-Pu-Zr alloy. Testing included actual casting and fluidity testing.

Historically, fuel casting has been operator-dependent, requiring the operator to know the proper "look" of the melt based on agitation, dross layer, color, etc. This method of casting does not lend itself to being repeatable. Therefore, the project team aims to better quantify the molten material fluid properties through fluidity testing, a common foundry practice of pouring the molten material into a mold of consistent cross section and then measuring how far the molten material flows. The distance the material flows is based on properties such as viscosity, surface tension, heat transfer, temperature, and oxide formation. These properties all combine into the somewhat qualitative term "fluidity."

To conduct fluidity testing on the casting furnace, project researchers draw the molten alloy up into a quartz tube using a predetermined and repeatable pressure differential. The researchers tested a range of copper–nickel alloys from pure copper to a composition of 70 wt% nickel and 30 wt% copper. For each test, they applied the same amount of superheat and the same pressure differential. Table 1 (page 67) and Figure 1 (page 68) show the resulting flows.

During this testing, the project team incorporated several design changes in the furnace that will help ensure a smooth transition to glovebox operation and successful casting of minor actinide-bearing fuels. The project team also continued efforts to prepare the glovebox for next steps. Team members have removed the old casting furnace, making space for the new furnace, and installed needed power and instrumentation feedthroughs.

Table 1. Resulting values from copper–nickel fluidity tests.

Run ID	Super Heat (°C)	ΔP (KPa)	Force (N)	Flow length (cm)	Length/ force (cm/N)	Average length/ force	Standard Deviation
Cu 3	45	57.47	113	34.7	0.307	0.321	±5.8%
Cu 4	43	56.19	110	36.8	0.334		
Cu 10Ni 2	48	57.73	113	33.2	0.293	0.297	±1.9%
Cu 10Ni 3	43	57.08	112	33.7	0.301		
Cu 20Ni 2	47	56.87	112	23.2	0.208	0.261	±17.7%
Cu 20Ni 3	43	57.12	112	31.7	0.283		
Cu 20Ni 4	44	56.96	112	32.7	0.292		
Cu 30Ni 2	55	57.09	112	24.5	0.219	0.211	±5.1%
Cu 30Ni 4*	47	56.86	112	22.7	0.203		
Cu 30Ni 5**	46	56.60	111	35.4	0.319	0.310	±3.9%
Cu 30Ni 5†	43	57.29	112	33.9	0.301		

* During the melting cycle, the melt was vibrated for 30 seconds at the end of the 8-minute hold.

** The melt was vibrated for 30 seconds at the onset of copper melting. After holding at final temperature for one minute, the melt was vibrated for one minute.

† The melt was vibrated continuously throughout the 8-minute hold and injection.

cm = centimeters

KPa = kilopascals

N = Newtons

Coating Development and Characterization

During the project's first year, researchers conducted dip testing of refractory materials into a U-Zr melt at 1600°C for 15 minutes. Results showed significant interaction between ZrO , HfC, and ZrC resulting in coating failures; while Y O , TaC, and TiC had more promising outcomes. Based on these results, the KAERI project team produced several multilayer samples incorporating the Y_2O_3 paired TiC, TaC, and ZrC, producing coatings for niobium and Y_2O_3 , exposing the more chemically inert Y_2O_3 to the molten U-Pu-Zr melt. These samples have been provided to the INL team for molten alloy exposure testing.

Continuous Casting

The KAERI team repaired and reconstructed an existing engineering-scale continuous casting apparatus. Researchers designed and machined a melting crucible and a casting mold, as well as replacing an induction coil, a cooling jacket for the mold, and a withdrawal servo-motor for the driving roller. The team conducted successful initial testing under an inert atmosphere at approximately 760 torr with copper as a surrogate for uranium, based on similar melting temperatures. The researchers cast and characterized a fuel pin that was 7 mm in diameter and 2300 mm long. Visual examination showed the surface finish to be smooth and generally free of defects. The researchers then used X-ray radiography to interrogate the pin's integrity. A number of small defects were found on the tail end of the casting, but in general the pin was sound. Although a significant amount of dross was retained in the crucible, overall losses were on the order of 0.1%.

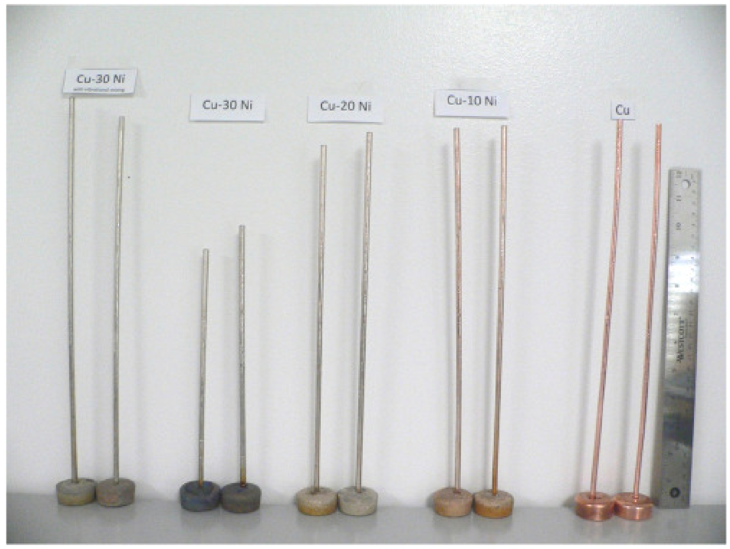


Figure 1. Selected fluidity test results. The far right tests incorporated vibrational mixing, while the other tests did not.

Planned Activities

Future activities will focus on casting development, including minor actinide (MA)-bearing alloys and crucible coating material development. INL plans to install the newly designed furnace into a minor actinide-capable glovebox, after which casting development will continue. Casting will include U-Zr, U-Pu-Zr, and U-Pu-Zr-MA. INL and KAERI researchers will identify and compare loss results, paying special attention to fuel losses and fuel loss paths in minor actinide-bearing batches. Both teams of researchers will conduct literature searches to identify potential coating candidates for testing in a U-Pu-Zr environment. Continued coating studies will include intermediate layers and optimized parameters, with KAERI producing samples for INL testing. If funding is available, the INL team will conduct americium loss testing and fluidity testing of U-Zr alloys.

Development and Characterization of Nanoparticle-Strengthened Dual-Phase Alloys for High-Temperature Nuclear Reactor Applications

Project Number: 2010-004-K

PI (U.S.): Thak Sang Byun and David T. Hoelzer, Oak Ridge National Laboratory

PI (ROK): Ji-Hyun Yoon, Korea Atomic Energy Research Institute

Collaborators: None

Program Area: FCR&D

Project Start Date: December 2011

Project End Date: November 2013

Research Objectives

The main objective of this research is to develop a nanoparticle-strengthened dual-phase steel with high fracture toughness for application to high-performance reactor core structures. To achieve this objective, the project will combine two advanced materials processing technologies used to produce nanostructured ferritic alloys (NFAs) and ferritic/ martensitic (F/M) dual-phase steels. To optimize post-extrusion thermomechanical treatment (TMT) conditions, the team will carry out both computational simulation for phase equilibrium and basic microstructural and mechanical characterizations for base and TMT materials.

This project also aims to produce a material characterization database for the optimized materials. At a minimum, the mechanical information will include tensile deformation data, fracture toughness data, and high-temperature deformation (including creep) data. The microstructural characterization will include grain structure, phase transformation and stability data, and high-resolution structure analysis data for the distribution and thermal stability of nanoparticles.

Research Progress

Research in the past year focused on process development and characterization of two oxide dispersion-strengthened (ODS) 9Cr alloys, designated 9YWTV-PM1 for the high-carbon heat and 9YWTV-PM2 for the low-carbon heat.

Process Development

Researchers at both Oak Ridge National Laboratory (ORNL) and at the Korea Atomic Energy Research Institute (KAERI) applied a variety of post-extrusion TMTs to the as- extruded (consolidation-processed) 9YWTV-PM1 and 9YWTV-PM2 alloys. To optimize the isothermal annealing condition, specimens were annealed at or near the intercritical temperatures (830°C–1000°C) for 30 minutes to 20 hours. This thermal annealing temperature range was determined according to the results of thermal stability calculation for phases performed in the last fiscal year. Further, researchers applied more intensive TMT, i.e., controlled hot-rolling for 20% or 50% reduction at 900°C–1000°C, to induce extensive deformation-recovery mechanisms in phase transformation condition. The research team characterized the thermomechanically treated materials to determine their microstructural and mechanical properties.

Microstructural Characterization

To confirm the differential scanning calorimetry results for the phase transformation temperature range and to provide basic feedback data for TMT optimization, the research team performed *in situ* high-temperature X-ray diffraction analysis (XRD). Figure 1 displays the volume fraction data for 9YWTV-PM1 and -PM2 for two phases: face-centered cubic (fcc, austenite) and body-centered cubic (bcc, ferrite). The fcc volume fraction was determined from the ratio of diffraction peak intensities for the fcc and bcc phases and the R-value for each phase. After annealing at 1000°C for 0–240 minutes, for example, the volume fraction of fcc in 9YWTV-PM1 gradually increased up to about 65% as a function of time.

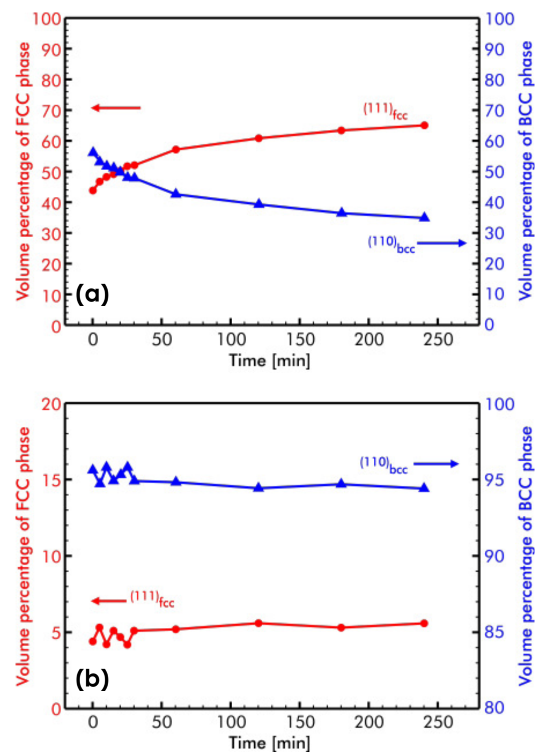


Figure 1. The volume fractions of the fcc phase (red) and bcc phase (blue) in (a) 9YWTV-PM1 and (b) 9YWTV-PM2 after heat treatment at 1000°C for up to 240 minutes.

As seen in Figure 1b, however, the volume fraction of fcc in 9YWTV-PM2 appeared to saturate at only about 5%. This limited phase transformation in the PM2 alloy might be due to the lower carbon content in ferrite, which should lead to an early depletion of carbon content available for the transformation.

Mechanical Characterization

Researchers carried out uniaxial tension testing for the as-extruded specimens and for the post-extrusion heat-treated specimens exposed to 975°C for 1 hour. The ultimate tensile strength of the base material at room temperature was 1.77 GPa. The alloy retained a relatively high strength up to 700°C, although there was a sudden drop in strength above 500°C, as shown in Figure 2. The measured uniform elongations were approximately 1% over the test temperature range. The partial phase transformation heat treatment had significant effects on the as-extruded 9YWTV-PM1, which showed increased strength except for the 700°C test. The uniform elongations also increased after heat treatment.uniform elongations also increased after heat treatment.

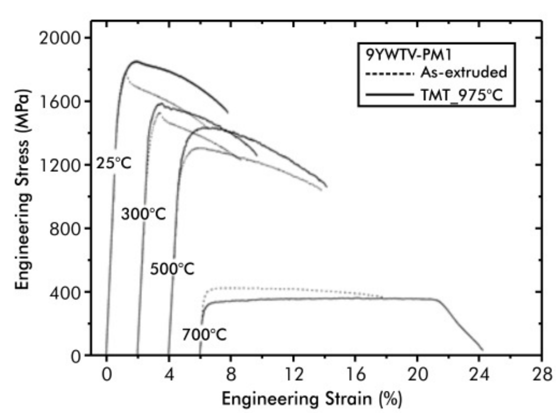


Figure 2. Stress-strain curves for 9YWTV-PM1 obtained from tests at various temperatures.

The primary objective for the fracture toughness testing campaign was finding the best TMT conditions. Compared to the as-extruded material and other NFAs, the fracture toughness after heat treating increased significantly. After annealing at 900°C, the 9YWTV-PM2 at room temperature had K_{JQ} values higher than 200 MPa√m, which was more than a 60% increase on average.

While thermal annealing greatly improved fracture toughness for 9YWTV-PM2, controlled hot-rolling resulted in even more significant increases, as indicated in Figure 3. Both the 20% and 50% rolling treatments at 900°C almost doubled the alloy's K_{JQ} to a value as high as the non-ODS F/M steels. The specimens with 50% rolling at 975°C and 1000°C showed similar K_{JQ} increases throughout the test temperature range, except at room temperature and 700°C. Controlled hot-rolling by 50% at 900°C showed particularly impressive toughness improvement: K_{JQ} was above 200 MPa√m over a wide temperature range up to 500°C and exceeded 150 MPa√m at 500°C–700°C.

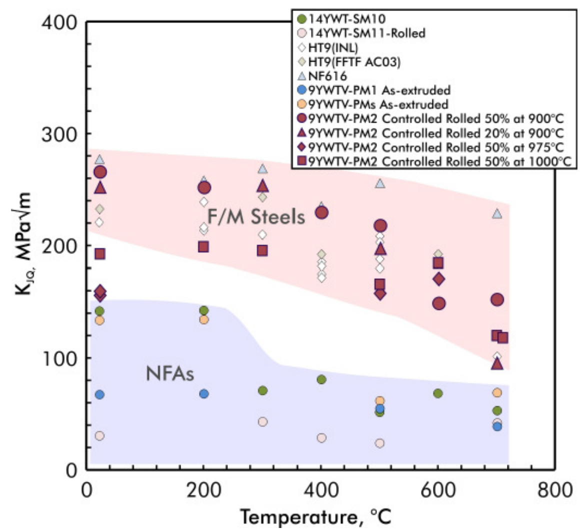


Figure 3. Fracture toughness data of 9YWTV-PM2 after hot-rolling at 900°C, 975°C, and 1000°C, compared to data for F/M steels and other NFAs.

The above characterization results are summarized as follows:

•Higher carbon content in a NFA does not improve fracture toughness.

•Simple annealing treatments performed in or around the intercritical temperature region have significant but limited effects on the fracture toughness.

•Controlled hot-rolling is the most effective measure to improve fracture toughness.

Planned Activities

The research activities in the last project year (December 2012–November 2013) will focus on detailed characterization of the NFAs treated in optimized post-extrusion process conditions. The researchers will perform the following activities:

- Treat the lower-carbon material, 9YWTV-PM2, in optimum conditions for multistep hot-rolling, i.e., 50% reduction at 900°C–1000°C, and machine the specimens for full characterization. The final product may be provided to other research programs for more detailed scientific studies.
- Perform systematic mechanical characterization, including tensile, fracture toughness and high-temperature deformation (or creep) tests over a wide range of temperatures up to 700°C, to build up a database for the final product.
- Carry out detailed microstructural characterization for select optimized materials using scanning electron microscopy, electron backscatter diffraction, X-ray analysis, high-resolution transmission electron microscopy and similar analytical technique to quantify the number of constituent phases and stability of nanoclusters. The phenomenon of slower-than-expected phase transformation in the intercritical temperature region will be also explored further.
- Propose the next stage of research following the development of high-toughness NFAs, which may include irradiation experiments, weldability studies, and/or fabrications in tube form.

Atomic Ordering in Alloy 690 and its Effect on Long-Term Structural Stability and Stress Corrosion Cracking Susceptibility

Project Number: 2011-001-K

PI (U.S.): Michael Kaufman, Colorado School of Mines

PI (ROK): Young Suk Kim, Korea Atomic Energy Research Institute

Collaborators: University of Michigan, University of North Texas

Program Area: Reactor Concepts RD&D

Project Start Date: November 2011

Project End Date: November 2014

Research Objectives

The objective of this project is to determine the effects of atomic ordering in Alloy 690 under pressurized water reactor (PWR) operating conditions on mechanical integrity and susceptibility to intergranular stress corrosion cracking (SCC). Nickel-based Alloy 690 is being considered as a possible replacement for Alloy 600 in reactor structural components due to its improved resistance to primary water SCC. However, thermally treated Alloy 690 subjected to 20% to 30% cold working has demonstrated crack growth rates as high as those of Alloy 600 during testing in high-temperature deaerated primary water.

Alloy 690 is prone to the formation of multiple chemical ordering reactions leading to the formation of embrittling precipitates, such as Ni_2Cr type ordering at 420°C and Fe_3Ni and Ni_3Fe type ordering at even lower temperatures. These ordering reactions involving nickel, chromium and iron atoms—the main components of Alloy 690—occur under exposure to light water reactor (LWR) operating conditions such as high temperatures and stresses. Such reactions adversely affect the long-term structural stability and mechanical properties of this alloy. Thus a fundamental understanding of these ordering reactions and their effects may be a critical factor in governing the lifetime of the structural reactor components made of Alloy 690.

Research Progress

Researchers at the Korea Atomic Energy Research Institute (KAERI) conducted a range of tests on samples of Alloy 690 they had subjected to thermomechanical treatments (Table 1) and provided some of the resulting materials to the U.S. research teams for testing as well. Tests included differential scanning calorimetry (DSC), neutron diffraction, transmission electron microscopy (TEM), and tensile testing. As Figure 1 shows, the DSC peaks—indicating changes in the alloy—occur between 400°C and

Table 1. Thermomechanical treatments conducted at KAERI.

Alloy 690 (8.64wt% Fe)	
Sample No.	Treatments
1	SA + WQ
2	SA + WQ + CW
3	SA + WQ + TT
4	SA + WQ + TT + CW
5	SA + WQ + Aged
6	SA + WQ + CW + Aged
7	SA + WQ + TT + Aged
8	SA + WQ + TT + CW + Aged

SA (solution annealed) = 1100°C-1h

TT (thermally treated) = 700°C-17h

CW = 20% cold worked

WQ = water quenched

600°C, corresponding to the temperature range of interest in PWRs. The results also indicate considerable differences in behavior depending on how the material was cooled from elevated temperatures and whether it was cold worked. The neutron diffraction results (Figure 2) indicate that lattice contraction occurs during aging at 400°C, presumably resulting from the development of short- range order (SRO) in the alloy during these low-temperature anneals. The tensile tests were conducted as a function of temperature (Figure 3); observations suggest that Alloy 690 properties are sensitive to temperature where serrated flow associated with dynamic strain aging (DSA) is apparent. In short, the KAERI team confirmed the adverse effects of PWR conditions on this alloy and the necessity of developing better fundamental understanding.

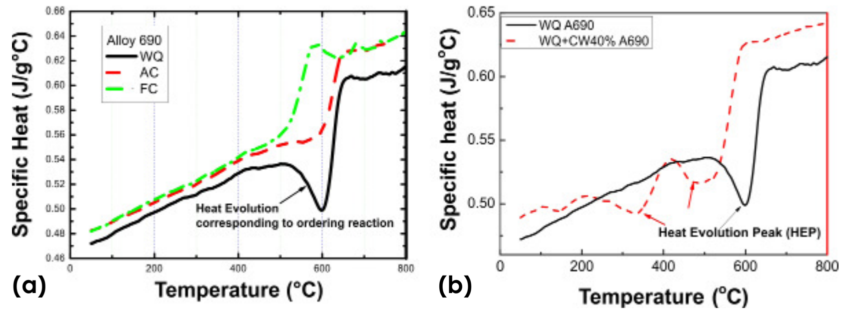


Figure 1: DSC analysis of Alloy 690 (a) after cooling at different rates from the solution annealing temperature and (b) after cold working.

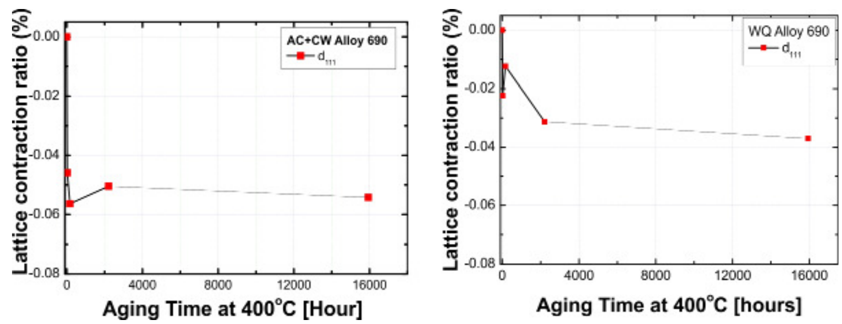


Figure 2. Neutron diffraction results showing a lattice contraction on ordering, enhanced with aging time and cold working.

Based on the preliminary results described above and previous literature, researchers at the University of Michigan initiated procurement of two alloys with different iron concentrations to determine the influence of iron on the ordering in Alloy 690. Thus far, the research team has identified and obtained sections of heat WP787 from Valinox, deemed to have a microstructure typical of Alloy 690 currently used as steam generator tubing.

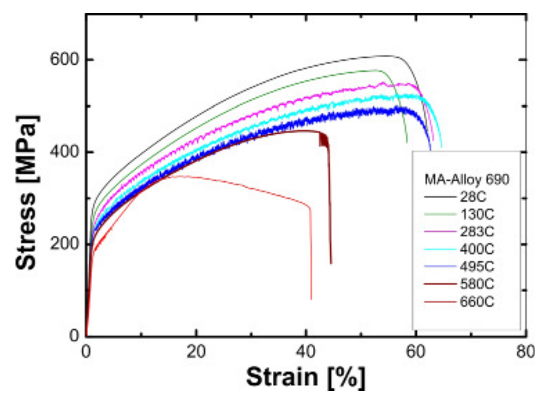


Figure 3. Stress-strain curves of Alloy 690 at different temperatures.

In parallel with these efforts, Colorado School of Mines (CSM) researchers are using advanced diffraction and imaging techniques in the TEM to examine KAERI-provided samples. The purpose is to confirm the above-noted assumption that SRO is occurring and, if so, determine the nature of the SRO and its dependence on both composition and thermomechanical history. Thus far, results indicate diffuse intensity in certain selected area diffraction patterns associated with SRO (see Figure 4).

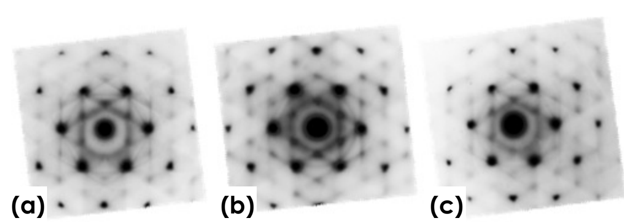
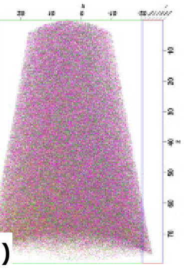
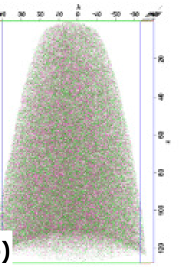


Figure 4. ⟨111⟩ SAD patterns of Alloy 690 showing the presence of diffuse peaks suggesting SRO: (a) SA+WQ+TT, (b) SA+WQ + aged at 475°C–3000h, and (c) SA+WQ+TT + aged 475°C–3000h.

To examine the nature of SRO, the project team must use higher-resolution techniques to correlate the diffuse intensities with some sort of structural characterization. Therefore, researchers at the University of North Texas performed 3D atom probe tomography (3D APT) on both Ni_2Cr and INCONEL 690 (10.21 wt% iron), aged for 8,000 and 3,000 hours, respectively. Visual inspection of the 3D reconstructed atomic images (Figure 5) revealed no precipitates or clustering in either system. However, preliminary results of a quantitative frequency distribution analysis indicate an 11% increase in cluster density in the INCONEL 690 specimen when compared to the Ni_2Cr binary. In the INCONEL 690 specimen, the average cluster composition appears to be chromium-enriched, although other complementary techniques (TEM, X-ray diffraction [XRD], and neutron diffraction) are needed to confirm this clustering and further explore the nature of the SRO.



(a)



(b)

(c)

Average Cluster Composition	
Element	at%
Cr	70.16
Ni	25.14
Fe	3.63
C	0.04
Al	0.30
Si	0.09
Mn	0.56
Ti	0.08

Figure 5. 3 DAPT reconstructions of (a) Alloy 690 and (b) Ni_2Cr, with (c) the corresponding compositional analysis of the clusters obtained using a cluster analysis algorithm.

Planned Activities

In FY 2013, the project team will continue efforts to characterize aged Alloy 690, such as conducting constant extension rate tests in high-temperature water to determine the influence of thermomechanical history on SCC initiation resistance. Researchers will also generate data to examine SCC initiation time and growth rate, atomic-scale data of ordered phases, and lattice spacing. This work largely involves performing tensile tests and SCC tests, correlated with detailed characterization using micro- and nano-hardness, scanning electron microscopy, small- angle neutron scattering, 3D APT, DSC, TEM, and XRD. In FY 2014, the project team will combine results from testing and characterization, creating a foundation to develop suggestions for improving Alloy 690 for use in PWRs.

Development of Microcharacterization Techniques for Nuclear Materials

Project Number: 2011-002-K

PI (U.S.): Peter Hosemann, University of California, Berkeley

PI (ROK): Chansun Shin, Korea Atomic Energy Research Institute

Collaborators: Los Alamos National Laboratory

Program Area: Reactor Concepts RD&D

Project Start Date: October 2011

Project End Date: September 2014

Research Objectives

This project will systematically investigate small-scale materials testing of structural materials for nuclear application, evaluating the full potential of these methods and standardizing the techniques for irradiated materials. The goal is to develop micro- and nanoscale mechanical testing techniques for irradiated materials (e.g., tensile strength, compressive strength, and creep) that will allow researchers to assess changes in mechanical properties on the macro scale.

Degradation of materials properties under neutron irradiation is a key issue limiting the lifetime of nuclear reactors. Evaluating the property changes of materials due to irradiation and understanding the role of microstructural changes on mechanical properties are required to ensure reliable long- term reactor operation and to develop high-dose concepts. Researchers have been conducting post-irradiation mechanical testing on bulk specimens for decades, but the task remains time- and cost-intensive. It is also challenging, given the need to handle large quantities of radioactive materials. While ion beam irradiation does not result in activation of the materials, the irradiated volume is often too small for conventional mechanical testing.

Small-scale materials testing (SSMT) has recently been applied to nuclear materials and shows potential to address these issues. SSMT is useful for both bulk sample irradiation in reactors and spallation sources, as well as ion beam irradiations. Micromechanical testing reduces material radioactivity and allows probing shallow ion-irradiated surface layers. Micro- and nanoindentation allow sampling an entire cross section. While initial studies are promising, this technique is far from being fully developed or standardized. Developing these techniques as part of an international collaboration is significant, as it is a step towards global acceptance of a more standardized and unified approach in mechanical testing for nuclear materials.

This project consists of the following major tasks:

- Fabricate micro/nanoscale testing samples, and conduct initial multiscale testing of the unirradiated materials.
- Perform ion beam irradiation of the materials.
- Conduct materials testing of the ion beam-irradiated samples, and compare results with computer models.
- Conduct small-scale materials testing on materials irradiated in a reactor.

Research Progress

During the first year of the project, the research team fabricated and tested an assortment of micro-sized samples for testing by indentation, compression, and cantilever techniques. The key question for all of these is relating micro or nano test results to macroscopic material properties. To account for differences in equipment, techniques are being cross-calibrated between the different laboratories, with some identical samples being tested in both the United States and Korea.

Results to date indicate that SSMT is a valuable technique for obtaining mechanical property measurements on irradiated materials and has great potential for studying the basic science phenomena behind radiation-induced effects in materials. Small-scale measurements allow probing mechanical property changes caused by ion irradiation.

Figure 1. Photograph of micro-indenter in operation.

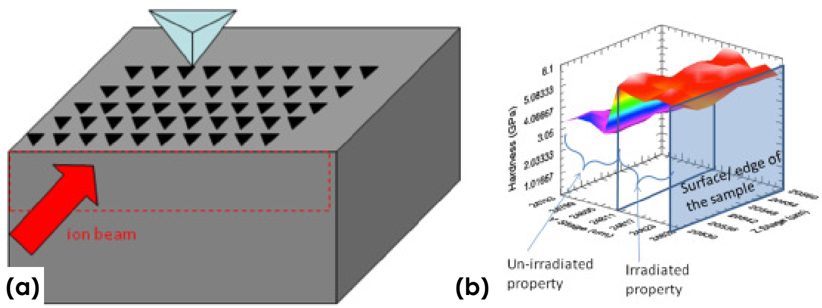


Figure 2. (a) Cross-section indentation allowing sampling of entire dose profile; (b) graphical results of 10 dpa proton-irradiated 304SS sample at 450°C.

Nanoindentation

Hardness values are difficult to determine using surface indentation. However, if irradiated and unirradiated areas are compared simultaneously in the same sample, the size effect and surface damage can be taken out of the equation. Deeper irradiation and cross-section nanoindentation (see Figure 2) can avoid some of these issues. Researchers performed cross-sectional indentation on 304SS samples irradiated at 10 dpa with 2 MeV protons.

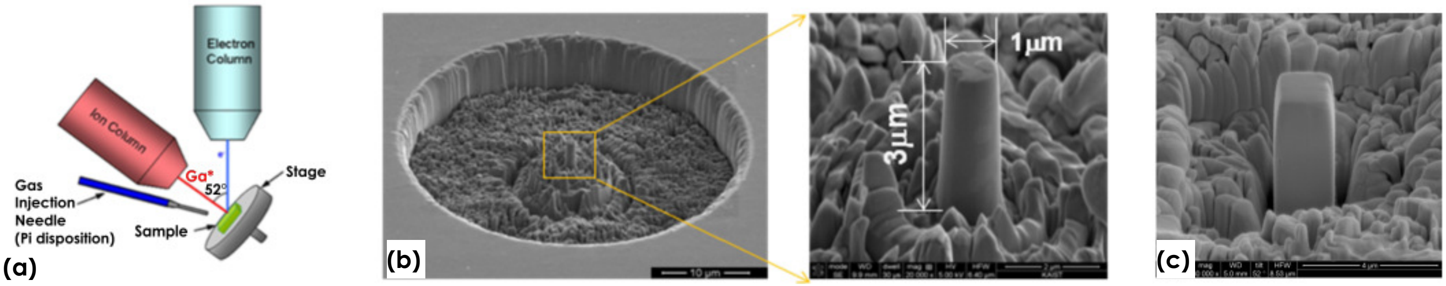


Figure 3. (a) Focused ion beam setup showing (b) milling of round micro-pillar and (c) micrograph image of square pillar.

Microcompression

The research team fabricated micro-pillars for compression testing using several techniques, including focused ion beam (FIB) milling, electrical discharge machining (µ-EDM), and lithography/plasma etching. Figure 3 shows samples produced by FIB. Intrinsic size factors (e.g., grain size and precipitates) and extrinsic factors (sample size) Pillar geometry also plays a role; square pillars provide better results than circular because the straighter sides result in a more uniform cross section. The stress of a tapered pillar is heterogeneous, resulting in increased stress at the topmost section due to a smaller diameter (see Figure 4).

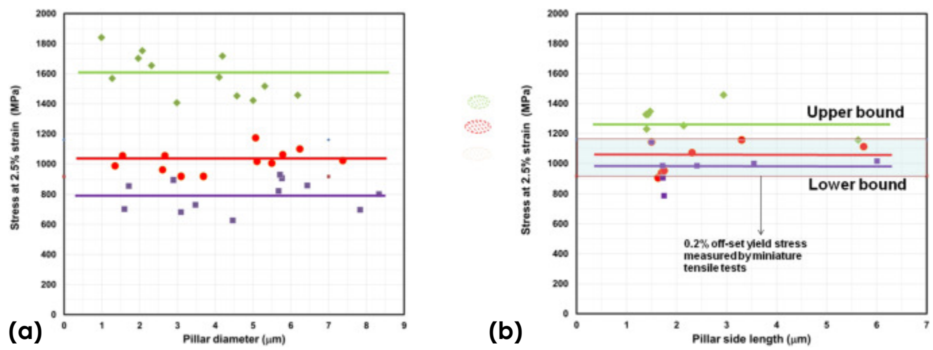


Figure 4. Stress distribution of (a) round pillar and (b) square pillar, where the upper and lower bounds are calculated with the pillar tip area and base area, respectively.

Researchers at the University of California, Berkeley, conducted micro-compression testing using a 20 µm flat diamond-tipped indenter, with a tilting goniometer stage to correct system misalignment. Indenter positioning is critical to ensure accurate measurement.

Materials investigated include oxide dispersion-strengthened (ODS) steel (Fe-15Cr-2W-1.2 Ti-0.35Y O with elongated fine grains of 450 nm), nanostructured austenitic stainless 2 3 steel (Fe-16Cr-10Ni-0.1Nb-0.02C with grain sizes from 0.3 to 2 µm), and 3C-silicon carbide (1 to 10 µm grain size and many twin boundaries). Results for ODS steels are very close to the macroscopic property values. For ultrafine-grained (UFG) stainless steels, a critical value is reached as a function of pillar size and grain size; smaller pillars result in a larger fraction of surface grains, which are inherently weaker than internal grains. Silicon carbide testing resulted in catastrophic brittle fractures, except for plastic deformation at 0.65 µm, which is consistent with theory.

The first *in situ* tests of ion beam-irradiated ODS pillars (14Cr ODS alloy) have been completed; results showed no difference in yield strength in the low-dose region compared to bulk properties.

Bend-bar testing

Microscale bend-bar testing on brittle materials is designed to gain information on fracture stress, strain, and fracture

toughness on notched samples. Depending on location, two geometries can be cut with FIB: a "house-shaped" cross section on the surface versus a square cross section

on a corner or edge. In situ testing is more difficult than ex situ, as the former requires use of a scanning electron microscope (SEM).

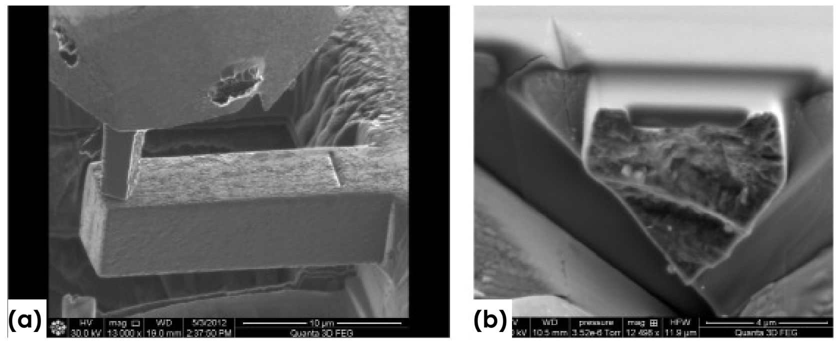


Figure 5. (a) Bend-bar test setup and (b) fracture surface of *ex situ* bend-bar test of SiC sample.

Planned Activities

In FY 2013, the second project year, the team expects to begin testing irradiated samples and expand their investigation to additional types of materials:

- Complete the unirradiated sample testing and publish the findings.
- Conduct ion beam irradiation of the samples at Los Alamos National Laboratory (1 dpa, room temperature).
- Investigate different sample materials, including other ODS alloys and ferretic/ martensitic steels.

Verification and Validation of High-Fidelity Multi-Physics Simulation Codes for Advanced Nuclear Reactors

Project Number: 2011-003-K

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

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

Collaborators: None

Program Area: Reactor Concepts RD&D

Project Start Date: October 2011

Project End Date: September 2014

Research Objectives

The objective of this project is to verify and validate a suite of high-fidelity multi-physics simulation methods and codes developed during a previous I-NERI project. The team compares results against Monte Carlo solutions and experimental measurements for light water reactor (LWR) and very high-temperature reactor (VHTR) cores. The main focus for FY 2012 was to refine and update the DeCART neutronics code in terms of capabilities and accuracy, to provide appropriate and realistic benchmark problems for reactor types of interest and perform verification tests with developed benchmark problems, and to update the cross-section libraries that are a critical factor in obtaining accurate solutions and consistent comparisons. Both the U.S. and ROK teams use the DeCART code, but apply different computational fluid dynamics (CFD) codes: STAR-CCM+ at Argonne National Laboratory (ANL) and CORONA at the Korea Atomic Energy Research Institute (KAERI).

Research Progress

Verification and Validation of the DeCART Neutronics Code for the VHTR

The project team has prepared many numerical benchmark problems for both single-effect tests and integral-effect tests for a wide range of core configurations in terms of fuel heterogeneity, depletion, and control rod arrangement. During the past year, the team focused particularly on verifying the DeCART code using numerical benchmark problems for single- effect tests.

Using a homogeneous VHTR fuel element with reflective boundary conditions, researchers examined both the nuclide chain of the DeCART code and the multi-group cross sections, including the resonance data. Some fission product nuclides with large resonances, such as Sm-152, Sm-147, Ag-109, Nd-145, and

Eu-153, were found to cause large errors, which were significantly reduced by treating them as resonant isotopes. Large errors were also found in the multi-group cross sections of some actinides, such as Pu-239, Pu-241, U-235, Pu-240, and Pu-242, especially in the thermal energy range.

The DeCART depletion routine for burnable poison materials, especially gadolinium (Gd) isotopes, was examined for a single- block problem. Results indicate that the current predictor-corrector method produced a large error because the depletion of gadolinium was treated in the same manner as a fuel isotope. Because the reaction rates of Gd-155 and Gd-157 change non-linearly with burnup due to their large cross sections, a specific model with finer burnup steps should be used for those isotopes. Figure 1 shows the error reduction in DeCART with updated fission product resonance treatment and gadolinium depletion models.

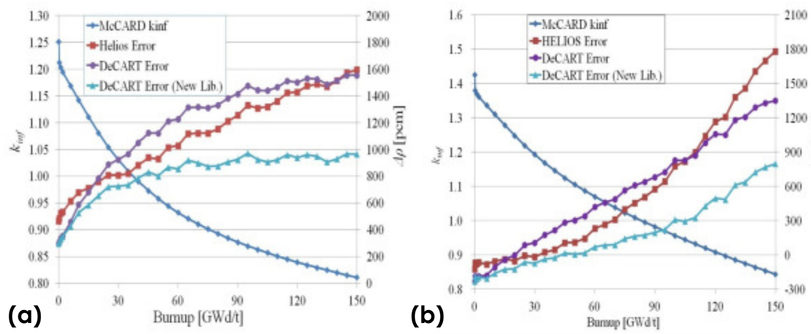


Figure 1. DeCART depletion results for the VHTR fuel elements: (a) a homogeneous fuel element and (b) a PMR-200 (VHTR) unit fuel cell.

The team performed a parametric study of the ray tracing parameters, which indicated that the four default azimuthal angles are not sufficient for either VHTR or pressurized water reactor (PWR) fuel elements. Therefore, they proposed a new azimuthal angle discretization scheme adopting Gaussian quadrature along with the corresponding modular ray tracing scheme. Numerical test results indicate that the new scheme reduces the error caused by the azimuthal angle discretization by a factor of 2 to 5 (see Table 1).

Table 1. Comparison of DeCART and Gaussian Scheme Errors in the PWR Fuel Model.

δ	Extrapolated		0.001		0.005		0.01		0.02		0.03		0.04	
Na	Default	Gauss	Default	Gauss	Default	Gauss	Default	Gauss	Default	Gauss	Default	Gauss	Default	Gauss
10	-154	-28	-153	-27	-141	-6	-163	-72	-211	-54	-109	-85	106	-69
8	-198	-89	-199	-89	-216	-85	-186	-124	-202	-141	-69	-31	-163	-108
6	-320	-101	-320	-101	-327	-99	-302	-132	-308	-109	-279	-70	-126	-242
4	-511	-149	-511	-148	-506	-132	-559	-165	-475	-207	-369	-220	-292	-299

(Default: equally spaced angles; error unit: pcm)

Verification and Validation of the DeCART Neutronics Codes for LWR

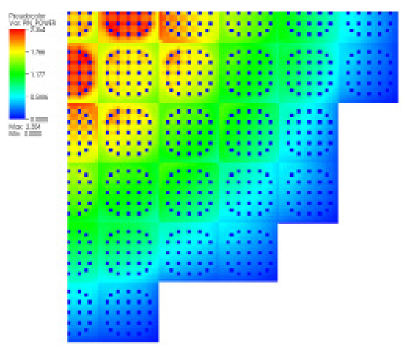
The team developed PWR and boiling water reactor (BWR) benchmark problems for DeCART verification tests. Realistic PWR core models were proposed and tested based on a conventional Westinghouse-type 17x17 fuel assembly in a 4-loop nuclear reactor. Each assembly consisted of 289 rods, comprising 264 fuel rods, 24 guide thimbles, and 1 instrument thimble. Initial fuel enrichment ranged from 4.0 to 4.5 wt%, although the fuel assemblies as modeled were depleted between 17 and 37 GWD/MTU to simulate a a a real core load scenario. Therefore, the fuel included about 30 fission products as well as higher actinides. The assembly also utilized integral fuel burnable absorbers (IFBAs). Small and large cores were constructed with 89 and 193 fuel assemblies, respectively, both including a baffle with a thickness of 2.52 cm at the core periphery.

For BWR benchmark problems, the project team used GE10 and Atrium10 assemblies with fuel rod arrays of 8x8 and 10x10, respectively. Small and large cores were constructed using 52 and 764 GE10 assemblies, respectively, with three different uranium fuel enrichments (2.2, 2.8, and 3.8 wt%) since GE10 was easier to use to construct a symmetric core than Atrium10.

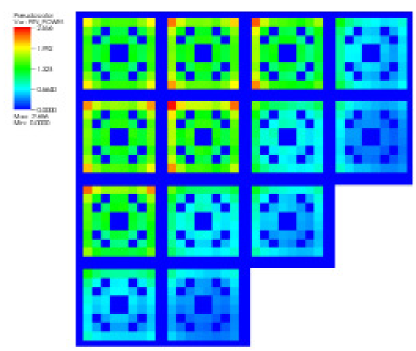
For the DeCART calculations, researchers used the ENDF/B-VI library: the 190-group library for fuel pin and fuel assembly calculations, and the 47-group library for the 2D cores. For the angular discretization, the rays were spaced 0.02 cm apart, and there were 8 azimuthal angles and 4 polar angles in a 90-degree domain. A uniform temperature condition of 300 K was used to simplify the comparison with MCNP5. Pin fission rates were tallied with more than 10 million particle histories in MCNP5. The asymmetries in the MCNP5 solution were symmetrically averaged out.

The researchers conducted step-by-step comparisons of eigenvalue and power between DeCART and MCNP5 for pin, assembly, and 2D cores. For PWR pins and assemblies, the DeCART eigenvalue agreed well with MCNP5 within 135 pcm $\Delta\rho$. Comparisons between the two codes are within 1.5% for the pin power distributions of the 104-IFBA fuel assembly. Eigenvalue differences between the two codes are -210 and -510 pcm $\Delta\rho$ for the small and large 2D cores, respectively. Fuel assembly powers are in very good agreement between the two codes for the small 2D core, with a maximum difference of 1.2% and a root mean square (RMS) of 0.6%; but differences are relatively large for the large 2D core, yielding a maximum of 4.2% and a RMS of 1.8%.

For BWR pins and assemblies, the maximum eigenvalue difference between the two codes was 108 pcm $\Delta\rho$. The pin power results for GE10 are within 1.0% and for Atrium10 are within 1.2%. Relatively large differences are observed in the pins neighboring the inside and outside water channels where more neutron thermalization results in power peaking. The eigenvalue difference of the small 2D core is within 118 pcm $\Delta\rho$. The large core was simulated using MCNP5 only, and thus no comparison was made. Figure 2 illustrates the pin power distributions of the small PWR and BWR benchmark problems.



PWR Core with 89 FAs



BWR Core with 52 FAs

Figure 2. Pin power distributions for small LWR benchmark cores.

procedure, using the reference resonance integrals using MCNP5. The procedure entails multiple fixed-source MCNP calculations for various temperatures and number densities in a pin-cell geometry, producing a table of resonance integrals as a function of temperature and background cross section for each isotope in the reference geometry. Preliminary test results of the subgroup cross-section libraries indicated that the estimated resonance cross sections for a few major actinides were accurate.

Planned Activities

U.S. research interests in reactor type for this project have changed from LWRs to sodium-cooled fast reactors (SFRs). Therefore, the project team will not pursue further LWR-based investigation. Instead, researchers will continue development of the subgroup cross-section library, utilizing DeCART as a verification tool. In addition, both sides will continue verification tests of the multi-physics simulation for the VHTR. In summary, the project team plans the following for FY 2013:

- Update the subgroup cross-section library.
- Further improve accuracy and computational efficiency of DeCART.
- Verify the updated cross-section libraries and the improved neutronics code using the numerical benchmark problems for VHTR cores.
- Initiate validation tests using the experimental benchmark problems.
- Initiate efforts to verify the multi-physics simulation.

Development of Diagnostics and Prognostics Methods for Sustainability of Nuclear Power Plant Safety Critical Functions

Project Number: 2011-004-K

PI (U.S.): Belle R. Upadhyaya and J. Wesley Hines, University of Tennessee

PI (ROK): Jung-Taek Kim, Korea Atomic Energy Research Institute

Collaborators: Chungnam National University, Kyung-Hee University, Pacific Northwest National Laboratory

Program Area: Reactor Concepts RD&D

Project Start Date: November 2011

Project End Date: November 2014

Research Objectives

The objective of this I-NERI project is to develop and demonstrate advanced plant monitoring, diagnostics, and prognostics methods during beyond design-basis accidents. Such events, marked by loss of residual heat removal and other safety-critical plant functions, coupled with station blackout and degradation of critical monitoring instrumentation, are difficult to mitigate without reliable information on critical parameters. This project is developing self-powered sensors that will provide plant information during extreme conditions, along with sensor networks incorporating these devices into plant communications, data transmission, and remote actuation systems. This effort includes demonstrating a remote sensing system consisting of a small number of self-powered sensors and processors in the form of a sensor network. Using dynamic simulation of a typical nuclear plant's operations, the U.S. and Korean collaborators will generate data needed to validate the proposed technical approach. Figure 1 shows a typical configuration of the proposed system consisting of an intelligent monitoring, diagnostic and prognostic system, remote sensing, sensor network, and self-powered sensing modules.

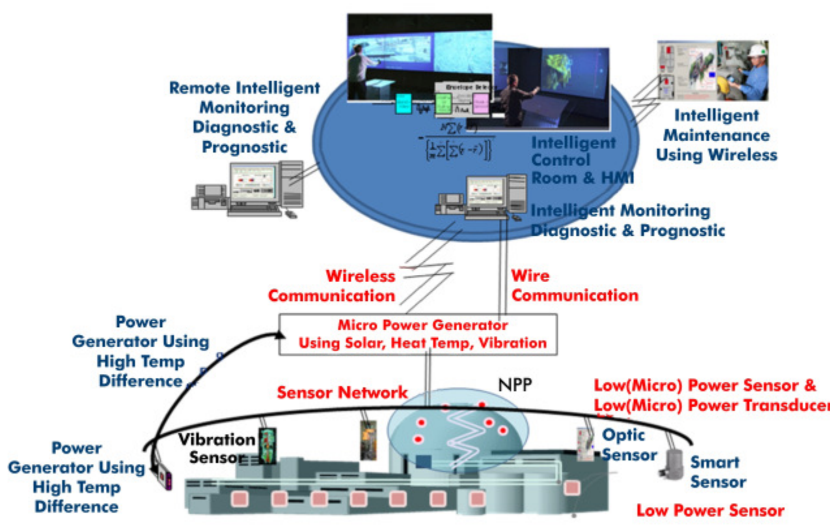


Figure 1. A typical configuration of the proposed system.

The project consists of the following key tasks:

1. Review operability of safety-critical functions and components in light water reactors (LWRs).
2. Develop algorithms for plant monitoring, diagnostics, and prognostics.
3. Develop low-energy, self-powered process sensors and networks, and determine optimal placement strategies.
4. Develop plant simulation models to generate normal operational and faulty condition data.
5. Demonstrate the integrated technology.

Research Progress

Task 1: Review of Operability of Safety Critical Functions and Components

Collaborators at Korea Atomic Energy Research Institute (KAERI) reviewed safety critical functions and components of typical LWRs in Korea, while the University of Tennessee (UT) conducted a parallel review for U.S. nuclear reactors. The objective was to collect input parameters related to critical safety functions. Both reviews considered the critical safety function and indication parameters found in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG)1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." These functions are keys to preventing core melt and minimizing radiation releases to the public. In addition to evaluating a standard LWR, KAERI conducted a similar review of the SMART modular reactor and outlined a plan for demonstrating techniques and algorithms in the ATLAS thermal-hydraulic test facility.

The U.S. team developed suggestions for instrumentation and supporting technology necessary to monitor beyond design-basis accidents at nuclear power plants, applying the latest guidance in RG 1.97 that establishes a strategy for selecting variables to monitor for accident conditions rather than simply listing parameters as in the earlier revision.

- Self-powered sensors (including solar and vibration-powered) to keep critical instrumentation functional during a severe accident.
- Wireless technology for transmitting important data to a remote location, thus avoiding issues concerning cable damage and other hard-wired transmission issues.
- Remote sensing of process variables (stand-off sensors) and inferring variables numerically.
- Prognostics of stand-by and emergency equipment using data acquired during periodic testing.
- Analytical on-line monitoring techniques to infer variables that are not directly measurable during an accident.

Task 2: Development of Plant Monitoring, Diagnostics, and Prognostics Algorithms

Researchers at Kyung-Hee University are identifying requirements, methodologies and resources for the development of a safety index monitoring during severe accidents (SIMSA) system. Characteristics of an independent and dedicated instrumentation and control (I&C) system for severe accident management are generating plant safety signals during and after the accident including station blackouts, tracking accident progression by identifying variations in critical parameters, representing the safety state through a single metric, and predicting the future state.

In several ways, the challenge resembles those addressed in condition-based maintenance (CBM), e.g., monitoring, fault diagnosis and prognosis (see Figure 2). Therefore, the research team reviewed the CBM framework to identify favorable techniques and resources for a SIMSA system and proposed a conceptual approach. Lack of physical models for material failure in accident conditions drives heavy reliance on data- driven computations and statistical learning algorithms. Active R&D has been going on in several CBM areas, and tools have been developed to address specific problems of interest.

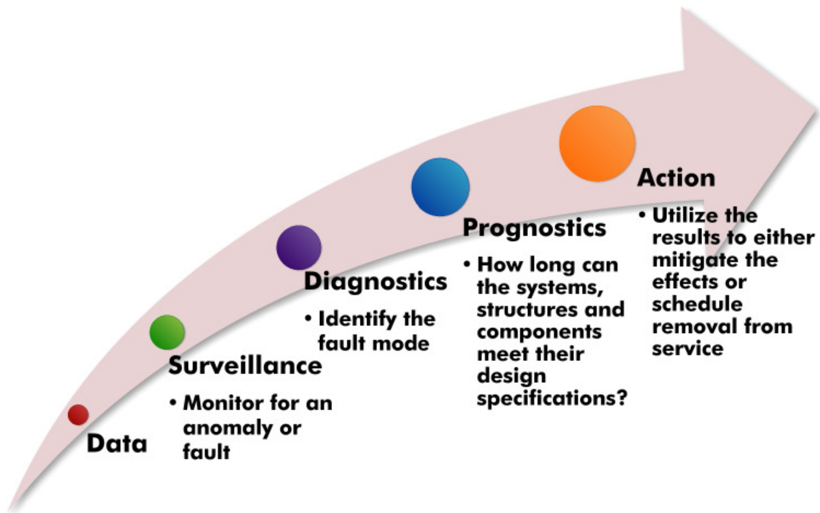


Figure 2. Stepwise application of data analysis for maintenance applications.

Researchers at UT have focused on computational techniques for equipment prognostics. Of the three types of models (failure data-based, stress-based, and effect-based), the UT team applies effect-based prognosis, which entails tracking equipment degradation over time and predicting when the total damage will exceed a predefined threshold. They are also researching combined prognostic-type models that utilize a combination of historical failure data, environmental data and effect-based data. To track equipment degradation in time, the team studied time distribution mapping and the transformation of a signal into a non-uniform bin space based on the empirical aspects of a transient signature. These experiments utilized other more traditional analysis methods to compare and validate the more novel transformation techniques for transient analysis.

Proof-of-concept testing conducted to demonstrate transient prognostic techniques included analysis of artificially induced degradation of neoprene pump impellers and motor-aging experiments. Twelve notched impellers were cycled until failure to self-prime, with such parameters as differential pressure, vibration, and current recorded for analysis as prognostic parameters. Researchers inferred a current life consumption (CLC) model (Figure 3), combined with a known lifetime run to produce a remaining useful life (RUL) estimate model. For the motor-aging tests, they subjected 5 HP motors to cyclic thermal stress and recorded current, voltage, vibration, and acoustic signatures. All signals showed progressive trends as the motor-aged. The team found time distribution mapping and signal transformation to a non-uniform bin space to be a clean and elegant method of generically quantifying the degradation of a transient signature.

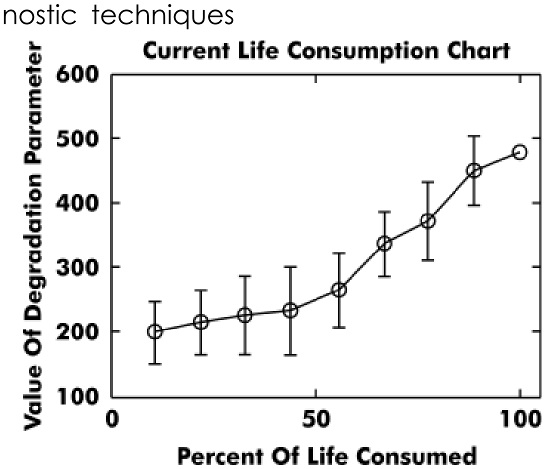


Figure 3. Impeller CLC chart.

UT researchers have also developed process equipment monitoring and prognostic tools for diagnostic application: a MATLAB process and equipment monitoring (PEM) toolbox (see Figure 4) and a process and equipment prognostics (PEP) toolbox. The PEM toolbox utilizes condition monitoring data and processes the data with wavelet transform, neural networks, signal processing, and auto-associative kernel regression to generate diagnostic signatures.

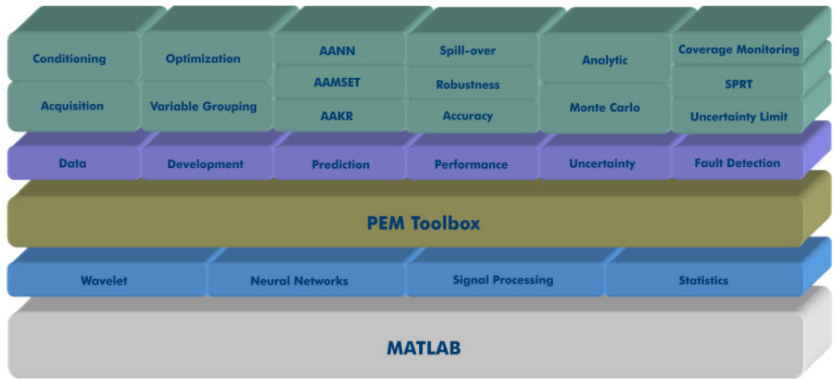


Figure 4. MATLAB-based PEM toolbox.

Task 3: Development of Low-Energy Self-Powered Process Sensors, Sensor Networks, Remote Monitoring, and Optimal Sensor Placement Strategy

KAERI is investigating low-energy sensors (solar, photo-voltaic, thermoelectric, and piezoelectric or electromagnetic vibration devices) for harvesting ambient light, heat, vibration, and electromagnetic energy from sensor surroundings and converting it into usable electrical energy to power portable electrical devices without batteries.

UT researchers are investigating *in situ* monitoring systems for critical equipment in small modular reactors (SMRs). They upgraded an existing experimental flow control loop with instrumentation designed to demonstrate the relationship between remotely measurable electrical signatures and process variables. Preliminary results show a strong correlation between the fluid pressure changes and the single-phase motor current (see Figure 5).

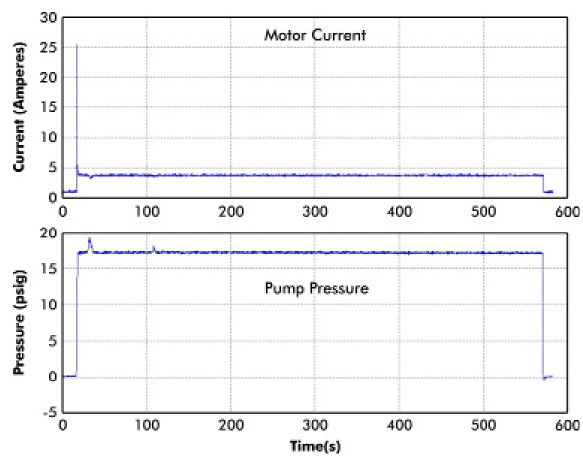


Figure 5. Motor current versus pump pressure in an experimental flow control loop.

Researchers at Chungnam National University (CNY) are developing and demonstrating wireless technologies for condition monitoring during a station blackout. Design issues identified include line of sight, frequency, antenna type (e.g., directional vs. omnidirectional), interference, modulation/demodulation, topologies, and network size. The team narrowed findings to the recommended network shown in Figure 6, consisting of three parts: the internal network, external network, and remote sensing room. Digital signals acquired from wired sensors inside the containment are combined in an internal network unit (INU) using a multiplexer to increase the efficiency. INUs placed on the outer wall of the containment (to avoid the harsh radio environment and to minimize damage during accidents) wirelessly transmit diagnostic informationto the internal network manager (INM) located in a remote sensing room, which in turn sends data to remote monitoring sites through an external network. A design framework has been developed to optimize sensor allocation based on detection and isolation of faults.

Planned Activities

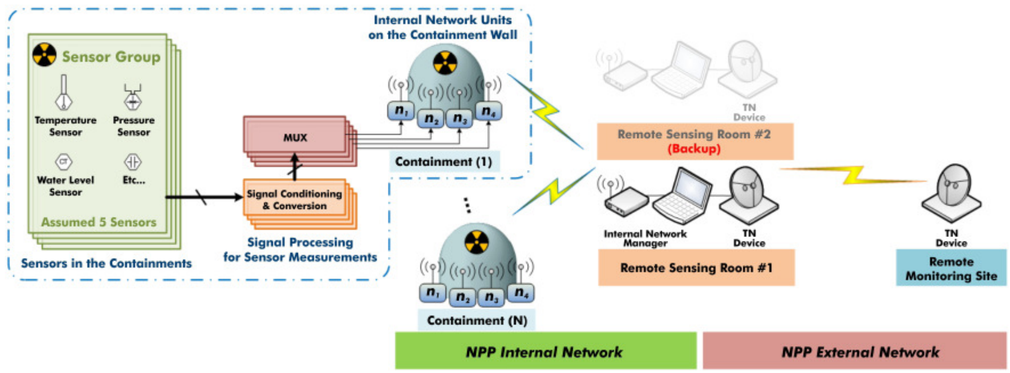


Figure 6. Recommended nuclear power plant network.

Task 1 is complete. During FY 2013, the research team will complete Task 2, continuing efforts to develop and evaluate internal and external plant monitoring, diagnostics, and prognostics networks. These efforts include identifying the requirements, methodologies and resources for SIMSA development, as well as development of diagnostic and prognostic models and use of these models to enhance prognostic tools such as the PEP. Task 3 efforts will continue, including 1) developing a test platform to evaluate feasibility of the recommended nuclear power plant network, 2) developing simulation models interfacing system dynamics with pump-motor models, and 3) identifying relationships between motor current, vibration, and process variables in the flow control loop. A parallel effort includes the development of models of SMR and pump-motor dynamics.

Fully Ceramic Microencapsulated Replacement Fuel for Light Water Reactor Sustainability

Project Number: 2011-005-K

PI (U.S.): Lance Snead, Oak Ridge National Laboratory

PI (ROK): Won-Jae Lee, Korea Atomic Energy Research Institute

Collaborators: Ultra Safe Nuclear Corporation, Inc.

Program Area: FCR&D

Project Start Date: October 2011

Project End Date: September 2014

Research Objectives

This project is assessing the feasibility of replacing conventional uranium oxide (UO_2) fuel assemblies in the existing fleet of light water reactors (LWRs) with accident-tolerant fully ceramic microencapsulated (FCM) fuel assemblies. FCM fuel is a TRISO-based dispersion fuel consisting of ceramic-coated microspheres of low-enriched uranium embedded in dense, uniform silicon carbide (SiC) matrix fuel pellets with similar geometry to conventional UO_2 pellets.

In order to replace conventional UO_2 fuel, FCM fuels must achieve comparable energy generation and heat transfer levels, as well as acceptable neutronic, thermal-hydraulic, and fuel performance characteristics. Acceptable designs must meet the limiting design basis accident (DBA) envelope for the plant and satisfy beyond design-basis accident (BDBA) criteria. The team will also model and qualify FCM fuel properties under irradiated conditions to ensure acceptable accident tolerance. The project consists of four tasks:

1. Core neutronics exploration
2. Core thermal-hydraulics assessment
3. Safety assessment
4. Fuel qualification

TThe project has selected two reference pressurized water reactor (PWR) cores for demonstration: the U.S. team will focus on a Westinghouse 1000-MWe core with 17×17 fuel assemblies, and the ROK team will examine their Optimized Power Reactor (OPR)-1000 core with 16×16 fuel assemblies.

Research Progress

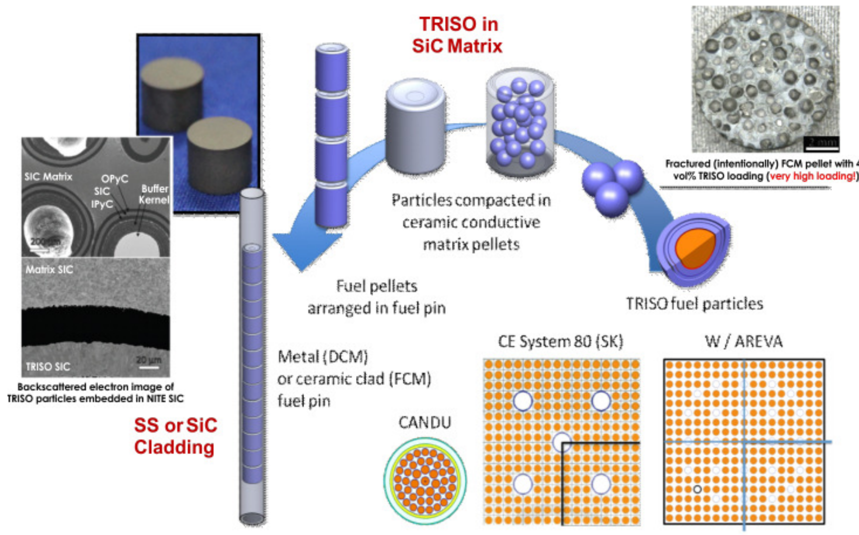


Figure 1. FCM replacement fuel design.

In the first year of this project, researchers developed and executed a screening process to evaluate and select potential FCM fuel assembly designs that can replace conventional UO_2 fuel assemblies and are compatible with existing OPR-1000 and Westinghouse cores. The team developed screening criteria for candidate fuel assembly designs set up to produce similar core reactivity, cycle length and safety parameters, compatible pressure drop characteristics and departure from nuclear boiling ratios (DNBRs), and mechanical and manufacturing compatibility with the reference cores and fuel. They then conducted quantitative analyses of the neutronic, thermal-hydraulic (T/H), and mechanical compatibility of each candidate design under realistic LWR environments. The team extended the evaluation of each fuel's performance over its expected lifetime to assess the safety of the FCM-fueled core.

Analysis methods and models applicable to the FCM fuel and core were established and applied to the preliminary compatibility analysis of neutronic, T/H and fuel performance, and to the safety evaluation. In parallel, FCM fuel samples were fabricated and irradiated up to 8 dpa (displacements per atom) for future examination and quantification of enhanced material properties and integrity.

The following sections summarize results in each task area.

Task 1: Neutronic Exploration

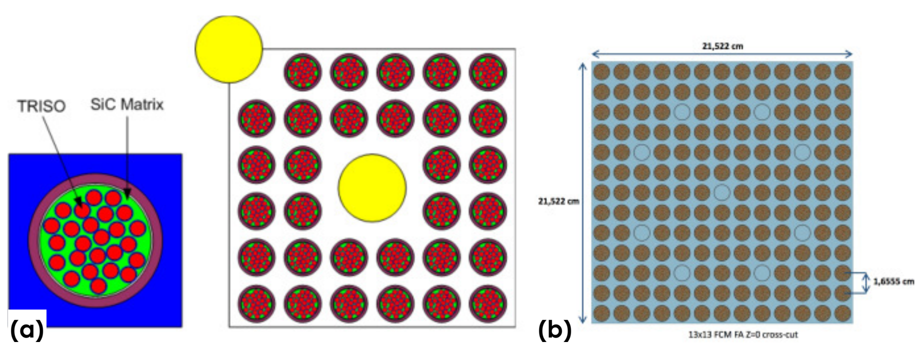


Figure 2. Candidate replacement FCM fuel assemblies for (a) OPR-1000 (12×12 quarter assembly shown) and (b) Westinghouse (13×13 full assembly shown).

Using the DeCART2D/MASTER and SERPENT neutronics codes, the project team performed scoping analysis using FCM fuels to identify possible replacement fuel assemblies. For the OPR1000 core, the team identified four feasible FCM fuel designs: uranium mononitride (UN) FCM fuel in 12×12 and 16×16 arrays, each with stainless steel or SiC as cladding materials. For the Westinghouse core, researchers selected UN FCM fuel in 13×13 arrays with zirconium alloy cladding; however, they may also consider alternate cladding materials such as stainless steels or new advanced iron–chromium–aluminum (FeCrAl) steel alloys. Figure 2 depicts two of the selected fuel assembly designs.

The project team used fuel assembly depletion calculations to identify optimized fuel assembly design parameters and analyzed neutron multiplication factors, reactivity coefficients, neutron spectrum effects of the FCM fuel, burnable poison effects, and fast neutron fluence (see Figure 3 for sample findings). Results to date indicate that the UN FCM fuel concept appears to be a feasible replacement fuel for existing LWRs from a neutronics perspective; however, some design choices (e.g., high TRISO packing fractions, high enrichments, and new fuel and cladding materials) may introduce technical and non-technical (e.g., regulatory or economic) challenges.

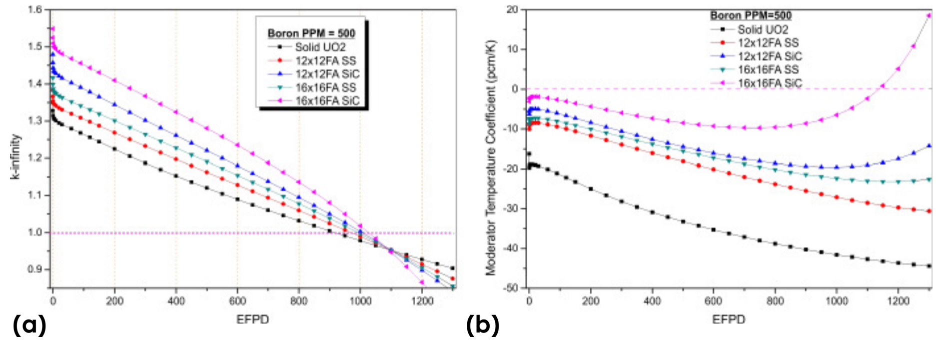


Figure 3. Analysis results for the OPR-1000 replacement fuels: (a) the neutron multiplication factor and (b) the moderator temperature coefficient.

Task 2: Core Thermal Hydraulic Assessment

Preliminary scoping analyses of the pressure drop and DNBR performance of OPR-1000 FCM fuel assemblies were carried out using the sub-channel analysis code MATRA for various combinations of assembly pitch-to-diameter ratio and spacer grid designs. Results identified feasible fuel assembly designs that are thermal-hydraulically compatible with the reference core. Sample findings are provided in Figure 4.

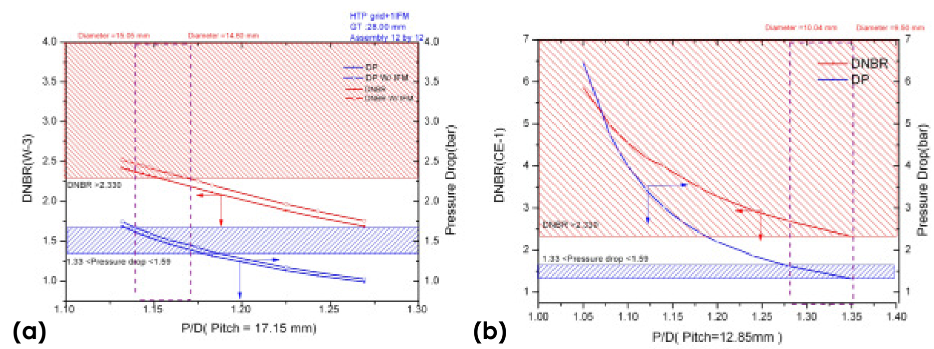


Figure 4. OPR-1000 fuel assembly pitch-to-diameter ratio impact on (a) DNBR and (b) pressure drop.

Task 3: Safety Assessment

The project team conducted preliminary safety assessments of an FCM-fueled OPR-1000 using MARS/MASTER, a three-dimensional coupled neutron kinetics–system T/H code. The FCM core demonstrated sufficient margin for preliminary safety criteria through analysis of the following DBA scenarios: a loss-of-flow accident (LOFA), loss-of-coolant accident (LOCA), and rod ejection accident (REA). Initial BDBA results for a LOCA without safety injection and station blackout indicate that FCM fuel survives longer than conventional fuel, which could provide more operator response time for mitigation actions (see Figure 5).

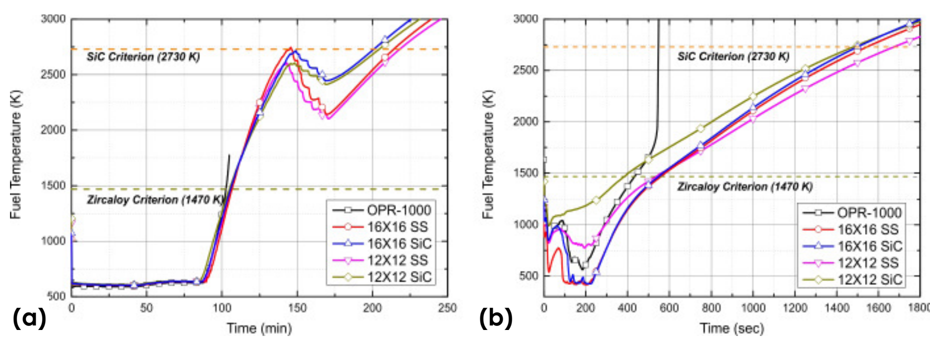


Figure 5. (a) Peak fuel temperature during station blackout and (b) peak clad temperature during a large break LOCA without safety injection.

Task 4: Fuel Qualification

The project team made significant progress in over-coating fuel to address specific FCM fuel fabrication issues, optimizing the FCM matrix for density and thermal conductivity, and demonstrating a route to fabricating a highly dense UN kernel. Researchers have initiated the first series of capsule irradiations in the High Flux Isotope Reactor, targeting the FCM matrix material's stability, and begun investigating fuel-clad interaction.

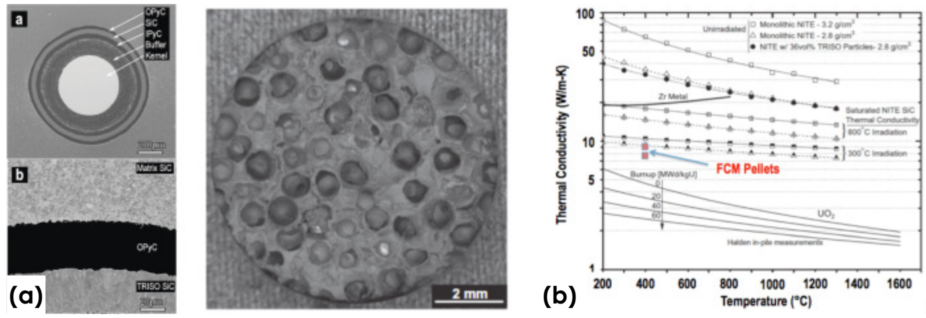


Figure 6. (a) FCM fuel TRISO-pellet sample (~45% packing) and (b) irradiated FCM pellet conductivity.

The team also performed preliminary FCM fuel performance assessments using the COPA/ABAQUS analysis system. For the selected FCM TRISO fuel, researchers assessed the coated particle performance over the fuel's expected lifetime using currently available preliminary material properties. Fission gas pressures (including noble gases and vapor species) at various operating conditions were shown to be maintained well below 4 MPa; however, the statistical SiC failure rate increases near 8 dpa. SiC failure is very sensitive to the properties of the irradiated inner pyrolytic carbon (IPyC) layer, which are not well known at fluences over 4 dpa. This area requires future work in refining the analytical models being used and improving the material property correlations.

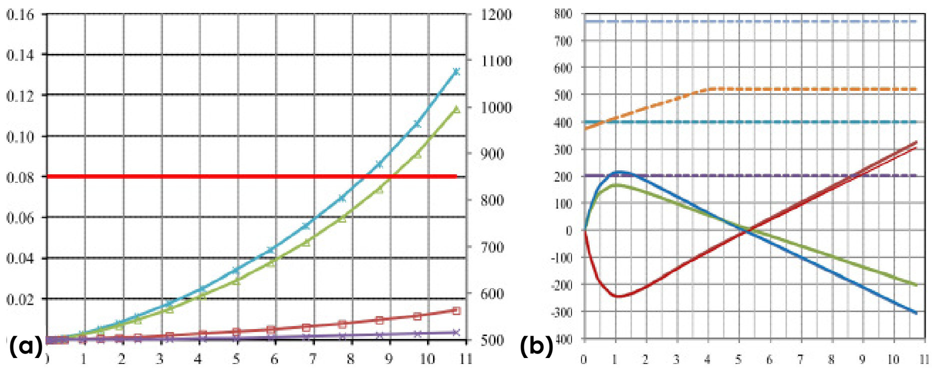


Figure 7. (a) Fission gas pressure in TRISO at 850°C and (b) stress at TRISO layers at 850°C.

Planned Activities

In the second project year, the research team will perform core neutronics, thermal-hydraulics, safety, and fuel performance analyses for a partially FCM-fueled transition core that also contains standard UO_2 fuel. FCM fuel manufacture and irradiation will continue as planned. Researchers will conduct the following activities specific to each task:

Task 1 (Neutronics Exploration)

- Search transition core loading patterns and optimize them using cycle-by-cycle core follow analysis, considering comparable cycle length and core physics parameters such as reactivity coefficients.
- Based on results of the core analysis, revisit FCM fuel designs and fast fluence.

Task 2 (Core T/H Assessment)

- Using the core physics data obtained from Task 1, analyze and evaluate core and sub-channel T/H compatibility for the transition cores.
- Improve T/H models and the graphical user interface utility program of the MATRA code.

Task 3 (Safety Assessment)

- Analyze and assess accident tolerance of the FCM-fueled transition cores for the limiting DBAs and BDBAs.
- Improve radiation heat transfer models for realistic simulation of BDBAs.

Task 4 (Fuel Qualification)

- Continue FCM fuel process development and measure relevant fuel material properties.
- Scale UN kernel fabrication to batch quantities, leading to TRISO coating and compaction.
- Irradiate FCM fuel and conduct post-irradiation examination to assess performance and identify any challenges or previously unknown issues.
- Improve and validate fuel performance analysis models using irradiation test results.
- Analyze and evaluate fuel performance in the transition cores, focusing on the thermomechanical and fission product transport behavior in the TRISO particle and SiC matrix.

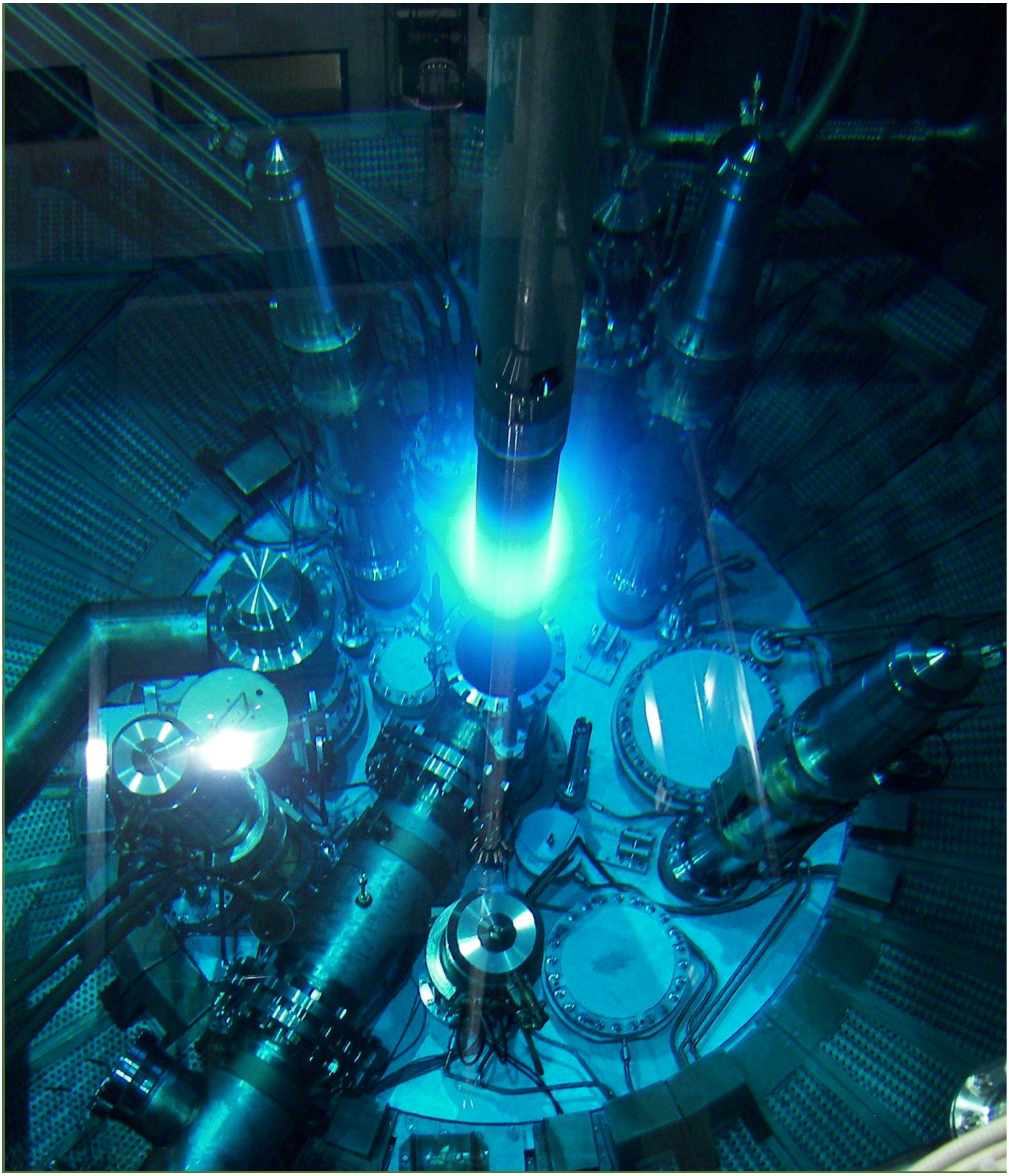


Photo courtesy of Public Domain

Appendix I. Acronyms

Misc

~	Approximately
Δ ρ	Delta Rho (denotes change in reactivity)
μm	Micrometer(s)
μg	Microgram(s)
2D	Two-Dimensional
3D	Three-Dimensional

A

AECL	Atomic Energy of Canada Limited
Ag	Silver
Al	Aluminum
ANL	Argonne National Laboratory
ANRE	Agency of Natural Resources and Energy
APS	Advanced Photon Source
APT	Atom Probe Tomography
Ar	Argon
ARC	Advanced Reactor Concepts
ATLAS	Advanced Thermal-Hydraulic Test Loop for Accident Simulation

B

bcc	Body-Centered Cubic
BDBA	Beyond Design-Basis Accident
BFS	Big Physical Stand (facility)
BWR	Boiling Water Reactor

C

C	Carbon
C	Celsius
Ca	Calcium
CASL	Consortium for Advanced Simulation of Light Water Reactors
CBM	Condition-Based Monitoring
Ce	Cerium
CEA	Commissariat à l'énergie atomique et aux énergies alternatives
CEN	Comité Européen de Normalisation
Cf	Californium
CFD	Computational Fluid Dynamics
CG	Coarse-Grained
CGR	Crack Growth Rate
Cl	Chlorine
CLC	Current Life Consumption
cm	Centimeter(s)
CNY	Chungnam National University
Cr	Chromium
CRP	Coordinated Research Project
CSM	Colorado School of Mines
CTR	Crystal Truncation Rod
Cu	Copper
CW	Cold Worked
CY	Calendar Year

D

D	Deuterium
DBA	Design-Basis Accident
DNBR	Departure from Nuclear Boiling Ratio
DNS	Direct Numerical Simulation
DOE	Department of Energy
dpa	Displacements Per Atom
DSA	Dynamic Strain Aging
DSC	Differential Scanning Calorimetry

E

EBSD	Electron Backscatter Diffraction
ECAE	Equal Channel Angular Extrusion
ECAP	Equal Channel Angular Processing
EDM	Electrical Discharge Machining
EDS	Energy Dispersive X-Ray Spectroscopy
ENDF	Evaluated Nuclear Data File
EPMA	Electron Probe Microanalyzer (EPMA)
EPRI	Electric Power Research Institute
EU	European Union
Eu	Europium

F

F	Flourine
F/M	Ferretic/Martensitic
FA	Fuel Assembly
fcc	Face-Centered Cubic
FCM	Fully Ceramic Microencapsulated (fuel)
FCR&D	Fuel Cycle Research and Development
Fe	Iron
FFTF	Fast Flux Test Facility
FIB	Focused Ion Beam
FIGARO	Fissile Interrogation Using Gamma Rays from Oxygen
FWHM	Full Width at Half Maximum
FY	Fiscal Year

G

Gd	Gadolinium
GELINA	Geel Linear Accelerator
Gen IV	Generation IV
GETMAT	Generation IV and Transmutation Materials (Euratom program)
GIF	Generation IV International Forum
GPa	Gigapascal(s)
GT	Guide Thimble
GWD/MTU	Gigawatt – Days per Metric Ton of Uranium

H

h	Hour(s)
H	Hydrogen
Hf	Hafnium
HIP	Hot Isostatic Press
HLM	Heavy Liquid Metal
HTGR	High-Temperature Gas-Cooled Reactor
HT-UPS	High-Temperature Ultrafine Precipitate-Strengthened (steel alloy)

I

I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
IFBA	Integral Fuel Burnable Absorber
IFNEC	International Framework for Nuclear Energy Cooperation
IMF	Inert Matrix Fuel
INC	International Nuclear Cooperation (framework)
I-NERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
INM	Internal Network Manager
INU	Internal Network Unit
IPPE	Institute of Physics and Power Engineering

IRIS	International Reactor Innovative and Secure
IRMM	Institute for Reference Materials and Measurements
ISO	International Organization for Standardization
IT	Instrument Thimble
ITU	Institute for Transuranium Elements

J

J	Joule
JEFF	Joint Evaluated Fission and Fusion
JRC	Joint Research Centre

K

K	Kelvin
K	Potassium
KAERI	Korea Atomic Energy Research Institute
k_{eff}	k-effective (neutron multiplication factor)
keV	Kiloelectron Volt(s)
keV_{ee}	Electron Equivalent Recoil Energy (measured by scintillation light)
kPa	Kilopascal(s)
kW	Kilowatt

L

LANL	Los Alamos National Laboratory
LANSCE	Los Alamos Neutron Science Center
LAS	Low-Alloy Steel
LCF	Low-Cycle Fatigue
LES	Large Eddy Simulation
Li	Lithium
LOCA	Loss-of-Coolant Accident
LOFA	Loss-of-Flow Accident
LRUS	Laser Resonant Ultrasonic Spectroscopy
LWR	Light Water Reactor

M

M	Molar
MA	Mechanically Alloyed
MA	Minor Actinide
MatDB	MIPS Arabidopsis Thaliana Database
MATLAB	Matrix Laboratory
MCNP	Monte Carlo N-Particle
MEST	Ministry of Education, Science and Technology
MeV	Mega-Electron Volt
MEXT	Ministry of Education, Culture, Sports, Science, and Technology
mg	Milligram(s)
mm	Millimeter(s)
Mn	Manganese
Mo	Molybdenum
MONNET	Mono Energetic Neutron Tower
MOST	Ministry of Science and Technology (superseded by MEST)
MOX	Mixed Oxide
MPa	Megapascal(s)
MPa√m	Megapascal Square Root Meter (measure of fracture toughness)
MPM	Mechanical Properties Microscope
MST	Ministério da Ciência e Tecnologia
MV	Megavolt(s)
MWe	MegaWatt electric

N

N	Newton(s)
Na	Sodium
NE	Office of Nuclear Energy
NE-6	Office of International Nuclear Energy Policy and Cooperation
NEA	Nuclear Energy Agency (of OECD)
NEAMS	Nuclear Energy Advanced Modeling and Simulation (program)
NFA	Nanostructured Ferritic Alloy
NGNP	Next Generation Nuclear Plant

Ni	Nickel
nm	Nanometer(s)
NRC	Nuclear Regulatory Commission
NRCan	Department of Natural Resources Canada

O

O	Oxygen
ODS	Oxide Dispersion-Strengthened
OECD	Organisation for Economic Co-operation and Development
OPR	Optimized Power Reactor
ORNL	Oak Ridge National Laboratory

P

Pb	Lead
pcm	Per Cent Mille
PEM	Process and Equipment Monitoring
PEP	Process and Equipment Prognostics
PFNS	Prompt Fission Neutron Spectrum
PI	Principal Investigator
ppm	Parts Per Million
ps	Picosecond(s)
Pu	Plutonium
PWR	Pressurized Water Reactor

R

R&D	Research and Development
RCH	Rotating Cylinder Hull
RD&D	Research, Development and Demonstration
REA	Rod Ejection Accident
RG	Regulatory Guide
RHEED	Reflection High-Energy Electron Diffraction
RMS	Root Mean Square
ROK	Republic of Korea
RUL	Remaining Useful Life

S

s	Second(s)
S	Sulfur
SA	Solution Annealed
SCC	Stress Corrosion Cracking
SEM	Scanning Electron Microscope/ Microscopy
SF	Spontaneous Fission
SFR	Sodium-Cooled Fast Reactor
SG	Steam Generator
SHARP	Simulation-Based High Accuracy Advanced Reactor Prototyping
SiC	Silicon Carbide
SIMS	Secondary Ion Mass Spectrometry
SIMSA	Safety Index Monitoring during Severe Accidents
SMART	System-Integrated Modular Advanced Reactor
SMR	Small Modular Reactor
SND	Standard Neutron Detector
SRO	Short-Range Ordering
SSMT	Small-Scale Materials Testing
STIP	Swiss Spallation Neutron Source Target Irradiation Program

T

T/H	Thermal-Hydraulic
TCM	Thermal Conductivity Microscope
TEM	Transmission Electron Microscope/ Microscopy
Th	Thorium
Ti	Titanium
TKE	Total Kinetic Energy
TMT	Thermomechanical Treatment
TOF	Time of Flight
TRISO	Tristructural-Isotropic
TRU	Transuranic
TT	Thermally Treated

U

U	Uranium
UC	University of California
UFFC	Ultrasonics, Ferroelectrics and Frequency Control
UFG	Ultrafine-Grained
UNIST	Ulsan National Institute of Science and Technology
URL	Underground Research Laboratory
UT	University of Tennessee

V

V	Volt(s)
VERDI	Velocity for Direct Particle Identification
VHTR	Very High-Temperature (Gas-Cooled) Reactor

W

W	Tungsten
W	Watt
WNR	Weapons Neutron Research (facility)
WQ	Water Quenched
wt%	Weight Percent

X

XML	Extensible Markup Language
XRD	X-Ray Diffraction

Y

Y	Yttrium

Z

ZPPR	Zero Power Plutonium Reactor
Zr	Zirconium

Appendix II: Index of Active I-NERI Projects

2009

2009-001-K	ZPPR-15 and BFS Critical Experiments Analysis for Generation of Physics Validation Database of Metallic-Fueled Fast Reactor Systems
2009-002-K	Enhanced Radiation Resistance Through Interface Modification of Nanostructured Steels for Gen IV In-Core Applications

2010

2010-001-E	Measurements of Fission Fragment Mass Distributions and Prompt Neutron Emission as a Function of Incident Neutron Energy for Major and Minor Actinides
2010-002-E	Spherical Particle Technology Research for Advanced Nuclear Fuel/Target Applications
2010-003-E	Irradiation and Testing of Advanced Oxide Dispersion-Strengthened and Ferritic–Martensitic Steels
2010-004-E	Development of a Standard Neutron Detector for the Energy Range up to 20 MeV and Its Application
2010-005-E	Interoperability of Material Databases
2010-006-E	State-of-the-Art Post-Irradiation Examination of Advanced Nuclear Fuels
2010-001-K	Investigation of Electrochemical Recovery of Zirconium from Spent Nuclear Fuels
2010-002-K	Science-Based Approach to Nickel Alloy Aging and Its Effect on Cracking in Pressurized Water Reactors
2010-003-K	Low-Loss Advanced Metallic Fuel Casting Evaluation
2010-004-K	Development and Characterization of Nanoparticle-Strengthened Dual-Phase Alloys for High-Temperature Nuclear Reactor Applications

2011

2011-001-E	Development of a 2E-2V Instrument for Fission Fragment Research
2011-001-K	Atomic Ordering in Alloy 690 and Its Effect on Long-Term Structural Stability and Stress Corrosion Cracking Susceptibility
2011-002-K	Development of Microcharacterization Techniques for Nuclear Materials
2011-003-K	Verification and Validation of High-Fidelity Multi-Physics Simulation Codes for Advanced Nuclear Reactors
2011-004-K	Development of Diagnostics and Prognostics Methods for Sustainability of Nuclear Power Plant Safety Critical Functions
2011-005-K	Fully Ceramic Microencapsulated Replacement Fuel for Light Water Reactor Sustainability

2012

2012-001-E	High-Fidelity Thermal Hydraulic Fuel Assembly Simulations for Nuclear Reactors

Kernkraftwerk Isar Nuclear Power Plant at night
Photo courtesy of Creative Commons