DOE/NE-127

NHI

Generation IV

AFCI

AEN NEA

INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

2004 Annual Report

Foreword

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

The international collaborative research of I-NERI allows DOE to better leverage its economic resources, expand its knowledge base on nuclear science and engineering, and establish valuable intellectual relationships with researchers from other countries. Current collaborating countries and international organizations include: Canada, the European Union, France, Japan, the Organization for Economic Cooperation and Development/Nuclear Energy Agency, and the Republic of Korea.

FY 2004 was a year of significant accomplishments for I-NERI. New projects were initiated in direct support of the Department's Generation IV Nuclear Energy Systems (Generation IV), the Advanced Fuel Cycle (AFCI), and the Nuclear Hydrogen (NHI) initiatives.

Some noteworthy I-NERI achievements during FY 2004 include: seven new projects initiated with Canada, eleven with France, and six with the Republic of Korea; a new Implementing Arrangement signed with Japan; and leveraged U.S. contributions of $59.8 million with $65.2 million of international contributions including $7.3 million from Canada, $2.5 million from the European Union, $38.4 million from France, and $17.0 million from the Republic of Korea.

This annual report includes FY 2004 programmatic accomplishments and summarizes research progress based on information submitted by the principal investigators for I-NERI projects initiated in FY 2001 through FY 2004.

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

Contents

Acronyms

ACR	Advanced CANDU Reactor
ADS	accelerator-driven systems
AECL	Atomic Energy of Canada Limited
AFCI	Advanced Fuel Cycle Initiative
ALWR	Advanced Light Water Reactor
ANL	Argonne National Laboratory
APT	atom probe tomography
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BOP	balance of plant
CANDU	Canada Deuterium Uranium
CCI	core-concrete interaction
CEA	Commisariat a l'Énergie Atomique (French Atomic Energy Commission)
CFD	computational fluid dynamic
CHF	critical heat flux
CNU	Chungnam National University
COMET	coolability of melt by water injection experiment
CRL	Chalk River Laboratories
CTFP	Computational Thermal Fluid Physics
CU	Chosun University
DeCART	Deterministic Core Analysis based on Ray Tracing
DNBR	Departure from Nucleate Boiling Ratio
DHRS	Decay Heat Removal System
DNS	direct numerical simulations
DOE	U.S. Department of Energy
DOE-NE	Office of Nuclear Energy, Science and Technology
DSM	differential second moment closure
DUPIC	direct use of spent PWR fuel in CANDU reactor
EAC2	European Accident Codes
ECP	electrochemical corrosion potential
ENDF-B	Evaluated Nuclear Data Files
ENIGMA	Experimental Neutron Investigation on Gasreactors at MAsurca
EPR	European Pressurized Water Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic and Simplified Boiling Water Reactor
ETDR	Experimental and Technology Demonstration Reactor
E.U.	European Union
EURATOM	European Atomic Energy Community
F-M	ferritic-martensitic
FAC	flow-accelerated corrosion
FY	fiscal year
GA	General Atomics
GCR	Gas-Cooled Reactor
Generation IV	Generation IV Nuclear Energy Systems
GFR	Gas-Cooled Fast Reactor
GIF	Generation IV International Forum
GTI	Gas Technology Institute
HEXNODYN	a European neutronics computer code

HTE	high-temperature electrolysis
HTGR	High-Temperature Gas-Cooled Reactor
ID	Idaho Operation Office
IE	(see JRC/IE)
IMF	inert matrix fuels
I-NERI	International Nuclear Energy Research Initiative
IPyC	inner pyrolytic carbon
INL	Idaho National Laboratory
ITU	Joint Research Center Institute for Transuranium Elements
IVR	in-vessel retention
JNT	Johnson noise thermometry
JRC/IE	Joint Research Center of the European Commission Institute for Energy
KAERI	Korea Atomic Energy Research Institute
KAIST	Korea Advanced Institute of Science and Technology
KHNP	Korea Hydro and Nuclear Power
KSNP	Korea Standard Nuclear Power Plants
LANL	Los Alamos National Laboratory
LDV	laser Doppler velocimetry
LEAP	local electrode atom probe
LES	large eddy simulations
LFR	Lead-Cooled Fast Reactors
LOCA	loss-of-coolant accident
LOFC	loss of forced convection
LPD	local power density
LWR	Light Water Reactor
MA	minor actinides
MACE	melt attack and coolability experiment
MASTER	Multipurpose Analyzer for Static and Transient Effects of Reactor
MCCI	Melt Coolability and Concrete Interaction
MCNP	Monte Carlo N-Particle Transport Code
MIR	Matched-Index-of-Refraction
MIT	Massachusetts Institute of Technology
MOST	Republic of Korea Ministry of Science and Technology
MOX	mixed oxide fuel
MSR	Molten Salt Reactor
Music	multiple signal classification
NEA	Nuclear Energy Agency
NERI	Nuclear Energy Research Initiative
NGNP	Next Generation Nuclear Plant
NHI	Nuclear Hydrogen Initiative
NRC	Nuclear Regulatory Commission
NTD	National Technical Directors
ODS	oxide dispersion strengthened steels
OECD	Organization for Economic Cooperation and Development
OREOX	Oxidation and Reduction of Oxide
ORNL	Oak Ridge National Laboratory
OSU	Ohio State University
PCAST	President's Committee of Advisors on Science and Technology
PID	proportional-integral-derivative method
PIV	particle image velocimetry
PNNL	Pacific Northwest National Laboratory
PR&PP	proliferation resistance and physical protection
PRA	Probabilistic Risk Assessment
PSU	Pennsylvania State University

PWR	Pressurized Water Reactor
R&D	research and development
RANS	Reynolds-averaged Navier-Stokes
RCCS	reactor cavity cool-down system
ROK	Republic of Korea
RPV	Reactor Pressure Vessel
SCC	stress corrosion cracking
S-CO_2	super-critical carbon dioxide
SFR	Sodium-Cooled Fast Reactor
SCW	super-critical water
SCWR	Super-Critical Water-Cooled Reactor
SiC	silicon carbide
SIM	System Integrator Manager
SNU	Seoul National University
SOFC	solid oxide fuel cell
SSFM	solid-state in-core flux monitor
SSWICS	Small Scale Water Ingression and Crust Strength
STAR	Secure Transportable Autonomous Reactor
STARCD	a general purpose computational fluid dynamics program
STAR-LM	Secure Transportable Autonomous Reactor -Liquid Metal
TEM	transmission electron microscopy
TEM-EELS	transmission electron microscope/energy-loss spectroscopy
TRISO	tristructural isotropic
UCSB	University of California, Santa Barbara
UM	University of Michigan
UOX	uranium oxide
U.S.	United States
UW	University of Wisconsin
VHTGR	Very High-Temperature Gas-Cooled Reactor
VHTR	Very High-Temperature Reactor

1.0 Introduction

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

This *I-NERI 2004 Annual Report* serves to inform interested parties on the program's organization, progress of the collaborative research projects, and future planning for the program. The report covers the four years of I-NERI activity since the program's inception in fiscal year (FY) 2001.

The motivation and series of events that led to the creation of the I-NERI program are discussed in Section 2. The participating countries in current I-NERI collaborative agreements are also presented.

Section 3 presents an overview of the I-NERI goals and objectives, a work scope summary for the collaborative projects, and a summary of research project awards through the end of FY 2004. It also includes FY 2005 accomplishments and planned FY 2005 activities.

A summary of programmatic accomplishments is presented in Section 4. Also included is a summary of new and existing projects, an overview of the I-NERI program funding, and the new collaborations anticipated in FY 2005.

The R&D work scope for current I-NERI collaborative projects with Canada, France, Japan, the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), and the Republic of Korea are presented in Sections 5 through 9, respectively.

For each participating country, an index of projects and a summary of FY 2004 technical accomplishments are presented in Appendices A through E.

2.0 Background

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

In response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI) in 1999. Information and annual reports on the NERI program are available at the NERI website: http://neri.ne.doe.gov.

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

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

To date, five I-NERI collaborative agreements have been fully implemented: the first between DOE and the Commisariat a l'Énergie Atomique (CEA) of France; the second between DOE and the Republic of Korea Ministry of Science and Technology (MOST); the third with the OECD/NEA; the fourth between DOE and the European Union (E.U.), and the fifth between DOE and the Department of Natural Resources Canada. Since the program's inception, sixteen projects with France, seventeen with the Republic of Korea, one with NEA, seven with Canada, and eight with the European Union have been initiated. New cooperative agreements were signed in FY 2003 with Brazil and FY 2004 with Japan. Actions to implement these new collaborations will be taken during FY 2005. Discussions are ongoing with the Republic of South Africa and the United Kingdom with the intent that at least two additional I-NERI collaborations will be established during FY 2005. A list of implemented I-NERI projects is provided in the Appendices. Abstracts for these projects are available on the DOE website: http://www.nuclear.gov.

3.0 I-NERI Program Description

3.1 Mission

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

3.2 Goals and Objectives

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

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

Since the I-NERI program's inception, the Office of Nuclear Energy, Science and Technology (DOE-NE) has coordinated wide-ranging discussions among governments, industry, and the worldwide research community regarding the development of Generation IV Nuclear Energy Systems (Generation IV) nuclear energy systems.

In FY 2004, DOE restructured I-NERI so that the program became a key collaboration mechanism for conducting research with international partners whose R&D work scopes are intimately linked to the Generation IV, Advanced Fuel Cycle Initiative (AFCI), and Nuclear Hydrogen Initiative (NHI) research areas.

3.3 International Agreements

In order to initiate any international collaborations, a government to government agreement must be in place. I-NERI agreements were established to allow international bilateral R&D collaborations in the area of nuclear technology. These agreements are the vehicle to conduct Generation IV, AFCI, and NHI R&D with member countries of the Generation IV International Forum (GIF). There are currently 11 GIF members -- Argentina, Brazil, Canada, European Union, France, Japan, Republic of Korea, Republic of South Africa, Switzerland, the United Kingdom, and the United States. The United States has established I-NERI bilateral agreements with six of these member countries (Brazil, Canada, European Union, France, Japan, and the Republic of Korea). The GIF partners are in the process of establishing multilateral agreements to conduct multilateral R&D among GIF countries. In the meantime, I-NERI, through the implementation of these bilateral agreements, enables R&D collaborations to begin developing next generation energy systems.

3.4 Areas of Cooperation

The areas of cooperation of the FY 2004 I-NERI projects were defined so that they supported the R&D needs of the Generation IV, AFCI, and NHI programs. Below is an overview of the individual work scopes for the three programs:

- **Generation IV Nuclear Energy Systems Initiative.** The Generation IV program is the development of next-generation nuclear energy systems that offer advantages in the areas of economics, safety and reliability, and sustainability, and can be commercially deployed by 2030. Using a technology roadmap that was created by member countries in the Generation IV International Forum (GIF), the Generation IV program was assigned six reactor system and fuel cycle concepts that were deemed most promising for achieving the aforementioned advantages. There are eight technology goals for the Generation IV program: (1) provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production; (2) minimize and manage nuclear waste, notably reducing the long-term stewardship burden in the future and thereby improving protection for the public health and the environment; (3) increase the assurance that systems are a very unattractive and least desirable route for diversion or theft of weapons-usable materials; (4) ensure that systems will excel in safety and reliability; (5) design systems that have a very low likelihood and degree of reactor core damage; (6) create reactor designs that eliminate the need for offsite emergency response; (7) ensure that systems have a clear life-cycle cost advantage over other energy sources; and (8) create systems that have a level of financial risk that is comparable to other energy projects. The six reactor systems selected to meet the Generation IV technology goals are: Gas-cooled Fast Reactor, Lead-cooled Fast Reactor, Molten Salt Reactor, Sodium-cooled Fast Reactor, Super-Critical WaterCooled Reactor, and the Very-High Temperature Reactor.

- **Advanced Fuel Cycle Initiative.** The 2001 National Energy Policy recommended that the United States "… develop reprocessing and fuel treatment technologies that are cleaner, more efficient, less waste-intensive, and more proliferation-resistant." These technologies are key components of the fuel cycles that are required for Generation IV nuclear energy systems since the advanced reactor designs of Generation IV systems will use fuel cycles that are significantly different from those used by existing U.S. reactors. The AFCI research on recycling, fuel treatment, and conditioning technologies has the potential to dramatically reduce the quantity and toxicity of spent nuclear fuel, thus decreasing the need for geological disposal. The AFCI mission is to develop proliferation-resistant spent nuclear fuel treatment and transmutation technologies in order to enable a transition from the current once-through nuclear fuel cycle to a future sustainable closed nuclear fuel cycle.

- **Nuclear Hydrogen Initiative.** The NHI program is part of the DOE's Hydrogen Fuel Initiative. The goal of the Hydrogen Fuel Initiative is to develop the technologies and infrastructure to economically produce, store, and distribute hydrogen for use in fuel cell vehicles and electricity generation. Hydrogen can be produced using a variety of technologies, each of which has its advantages and limitations. The primary advantage of nuclear energy production technologies is the ability to produce hydrogen in large quantities, at a relatively low cost, without the emission of any greenhouse gases. The goal of the NHI is to demonstrate the commercial-scale, economically-feasible production of hydrogen using nuclear energy by the year 2017. The NHI will conduct R&D on enabling technologies, demonstrate nuclear-based hydrogen producing technologies, study potential hydrogen production schemes, and develop deployment alternatives to meet future needs for increased hydrogen consumption.

The specific work scope of each I-NERI collaboration is established by agreement between DOE and the respective agency of the collaborating international country.

3.5 Program Organization

The Office of Nuclear Energy, Science and Technology (NE) manages the U.S. element of I-NERI, with advice from the Nuclear Energy Research Advisory Committee (NERAC). NE's Office of International Nuclear Affairs negotiates and establishes the I-NERI bilateral agreements.

I-NERI is part of the Generation IV program, and it is currently the only vehicle for international R&D collaboration in Generation IV technology. I-NERI enables collaboration with the GIF countries on a bilateral basis until multilateral agreements are established.

The I-NERI Program Manager manages the implementation of the I-NERI agreements and administers all international collaborations under these agreements. The Idaho Operations Office (ID) negotiates and monitors the cooperation agreements with U.S. entities.

The U.S. appoints a DOE Country Coordinator for each country that we collaborate with; the collaborating country establishes a similar function. The DOE Country Coordinator represents the United States in bilateral meetings and negotiates areas of collaboration, selects new projects, and evaluates existing projects. Figure 1 shows the DOE I-NERI organizational chart.

The I-NERI projects are periodically reviewed by technical experts. In the U.S., the National Technical Directors (NTDs), who are assisted by System Integrator Managers (SIMs), serve as the technical experts. There are seven NTDs that represent the Generation IV/AFCI technology areas of systems analysis, fuels, materials, energy conversion, chemical separations, transmutations, and system design and evaluation. The SIMS manage the technologies for each of the following concepts: VHTR, SCWR, GFR, LFR, SFR, and MSR. The NTDs and SIMs also assist the DOE Country Coordinator in identifying areas of research cooperation and specific work scopes.

3.6 Funding

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

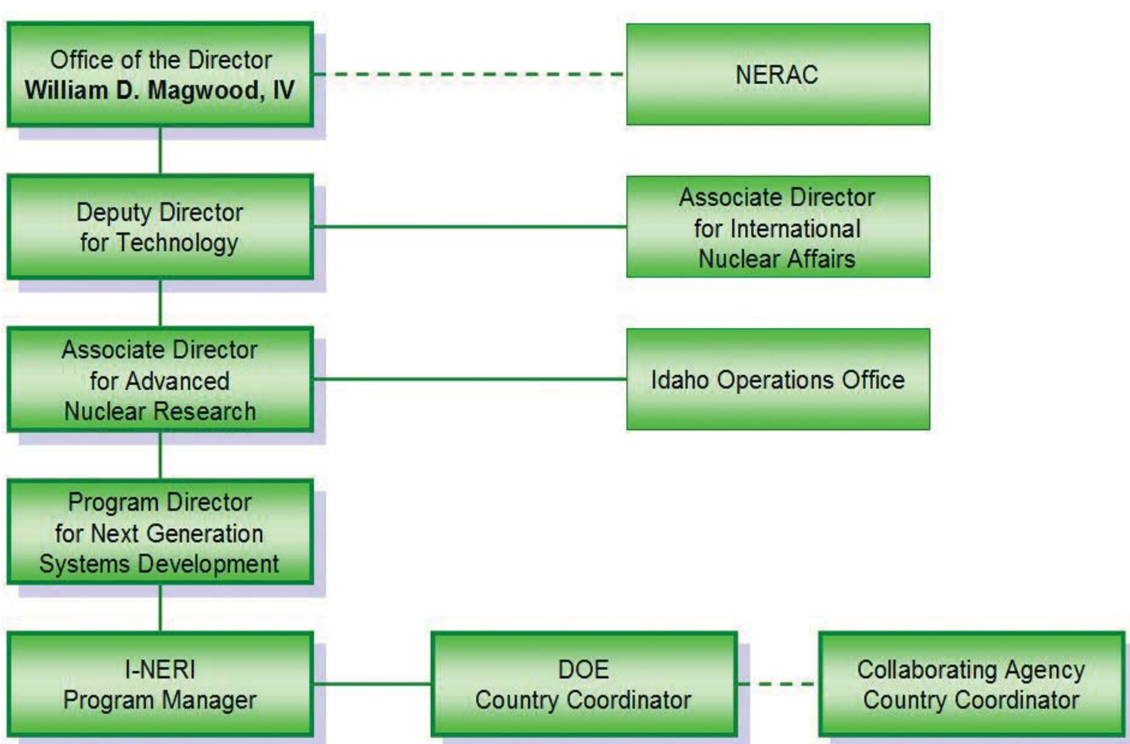


Figure 1. Office of Nuclear Energy, Science and Technology I-NERI Organizational Chart.

4.0 I-NERI Program Accomplishments

The I-NERI program effectively began in the second quarter of FY 2001 with an initial focus on developing international collaborations, program planning, and project procurements. Awards for the first set of I-NERI collaborative research projects with France were made at the end of FY 2001. The I-NERI program's progress for FY 2001 through 2004 is summarized in Section 4.1.

4.1 Programmatic Accomplishments

The primary programmatic accomplishments during FY 2001 through FY 2004 and the planned accomplishments for FY 2005 are briefly described below.

4.1.1 FY 2001 Programmatic Accomplishments

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

4.1.2 FY 2002 Programmatic Accomplishments

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

4.1.3 FY 2003 Programmatic Accomplishments

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

4.1.4 FY 2004 Programmatic Accomplishments

- Completed FY 2003 annual project performance reviews for the U.S./France and U.S./Republic of Korea collaborations and confirmed projects approved for ongoing support.
- Completed two U.S./France projects awarded in FY 2001.
- Added seven new projects to the U.S./Canada collaboration.
- Added eleven new projects to the U.S./France collaboration.
- Added six new projects to the U.S./Korea collaboration.
- Approved eight research project proposals under the agreement with the European Union.
- Signed new I-NERI cooperative agreement with Japan.

4.1.5 Planned FY 2005 Programmatic Accomplishments

- Initiate a research project under the agreement with Brazil.
- Complete FY 2004 annual project performance review for the U.S./Canada, European Union, France, and Republic of Korea collaborations.

- Initiate a research project under the agreement with Japan.
- Initiate new cooperative projects under existing agreements.

4.2 Current I-NERI Collaborations

In FY 2004, DOE established new projects under new and existing international agreements. FY 2004 was the transition year to more focused R&D areas of collaboration in support of Generation IV, AFCI, and NHI. New projects initiated in FY 2004 were funded by these three programs. Brief descriptions of the current I-NERI collaborations are provided in the sections that follow. Descriptions of the work scopes, listings of funded projects, and brief project status reports are provided in Sections 5 through 9 and Appendices A through E for the U.S./Canada, U.S./European Union, U.S./France, U.S./Republic of Korea, and U.S./OECD collaborations, respectively.

4.2.1 U.S./Canada

The collaborating agency in Canada is Atomic Energy of Canada Limited (AECL). The U.S./Canada collaboration includes R&D proposals in the Generation IV, AFCI, and NHI areas. The FY 2004 awarded projects are the first under this bilateral collaboration and the projects range from two to four years in duration.

4.2.2 U.S./European Union

The collaborating agency for the European Union (E.U.) is the European Commission. The U.S./E.U. collaboration includes R&D proposals in the Generation IV, AFCI, and NHI areas. The FY 2004 awarded projects are the first under this bilateral collaboration and the projects range from two to four years in duration.

4.2.3 U.S./France

The collaborating agency in France is CEA. The U.S./France collaboration focuses on developing Generation IV advanced nuclear system technologies that will enable the U.S. and France to move forward with cutting-edge, generic R&D that will benefit the range of anticipated future reactor and fuel cycle designs.

4.2.4 U.S./Japan

An agreement was signed with the Agency of Natural Resources and Energy (ANRE) of Japan on May 26, 2004. The areas of collaboration under this agreement are innovative light water technologies, innovative processing technologies of oxide fuel for light water reactors, and innovative fuel technologies using solvent extraction. Another agreement with the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT) is planned for FY 2005.

4.2.5 U.S./Republic of Korea

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

4.2.6 U.S./OECD

The U.S. teamed with the NEA of the OECD and a number of its 30 member states in order to conduct reactor materials experiments and associated analysis. The U.S. funding team consists of the NRC, EPRI, and DOE. The OECD features a single project.

Table 1: Summary of Awarded Projects

Collaborating Country	Number of Awarded Projects			
	FY-01	FY-02	FY-03	FY-04
Canada	-	-	-	7
European Union	-	-	-	8
France	4	1	-	11
Republic of Korea	-	6	5	6
OECD	-	1	-	-

4.3 Program Participants

The following subsections provide an organizational profile for the five bilateral agreements and a complete list of the I-NERI program participants.

4.3.1 Award Profiles

Figure 2 illustrates the number of I-NERI participants in current collaborations, listed below.

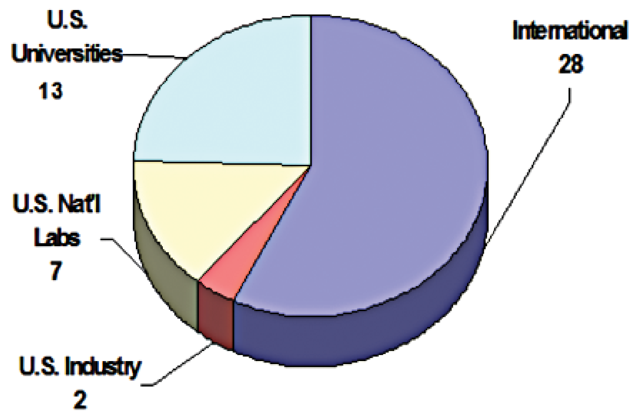


Figure 2. I-NERI Organizational Profile.

4.3.2 U.S. National Laboratories

Argonne
Brookhaven
Idaho
Los Alamos
Oak Ridge
Pacific Northwest
Sandia

4.3.3 U.S. Universities

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

4.3.4 U.S. Industrial Organizations

General Atomics
Westinghouse Electric

4.3.5 International Collaborators

Atomic Energy of Canada Limited
Chalk River Laboratory
Cheju University
Chosun University
Chungnam National University
Commisariat a I'Énergie Atomique
Ecole Polytechnique de Montreal
Framatome, ANP
Gamma Engineering
Gas Technology Institute
Hanyang University
Hitachi, Ltd
Hitachi Works
Joint Research Center Institute for Transuranium Elements
Korea Hydro and Nuclear Power Company
Korean Electric Power Research Institute
Organization for Economic Cooperation and Nuclear Energy Agency
Pusan National University
Republic of Korea Advanced Institute of Science and Technology
Republic of Korea Atomic Energy Research Institute
Republic of Korean Maritime University
Seoul National University
Tohoku University
Toshiba Corporation
University of Bordeaux
University of Manchester
University of Manitoba
University of Tokyo
University of Sherbrooke

5.0 U.S./Canada Collaboration

The Director of the U.S. Office of Nuclear Energy, Science and Technology, William D. Magwood IV, signed a bilateral agreement on June 17, 2003 with the Assistant Deputy Minister of the Department of Natural Resources Canada, Ric Cameron, and the Senior Vice-President Technology of Atomic Energy of Canada Limited, David F. Torgerson.

The first U.S./Canada collaborative research projects were awarded in FY 2004.

This bilateral agreement made Canada the fifth country to participate in the I-NERI program. Canada is a charter GIF member.

5.1 Work Scope

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

- Hydrogen Production by Nuclear Systems
- Sustainable and Advanced Fuel Cycles
- Super-Critical Water-Cooled Reactor Concepts

5.2 Projects

In FY 2004, the initial year of the collaboration, seven research projects were initiated. Appendix A provides a list of the I-NERI U.S./Canada FY 2004 project abstracts. Since 2004 projects are too new to indicate status, only abstracts are included in this report.

6.0 U.S./European Union Collaboration

DOE and the European Atomic Energy Community (EURATOM) signed a bilateral agreement on March 6, 2003. On February 24, 2004, the U.S. and EURATOM selected eight projects for collaboration.

The European Union is a charter GIF member.

6.1 Work Scope

R&D topical areas for the U.S./E.U. collaboration include:

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

6.2 Projects

Eight new projects were initiated in FY 2004. Appendix B provides a list of the I-NERI U.S./E.U. FY 2004 project abstracts. Since 2004 projects are too new to indicate status, only abstracts are included in this report.

7.0 U.S./France Collaboration

U.S. Secretary of Energy, Spencer Abraham, and CEA Chairman, Pascal Colombani, signed a bilateral agreement on July 9, 2001 to jointly fund innovative U.S./French research in advanced reactors and fuel cycle development. The U.S./France collaboration was the first INERI agreement to be fully implemented; U.S./France collaborative research projects were awarded in FY 2001.

This bilateral agreement made France the second country to participate in the I-NERI program. France is a charter GIF member.

7.1 Work Scope

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

- Advanced Gas Cooled Reactors
- Advanced Fuel and Materials Development
- Radiation Damage Simulation
- Hydrogen Production using Nuclear Energy

7.2 Projects

In FY 2001, the initial year of the collaboration, four research projects were awarded. An additional project was awarded in FY 2002. Two projects initiated in FY 2001 were completed and the three remaining projects are planned for completion in FY 2005. Eleven new collaborative projects were initiated in FY 2004. Appendix C provides a list of the I-NERI U.S./France FY 2001 to FY 2002 project summaries and FY 2004 abstracts. Since 2004 projects are too new to indicate status, only abstracts are included in this report.

8.0 U.S./Republic of Korea Collaboration

Director for Nuclear Energy, Science and Technology, William D. Magwood IV, signed the first bilateral I-NERI Agreement on May 16, 2001 with Dr. ChungWon Cho, Director General of Korea's Atomic Energy Bureau, signing for the Republic of Korea's Ministry of Science and Technology. The first U.S./Republic of Korea collaborative research projects were awarded in FY 2002.

Republic of Korea is a charter GIF member.

8.1 Work Scope

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

2002 projects:

- Advanced Instrumentation, Controls, and Diagnostics
- Advanced Light Water Reactor (LWR) Technology
- Advanced LWR Fuels and Materials Technology
- LWR Safety Technology
- Advanced LWR Computational Methods 2003 projects:
- Next generation reactor and fuel cycle technology
- Innovative nuclear plant design
- Advanced nuclear fuels and materials 2004 projects:
- Advanced Gas-Cooled Fast Reactor
- Hydrogen Production by Nuclear Systems
- Advanced Fuels and Materials Development
- Super-Critical Water-Cooled Reactor Concepts

8.2 Projects

In FY 2002, the initial year of the collaboration, six projects were awarded. Five additional projects were awarded in FY 2003 and six new collaborative projects were initiated in FY 2004. The FY 2002 projects will be completed in FY 2005.

Appendix D provides a list of the I-NERI U.S./ Republic of Korea FY 2002 and FY 2003 project summaries and FY 2004 project abstracts. Since 2004 projects are too new to indicate status, only abstracts are included in this report.

9.0 U.S./OECD Collaboration

The U.S. and an international consortium under the auspices of the OEC-DNEA signed a bilateral I-NERI Agreement in March 2002. The U.S./OECD-NEA collaboration has only one project, Melt Coolability and Concrete Interaction, and the agreement specifies equal funding from the OECD-NEA and the U.S for a maximum of five years. The U.S. funding is provided by the U.S. Nuclear Regulatory Commission and the DOE, with DOE providing funding for the initial three year period.

9.1 Work Scope

R&D topical areas for the U.S./OECD-NEA collaboration include:

- Resolving ex-vessel debris coolability issues through a program that focuses on providing both confirmatory evidence and test data for the coolability mechanisms identified in the Melt Attack and Coolability Experiments integral effects tests.
- Addressing remaining uncertainties related to long-term, two-dimensional, molten core-concrete interaction under both wet and dry cavity conditions.

9.2 Project

This project will be completed in FY 2005. Appendix E lists the U.S./OECD-NEA project summary.

Appendix A

U.S./Canada Collaboration Project Abstracts

International Nuclear Energy Research Initiative

Project No.	Title
2004-001-C	High-Temperature Electrolyzer Optimization
2004-002-C	Development of Inert Matrix Fuels for Plutonium and Minor Actinide Management in Power Reactors
2004-003-C	Evaluation of Materials for Super-Critical Water-Cooled Reactors
2004-004-C	ACR Hydrogen Production for Heavy Oil Recovery
2004-005-C	Super-Critical Water-Cooled Reactor Stability Analysis
2004-006-C	Thermal-Hydraulic Benchmark Studies for SCWR Safety
2004-007-C	Lower-Temperature Thermochemical Hydrogen Production

U.S./Canada Collaborators

U.S. National Laboratories

Argonne
Brookhaven
Idaho
Los Alamos
Oak Ridge

U.S. Universities

University of Florida
Iowa State University
Massachusetts Institute of Technology
University of Michigan
University of Notre Dame
University of Wisconsin

U.S. Industry

None

International

Atomic Energy of Canada Limited
Chalk River Laboratory
Ecole Polytechnique de Montreal
Gamma Engineering
Gas Technology Institute
University of Manitoba
University of Sherbrooke

International Nuclear Energy Research Initiative

High-Temperature Electrolyzer Development

Principal Investigator (U.S.): R. Doctor, Argonne National Laboratory (ANL)

Foreign Institution (Canada): R. Sadhankar, Atomic Energy of Canada Ltd. (AECL)

Collaborators: Chalk River Laboratories (CRL), Idaho National Laboratory (INL)

Project Number: 2004-001-C

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

Argonne National Laboratory (ANL) has been building a set of tools and guidelines that the developers of steam electrolysis cells can use to speed up the process of optimizing flow distributor designs. Their work employs Secure Transportable Autonomous Rector (STAR)-CD (a general purpose computational fluid dynamics program) and fluent codes for the computational fluid dynamic (CFD) modeling of a steam electrolyzer, which is being developed by Idaho National Laboratory (INL) in partnership with Ceramatec. The CFD model is linked to ASPEN process simulation software to analyze the balance of plant issues in a high-temperature electrolysis hydrogen production system. Under this joint I-NERI proposal, AECL will provide data on low-temperature electrolyzers for a comparison with the INL electrolyzer.

The two basic problems that limit the operation and efficiency of an electrolysis unit are non-uniform flow within the individual cell cavities and non-uniform flow into separate cells constituting a cell stack. Heat transfer and electrochemical efficiencies will degrade depending upon the extent of these flow non-uniformities. The key objective of the collaboration, then, is to test and compare electrolysis modules and perform computational fluid dynamics calculations to optimize the flow geometry for high-temperature electrolysis applications. ANL will perform CFD calculations on the flow conditions for electrolysis cells and stacks of the INL-based design and the designs identified by AECL. AECL will test other available electrolysis stacks and provide data for the ANL studies.

High-temperature electrolyzers can also benefit from the use of improved materials. The current approach is to use more or less conventional solid oxide fuel cell (SOFC) materials in the electrolyzer, which is treated as an SOFC operated in hydrogen production rather than in hydrogen consumption mode. There are significant operational differences, however, between fuel cell and electrolysis electrochemical reactions, and materials development can help to improve the performance of the electrolyzers.

Proper design and operation of the balance of plant is crucial to overall efficiency of the high temperature electrolyzer system. In particular, because both the hydrogen-rich and the oxygen-rich product streams emerge at high temperatures, the common approach is to quench these streams, with inevitable thermodynamic irreversibilities. Quenching reduces temperatures to a range in which conventional structural materials can provide reasonable service life. Opportunities for process optimization that will consider the trade-offs of cell costs will be pursued. It is possible that membranes useful for separations in this application may emerge from other research groups. Such developments will be integrated into the designs as appropriate.

International Nuclear Energy Research Initiative

Development of Inert Matrix Fuels for Plutonium and Minor Actinide Management in Power Reactors

Principal Investigator (U.S.): M. Meyer, Idaho National Laboratory (INL)

Foreign Institution (Canada): P. Boczar, Atomic Energy Canada Ltd. (AECL)

Collaborators: University of Florida, Los Alamos National Laboratory (LANL), Brookhaven National Laboratory (BNL)

Project Number: 2004-002-C

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

There is interest in the investigation of inert matrix fuels (IMFs) for scenarios involving stabilization or burn down of plutonium in the fleet of existing commercial power reactors. IMFs offer the potential advantage for more efficient destruction of plutonium and minor actinides (MA) relative to mixed oxide (MOX) fuel. Greater efficiency in plutonium reduction results in greater flexibility in managing plutonium inventories and in developing strategies for disposition of MA, as well as a potential for fuel cycle cost savings. Because fabrication of plutonium-bearing (and MA-bearing) fuel is expensive relative to UO_2 in terms of both capital and production, cost benefit can be realized through a reduction in the number of plutonium-bearing elements required for a given burn rate. In addition, the choice of matrix material may be manipulated either to facilitate fuel recycling or to make plutonium recovery extremely difficult. In addition to plutonium/actinide management, an inert matrix fuel having high thermal conductivity may have operational and safety benefits; lower fuel temperatures could be used either to increase operating and safety margins, to uprate reactor power, or a combination of both.

The Canada Deuterium Uranium (CANDU) reactor offers flexibility in plutonium management and MA-burning by virtue of on-line refueling, a simple bundle design, and good neutron economy. A full core of inert matrix fuel containing either plutonium or a plutonium-actinide mix can be utilized, with plutonium destruction efficiencies greater than 90% and high (>60%) actinide destruction efficiencies. The Advanced CANDU Reactor (ACR) could allow additional possibilities in the design of an IMF bundle, since the tighter lattice pitch and light-water coolant reduce or eliminate the need to suppress coolant void reactivity, allowing the center region of the bundle to include additional fissile material and to improve actinide burning. The ACR would provide flexibility for management of plutonium and MA from the existing LWR fleet, and would be complementary to the AFCI program in the U.S. Many of the fundamental principles concerning the use of IMF are nearly identical in LWRs and the ACR, including fuel/coolant compatibility, fuel fabrication, and fuel irradiation behavior. In addition, the U.S. and Canada both have interest in the development of Generation IV Super-Critical Water-Cooled Reactor (SCWR) technology, to which this fuel type would be applicable for plutonium and MA management. An inert matrix fuel with high thermal conductivity would be particularly beneficial to any SCWR concept.

International Nuclear Energy Research Initiative

Evaluation of Materials for Super-Critical Water-Cooled Reactors

Principal Investigator (U.S.): D. Wilson, Oak Ridge National Laboratory (ORNL)

Foreign Institution (Canada):
H. Khartabil, Atomic Energy Canada Ltd. (AECL)

Collaborators: University of Wisconsin, University of Michigan, Gamma Engineering, University of Sherbrooke, University of Notre Dame

Project Number: 2004-003-C

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

To meet the goals of the Generation IV Nuclear Energy Systems Initiative, international collaborations are critical in terms of shared resources and shared expertise. Both Canada and the United States have a shared interest in the development of advanced reactor systems that employ super-critical water as a coolant. The goal of this project is to establish candidate materials for SCWR designs and to evaluate their mechanical properties, dimensional stability, and corrosion resistance. This project will address critical issues related to radiation stability, corrosion, and stress corrosion cracking performance in candidate materials for SCWR.

Super-critical water presents unique challenges to the long-term operation of engineering materials. In addition to developing materials with adequate response to radiation damage, materials must be developed with adequate corrosion and stress corrosion cracking response to high-temperature and high-pressure super-critical water. The high temperature and pressure, along with the generation of oxygen and oxidizing species (OH, H_2O_2, HO_2/O_2) generated by transient radiolytic decomposition, may result in higher corrosion and stress corrosion cracking rates than experienced with other reactor designs. In addition, radiation may accelerate or assist the stress corrosion cracking in the reactor region, and stress corrosion cracking and accelerated corrosion may occur in the preheat and cool-down sections of the circuit. The existing database on the corrosion and stress corrosion cracking of austenitic stainless steel, ferritic-martensitic steels, and nickel-based alloys in super-critical water is very sparse. Data on the behavior of irradiated alloys is non-existent.

International Nuclear Energy Research Initiative

ACR Hydrogen Production for Heavy Oil Recovery

Principal Investigator (U.S.): R. Anderson, Idaho National Laboratory (INL)

Foreign Institution (Canada): R. Sadhankar, Atomic Energy Canada Ltd. (AECL)

Collaborator: Chalk River Laboratories (CRL)

Project Number: 2004-004-C

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

The oil sands in Northern Alberta are estimated to contain 300 billion barrels of recoverable petroleum. As recovered, this is a very heavy crude and requires large additions of hydrogen to produce a synthetic crude comparable to light oils. Production of this resource is quite energy intensive, with the energy currently coming from natural gas. Since this is a far smaller resource than the oil sands, new energy sources are needed. Nuclear-produced steam could supply heat for the Steam-Assisted Gravity-Drainage process currently favored for new oil sands production. In addition, since about one-third of the current natural gas input is used to produce hydrogen, a nuclear source of hydrogen is also desirable.

This project will analyze the engineering feasibility of using the Advanced CANDU Reactor as a source of steam and of hydrogen for enhanced oil recovery, especially as needed by oil sands projects now under construction in and planned for the Athabasca region of Alberta. The main emphasis will be on assessing methods to produce hydrogen.

International Nuclear Energy Research Initiative

Super-Critical Water-Cooled Reactor Stability Analysis

Principal Investigator (U.S.): W. Yang, Argonne National Laboratory (ANL)

Foreign Institution (Canada):
H. Khartabil, Atomic Energy Canada Ltd. (AECL)

Collaborators: Massachusetts Institute of Technology, University of Manitoba, Ecole Polytechnique de Montreal

Project Number: 2004-005-C

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

SCWR design and analysis activities are in progress in the U.S. and Canada within the framework of the Generation IV Nuclear Energy Systems Initiative. One of the important SCWR technology gaps is the power-flow stability related to thermal-hydraulic and thermal nuclear coupled instabilities. The objective of the proposed project is to improve our understanding of potential instability problems at super-critical conditions and to provide SCWR stability analysis tools and appropriate ranges of important design parameters that ensure system stability.

International Nuclear Energy Research Initiative

Thermal-Hydraulic Benchmark Studies for SCWR Safety

Principal Investigator (U.S.): M. Modro, Idaho National Laboratory (INL)

Foreign Institution (Canada): H. Khartabil, Atomic Energy Canada Ltd. (AECL)

Collaborators: Ecole Polytechnique de Montreal, Iowa State University, University of Manitoba, University of Wisconsin

Project Number: 2004-006-C

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

The objectives of this project are to address the critical issue of measuring heat transfer to super-critical water at prototypical SCWR conditions and to develop the tools to predict the SCWR thermal behavior. In addition to actual super-critical water, surrogate fluids at super-critical conditions will be used because: 1) valuable insight of the physical phenomena can be gained with these fluids, and 2) some existing facilities use such fluids, which in general have lower critical pressure and temperature, thus affording significant cost and time savings in constructing and operating experimental facilities. The unique INL Matched-Index-of-Refraction (MIR) flow system will also be used for benchmark measurements of velocity and turbulence fields around the complex geometries involved.

International Nuclear Energy Research Initiative

Lower-Temperature Thermochemical Hydrogen Production

Principal Investigator (U.S.): M. Lewis, Argonne National Laboratory (ANL)

Foreign Institution (Canada): A. Miller, Atomic Energy Canada Ltd. (AECL)

Collaborators: Chalk River Laboratories (CRL), Gas Technology Institute (GTI)

Project Number: 2004-007-C

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

The goal of this project is to evaluate the commercial viability of matching the copper-chlorine (Cu-Cl) cycle with the SCWR being considered within the DOE-NE Generation IV Nuclear Energy Systems Initiative. The availability of alternative hydrogen production processes has been recognized in the DOE-NE Nuclear Hydrogen R&D Plan as an important element in the Nuclear Hydrogen Initiative. The Cu-Cl cycle was identified as a promising low-temperature means of H_2 production from previous studies at ANL. Proof-of-principle experiments and thermodynamic feasibility studies have been completed. The maximum temperature required is 530°C in laboratory-scale experiments. All of the thermal reactions go to completion with minimal side production formation. This cycle also contains an electrochemical step for which the Gas Technology Institute (GTI) has significant expertise. Preliminary experiments with their graphite electrodes and a recently patented membrane design have demonstrated excellent performance. These results, along with the reduced thermal burden and the potential ability to match a wider range of reactor concepts, justify further development of the family of Cu-Cl cycles.

ANL proposes further development of the cycle in order to provide the data for the simulation necessary to cost hydrogen production using Cu-Cl chemistry and heat from the SWCR. ANL proposes the following experimental and simulation work: (1) simulations using commercial process design programs to optimize process design and to calculate cycle efficiency, and (2) experimental work such as reaction kinetics, design, and testing of various reactor configurations, etc., as required for the simulation. GTI proposes further development of the electrochemical cell.

AECL proposes the following: (1) to study the interface of low-temperature thermochemical cycles and other hydrogen production methods with an SCWR, (2) to work with ANL to optimize heat matching between the cycle and the power plant, (3) to determine the overall efficiency of the combined plants, and (4) to project the costs associated with hydrogen production and cogeneration. AECL's focus will also include the costs of distributing and maintaining a dependable hydrogen supply for the various combinations of hydrogen production methods and the SCWR, and comparison with other hydrogen-production technologies.

Appendix B

U.S./European Union Collaboration Project Abstracts

International Nuclear Energy Research Initiative

Project No.	Title
2004-001-E	Development of Inert Matrix Fuels for Plutonium and Minor Actinide Management in LWRs
2004-002-E	Development of Fuels for the Gas-Cooled Fast Reactor
2004-003-E	Lead-Cooled Fast Reactor Engineering and Analysis
2004-004-E	Proliferation Resistance and Physical Protection Assessment Methodology
2004-005-E	Characterization of Nuclear Waste Forms and their Corrosion Products
2004-006-E	Nitride Fuel Fabrication Research
2004-007-E	Molten Salt Technology for Reactor Applications
2004-008-E	Safety Calculations for Gas-Cooled Fast Reactors (GFRs)

U.S./European Union Collaborators

U.S. National Laboratories

Argonne
Idaho
Los Alamos
Oak Ridge
Pacific Northwest

U.S. Universities

None

U.S. Industry

None

International

Joint Research Center Institute for Transuranium Elements (ITU)

Joint Research Center of the European Commission Institute of Energy

International Nuclear Energy Research Initiative

Development of Inert Matrix Fuels for Plutonium and Minor Actinide Management in LWRs

Principal Investigator (U.S.): J. Carmack, Idaho National Laboratory (INL)

Foreign Institution (E.U.): J. Somers, Joint Research Center Institute for Transuranium Elements (ITU)

Collaborators: Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL)

Project Number: 2004-001-E

Project Start Date: October 1, 2004

Project End Date: September 30, 2008

Project Abstract

Commercial power reactors are the only viable option for short- to mid-term (10-20 years) active management of plutonium and minor actinides. Thus, there is worldwide interest in the use of the existing commercial reactor fleet for fuel cycle scenarios involving burn down of plutonium and stabilization of minor actinide inventories. This proposal seeks to develop feasibility data related to the use of IMFs as fuels and minor actinide targets in the existing fleet of LWRs.

IMF offers potential advantages when compared to conventional uranium matrix MOX fuel for plutonium management, allowing for more efficient destruction of plutonium relative to MOX, since the exclusion (or significant reduction) of ^{238}U from the fuel precludes the breeding of additional plutonium. Greater efficiency in plutonium reduction results in greater flexibility in managing plutonium inventories. The potential for fuel cycle cost savings also exists due to the reduced number of rods required to effect a given plutonium burn rate. In addition, IMF can be used in strategies for disposition of MA, particularly americium and neptunium. The choice of matrix material may be manipulated to facilitate either fuel recycling or direct disposal; plutonium recovery can be made relatively straightforward or extremely difficult. Inert matrix fuels having high thermal conductivity may also have operational and safety benefits. Cermet fuel, for example, operates at very low fuel temperatures; this fact can be used to increase operating and safety margins or uprate reactor power. Minor actinide targets contain high MA contents and are designed for heterogeneous MA burning schemes.

As a consequence of the desire to use existing water-cooled reactors, all research and development concepts considered in this proposal will be suitable for loading into present-day and near-term (Generation II and III) power reactor designs. Completion of this program will result in generation of comparative data on the fabrication, properties, and irradiation behavior of several IMF fuel candidates. This data will provide valuable information on the feasibility of IMF for plutonium and MA management in LWRs.

International Nuclear Energy Research Initiative

Development of Fuels for the Gas-Cooled Fast Reactor

Principal Investigator (U.S.): M. Meyer, Idaho National Laboratory (INL)

Foreign Institution (E.U.): J. Somers, Joint Research Center Institute for Transuranium Elements (ITU)

Collaborators: Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL)

Project Number: 2004-002-E

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

Gas-Cooled Fast Reactor (GFR) fuel operating parameters and physical requirements are outside of the envelope of the current experimental nuclear fuel database. Many basic viability issues will need to be addressed experimentally to demonstrate the feasibility of proposed GFR fuels.

Two basic fuel types appear to be viable for GFR service: refractory matrix dispersions and refractory metal or ceramic clad pin-type fuels. This project seeks to develop fuels of these types suitable for GFR service and demonstrate feasibility of these fuels through analysis of fuel requirements, simulation of fuel behavior using fuel performance models, fabrication of fuel specimens, characterization of microstructure and properties, and scoping fuel irradiation testing. Ion irradiation testing of materials will be conducted to simulate material behavior at high irradiation doses for short times. The GFR-F1 test in the Advanced Test Reactor (ATR) (and also the FUTURIX-MI test in Phénix) also addresses basic issues regarding the irradiation behavior of the 'exotic' refractory materials required for GFR fuel service in a neutron-only environment. Ultimately, proof-of-concept for GFR fuel can only be demonstrated through irradiation testing of fissile-bearing specimens. The GFR-F2 scoping fuel irradiation test in the ATR at INL is planned as an integral fuel behavior test that will give the first true indication of fuel feasibility. Data collected as a result of this work will be leveraged to the extent possible to provide data relevant to small modular reactor and LWR inert matrix fuel development efforts.

International Nuclear Energy Research Initiative

Lead-Cooled Fast Reactor Engineering and Analysis

Principal Investigator (U.S.): J. Sienicki, Idaho National Laboratory (INL)

Foreign Institution (E.U.): H. Wider, Joint Research Center for the European Commission Institute for Energy (JRC/IE)

Project Number: 2004-003-E

Project Start Date: April 1, 2004

Project End Date: April 30, 2007

Project Abstract

Lead-Cooled Fast Reactors (LFRs) are ideally suited for the development of modular nuclear power plants for the production of electricity, hydrogen, and, optionally, desalinated water that meet the requirements of a future sustainable world energy supply architecture optimized for nuclear rather than fossil energy. Those requirements include features that facilitate deployment in developing as well as developed nations such as proliferation resistance, sustainability, economy, nearly autonomous operation, and a range of plant power levels compatible with widely varying extents of national nuclear infrastructure and local electric grid development.

The Secure Transportable Autonomous Reactor (STAR) project at ANL has been developing suitable LFR concepts for the future world energy supply architecture:

- STAR-LM (Liquid Metal) for electric power generation with optional desalinated water production at a 400 MWth power level
- STAR-H2 (Hydrogen) for the production of hydrogen and fresh water at a 400 MWth power
- SSTAR (Small Secure Transportable Autonomous Reactor) for electricity and desalinated water production to serve remote sites such as those in Alaska, Hawaii, and elsewhere at about a 25 MWth power level.

The LFR thus forms the basis for a portfolio of plant concepts to match customer needs in developing and developed nations.

Specific features of LFRs, as embodied by STAR-LM, include proliferation resistance; small modular reactor size; sustainable closed fuel cycle; high power conversion efficiency at low temperatures; efficient production of fresh water; nearly autonomous operation; passive safety; natural circulation primary coolant heat transport; and factory fabrication, transportability, and modular assembly at the site.

The objectives of the joint U.S./European Union project are to advance the development of LFRs and supporting analyses applicable to the STAR concepts in the U.S. Successful outcomes will include core neutronics calculations, system thermal-hydraulic and CFD code calculations, meaningful validation of CFD codes under regimes and conditions relevant to LFRs, analyses of operational transients or postulated accidents, and approaches for in-service inspection and maintenance of LFRs during operation.

Lead-Cooled Fast Reactor engineering and analysis will be initiated on the U.S. side within the framework of the Generation IV Nuclear Energy Systems Initiative in the areas of system design and evaluation, instrumentation and control, safety, fuels, and components.

International Nuclear Energy Research Initiative

Proliferation Resistance and Physical Protection Assessment Methodology

Principal Investigator (U.S.): J. Roglans, Idaho National Laboratory (INL)

Foreign Institution (E.U.): A. Poucet, Joint Research Center of the European Commission Institute for Energy (JRC/IE)

Project Number: 2004-004-E

Project Start Date: January 1, 2004

Project End Date: September 30, 2006

Project Abstract

An expert group was formed for the development of an evaluation methodology for proliferation resistance and physical protection (PR&PP) of Generation IV, reporting to the GIF Expert Group created in December 2002. During the first year of operation, the PR&PP developed a methodological approach for the assessment of Generation IV nuclear energy systems for proliferation resistance and physical protection robustness.

Current development has progressed to the point of defining the overall methodological approach, the segmentation of the assessment in three basic elements (threats, methods, measures), and the establishment of the approach to performing each of the three elements. The implementation of each of the elements, however, has not been defined. Only guidelines for the implementation have been provided. The specific implementation of the methodology needs to be completed with the conduct of a development study.

Beginning in calendar year 2004, the conceptual framework developed by the group will be tested and its implementation developed with the use of a specific sample case (development study). This will ensure that the methodology has the appropriate emphasis and structure for subsequent use by Generation IV program policy-makers and concept design teams.

A critical element of the completion of the methodology will be to establish the relationships between the system characteristics and the PR&PP measures. Therefore, sufficient detail about the system characteristics is needed to permit the test application to progress beyond the initial phase.

The scope of the proposed collaborative project is the performance of the development study and the update of the PR&PP assessment methodology during approximately the first year, followed by a demonstration study with the updated methodology in the second year, and the application of the verified methodology to Generation IV systems in coordination with development teams.

International Nuclear Energy Research Initiative

Characterization of Nuclear Waste Forms and their Corrosion Products

Principal Investigators (U.S.): B. Finch, Argonne National Laboratory (ANL), B. Hanson, Pacific Northwest National Laboratory (PNNL)

Foreign Institution (E.U.):
V. Rondinella, Joint Research Center Institute for Transuranium Elements (ITU)

Collaborators: Pacific Northwest National Laboratory (PNNL)

Project Number: 2004-005-E

Project Start Date: January 1, 2005

Project End Date: December 31, 2007

Project Abstract

The objective of this project is to understand and describe the conditions for the formation and overall effects of altered or secondary phases on the behavior of waste form (spent fuel and/or conditioning matrix) during storage and/or in contact with groundwater. The main processes being investigated are the development of waste alteration caused by large accumulation of alpha decay damage (structural, property changes) and the formation of secondary phases on the waste form surface and its effect on the waste corrosion behavior (e.g., corrosion rate, formation of layers blocking further corrosion on the waste form surface, etc.).

1) Monitor the effects of radiation damage accumulation through measurement of relevant quantities/properties (e.g., lattice parameter, macroscopic swelling, hardness, thermophysical properties, etc.) and through microstructure characterization (e.g., transmission electron microscopy [TEM]). Investigate recovery mechanisms and study the accumulation/release behavior of He in different materials (comparison among irradiated fuels and alpha-doped fuels/ matrices).

2) Evaluate possible relationships between observed property changes and corrosion behavior through experiments of aqueous corrosion of “aged” materials (e.g., preferential etching sites, isotopic fractions of released actinides, etc.).

3) Examine corrosion mechanisms and characterize solid corrosion products formed during the aqueous alteration of spent nuclear fuel (e.g., using x-ray photoelectron spectroscopy, transmission electron microscope/energy-loss spectroscopy [TEM-EELS], scanning electron microscopy and energy dispersive spectrometer [SEM-EDS], and x-ray diffraction [XRD]). Determine the fates of various radionuclides following their release from altered spent fuel, especially in terms of secondary phase formation and co-precipitation phenomena. Characterize re-precipitated phases on leached surfaces in terms of composition and potential effects on corrosion.

The relevant experimental capabilities for microstructural and macrostructural analysis of the E.U. and U.S. partners will be applied to the above-mentioned fuel investigations. Experimental setups and data obtained will be jointly discussed. This will include the possible exchange of suitable samples (e.g., “alpha-doped” materials) for full characterization by the U.S. and E.U. facilities or the acquisition of suitable samples.

International Nuclear Energy Research Initiative

Nitride Fuel Fabrication Research

Principal Investigator (U.S.): S. Voit, Los Alamos National Laboratory (LANL)

Foreign Institution (E.U.): S. Fernandez, Joint Research Center Institute for Transuranium Elements (ITU)

Project Number: 2004-006-E

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

Implementation of accelerator-driven systems (ADSs) for the purpose of burning americium and degraded plutonium may possibly enable a reduction of radio-toxic inventories directed to geological repositories by a factor of 100. ADSs are designed to operate on uranium-free fuels in order to maximize transuranic (TRU) destruction rates and thus minimize added costs to the nuclear fuel cycle induced by ADS operation and recycling of the higher actinides. The particular choice of fuel type, however, remains an open question, since very little experience on the performance of uranium-free fuels is available.

While conventional oxide fuels have an indisputable advantage in terms of the vast experience accumulated, the low solubility rate of plutonium oxide in nitric acid appears to require a large-scale development of non-aqueous reprocessing methods with much smaller secondary waste streams than has been achieved to date.

Uranium-free nitride fuels, on the other hand, appear to be compatible with the industrialized PUREX process. They further have the advantage of allowing higher linear ratings, typically a factor of two higher than corresponding oxide or metallic fuels. Thus, the number of fuel pins needed to be fabricated and irradiated in ADS facilities can be halved, with a corresponding beneficial decrease in the number of advanced cores and pump systems subject to potential failure.

The lack of data on uranium-free nitrides necessitates a significant R&D program before nitrides can be qualified and validated as a suitable ADS fuel. Completion of this program will result in generation of comparative data on the fabrication and properties of nitrides. This data will provide valuable information on the feasibility of nitrides for plutonium and minor actinide management in ADS reactors.

International Nuclear Energy Research Initiative

Molten Salt Technology for Reactor Applications

Principal Investigator (U.S.): D. Williams, Oak Ridge National Laboratory (ORNL)

Foreign Institution (E.U.): R. Konings, Joint Research Center Institute for Transuranium Elements (ITU)

Project Number: 2004-007-E

Project Start Date: October 1, 2004

Project End Date: September 30, 2008

Project Abstract

Molten fluoride salts have been studied as fuel and as coolant for nuclear reactor systems in the past, and a considerable development program has been performed at Oak Ridge National Laboratory (ORNL). In addition, clean molten salts are a candidate to transfer heat from a Very High-Temperature Reactor (VHTR) to a thermochemical hydrogen production plant. In the U.S., the VHTR and hydrogen production are high-priority activities. Finally, the fusion community also considers molten fluorides an important option for use as coolants and tritium-breeding media for fusion power chambers. At ORNL, the present research activities include the operating molten-salt corrosion test loop to develop clean molten salts for heat transfer, such as between a VHTR and a hydrogen production plant, and the development of a conceptual design of a modern Molten Salt Reactor (MSR) as part of the Generation IV program.

Presently, there is a growing interest in these and other applications of molten salts. The renewed interest in the U.S., Europe, and Japan in molten salt coolants and the MSR concept in the frame of Generation IV and the partitioning and transmutation (P&T) studies has led to new research programs in several institutes, among them the Institute for Transuranium Elements (ITU) of the JRC. ITU's research activities on molten salts for reactor technology are emerging and they focus on the assessment of the physicochemical properties and the calculation and measurement of phase diagrams of fluoride salts. Fuel salts for thorium/uranium (Th/U) fuel cycles and transuranium transmutation cycles are studied as are coolant salts.

The objective of the present project is to enhance the collaboration between ORNL and ITU through a common work program. The activities will be supported by the institutional funds of both organizations. Moreover, both organizations are involved in the MOST project, a Shared Cost Action of the 5th Framework program of the European Union. The extent and scope of the effort described here will also depend on an eventual follow-up project to MOST.

A total of six tasks have been identified: four represent activities that will strengthen the scientific basis for molten salt technology and one addresses the important technological issue of salt cleaning. A firm scientific basis is essential for molten salt technology, as recent developments indicate higher demands due to the higher operating temperatures and larger temperature gradients. These have an impact on the behavior of the salt itself and the structural materials of the reactor, primarily as a result of corrosion, which is one of the key issues.

International Nuclear Energy Research Initiative

Safety Calculations for Gas-Cooled Fast Reactors (GFRs)

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

Foreign Institution (E.U.): H. Wider, Joint Research Center of the European Commission Institute for Energy (JRC/IE)

Project Number: 2004-008-E

Project Start Date: October 1, 2004

Project End Date: October 31, 2007

Project Abstract

The objective of this Generation IV project is to perform safety calculations for the GFR. This will be carried out in coordination with the ongoing US–France I-NERI GFR development project between CEA-Cadarache and ANL. The project will be initiated within the framework of Generation IV in the area of System Design and Evaluation, "GFR safety system optimization and transient analysis support."

Task 1 - Core Design Optimization: For the larger plant size currently under consideration, optimize the design to enhance core performance, increase fuel margins, and improve safety system response. (ANL)

Task 2 - Primary System/Balance of Plant (BOP) Interfacing: Provide interfacing with the development of the primary system and BOP layout to establish key features and requirements for the performance of the decay heat removal systems (DHRS). This will include the delineation of the various accident sequences. (ANL)

Task 3 - Reactivity Coefficient Study: JRC/IE has already performed many steady-state calculations with HEXNODYN for a CEA Gas-Cooled Fast Reactor with block-type fuel elements. IE could provide the geometrical data, isotopic mixtures, and temperatures that were used. A comparison of the reactivity coefficients is particularly relevant for IE because it gets a considerably smaller reactivity effect for an assumed gas depressurization than comparable analyses. IE will also perform Monte Carlo N-Particle Transport Code (MCNP) calculations for taking into account the neutron streaming effects.

Task 4 - Control Rod Ejection Accident: Perform HEXNODYN calculations with an agreed-upon reactivity excursion and with Doppler as the only negative reactivity feedback. This transient calculation should be compared to similar ANL analyses. (JRC/IE)

Task 5 - Anticipated Transient Without Scram (ATWS) Evaluation: Perform GFR safety calculations with the STAR-CD and the multi-channel European Accident (EAC2) Codes concerning unprotected loss-of-flow, depressurization, loss-of-power and reactivity accidents, and emergency decay-heat removal. The EAC2 code will be limited to the treatment of GFRs with fuel pins. A simplified few channel model using block-type fuel elements may be developed. (JRC/IE)

Task 6 - DHRS Evaluation: Evaluate and implement the Integrated System-type thermal-hydraulic models required for the DHRS analysis. Perform the analysis of decay heat removal capability for candidate designs under various postulated accident conditions. The performance of the secondary (guard) containment will be included. (ANL)

Task 7 - System Design Integration: Integrate the candidate DHRS design into the evaluation of the primary system and BOP definitions, the core performance, and the assessment of the transient and ATWS response of the plant. (ANL)

Appendix C

U.S./France Collaboration
Project Summaries/Abstracts

International Nuclear Energy Research Initiative

Project No.	Title
2001-002-F	Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum
2001-003-F	Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels
2001-006-F	OSMOSE – An Experimental Program for Improving Neutronics Predictions of Advanced Nuclear Fuels
2001-007-F	Nano-Composited Steels for Nuclear Applications
2002-001-F	High-Efficiency Hydrogen Production from Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting
2004-001-F	Hydrogen Process to High Temperature Heat Source Coupling Technology
2004-002-F	OSMOSE – An Experimental Program for Improving Neutronic Predictions of Advanced Nuclear Fuels (continuation)
2004-003-F	Thermal-Hydraulic Analyses and Experiments for GCR Safety
2004-004-F	SiC/SiC for Control Rod Structures for Next Generation Nuclear Plants
2004-005-F	Assessment of Existing Physics Experiments Relevant to VHTR Designs
2004-006-F	GFR Physics Experiments in the CEA-Cadarache MASURCA Facility
2004-007-F	Evaluation of Materials for Gas-Cooled Fast Reactors
2004-008-F	Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum
2004-009-F	Development of Fuels for the Gas-Cooled Fast Reactor
2004-010-F	PRA-Aided Design of Advanced Reactors with an Application to GFR Safety-Related Systems
2004-011-F	Thermochemical Hydrogen Production Process Analysis

U.S./France Collaborators

U.S. National Laboratories

Argonne
Brookhaven
Idaho
Los Alamos
Oak Ridge
Pacific Northwest
Sandia

U.S. Universities

University of California, Santa Barbara
Iowa State University
Massachusetts Institute of Technology
University of Michigan
University of Wisconsin

U.S. Industry

General Atomics

International

Commisariat a l'Énergie Atomique

Framatome, ANP

Joint Research Center Institute for Transuranium Elements

University of Bordeaux

International Nuclear Energy Research Initiative

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

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

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

Collaborators: Brookhaven National Laboratory (BNL); General Atomics (GA); Massachusetts Institute of Technology (MIT); Oak Ridge National Laboratory (ORNL); Framatome - ANP (FRA-ANP), Lyon

Project Number: 2001-002-F

Project Start Date: March 1, 2002

Project End Date: February 28, 2005

Research Objective

The project objective is to design a GFR with a high level of safety and full recycling of the actinides, that also is highly proliferation-resistant and attractive in terms of economics. This three-year project started in March 2002.

Research Progress

The project has reached the stage where the effort has focused on the characterization of point designs in Year 3. In the initial phase of the project, exploratory studies were performed on a broad range of fuel forms and types, core configurations, coolant types, and primary system/ BOP concepts. Design goals and criteria were specifically formulated for the GFR, and documented to meet the Generation IV criteria/ metrics on economics, sustainability, safety, and non-proliferation. During the course of the initial phase, the major focus of the effort was on innovative concepts to significantly improve the safety level of the GFR over that attained for the Gas-Cooled Fast Reactor (GFR) design of twenty-five years ago, particularly in the area of depressurized decay heat-removal accidents, without compromising the economic competitiveness of the design. Concurrently, an effort was made to preserve the sustainability goal for the GFR core design, which, as all fast reactor core designs, has great flexibility in the choice of fuel cycles. This neutronic flexibility of the GFR core allows the non-proliferation criteria to be successfully incorporated into the design in parallel with the exploration of the sustainability goal. Based on this work, the project is no longer considering the pebble fuel form, oxide fuel, in-core heat exchangers or conduction cooldown. To summarize the conclusions of the exploratory studies, this collaboration between CEA-France and ANL-US partners:

- Developed reactor core design concepts which meet the goal of sustainability (conversion ratio = 1.0) with low proliferation risk (no external blankets) and homogeneous recycling of minor actinides.
- Developed primary system passive decay heat removal concepts for use in combination with the active systems.
- Developed a modular GFR concept (600 MWth), shown in Figure C-1, which incorporates the sustainable core design and passive heat removal features, and makes maximum use of high-temperature VHTR technology (direct cycle, cogeneration capability) to minimize R&D costs and development time.

With the completion of the exploratory studies, the I-NERI project began a series of trade studies focused on a number of design options identified by the exploratory assessments. The focus was on unit size and a scale-up to a large plant size (2400 MWth) to take advantage not only of economics of scale but also to benefit from the decreased neutron leakage which allows reaching the conversion ratio of 1.0 with less challenging fuels while still working at a significant power density (100 MW/m^3). In particular, a large factor was to try to reduce the cost of passive design features specifically introduced to significantly improve the decay heat removal performance of the GFR during depressurization accidents with concurrent total loss of a/c power. The exploratory studies had concluded that natural convection should be the passive decay removal mechanism of choice. This choice led to the selection of a double containment vessel (guard/proximate) which increases the capital cost of the plant. During the Year 2 trade study phase, significant effort was concentrated on options to reduce this cost. At the start of project year 3, point designs were assembled from different options and combinations, which have been filtered down to:

1. Fuel choice: Dispersed fuel in plate subassemblies as the reference; silicon carbide (SiC)-cladded pellets in pin subassemblies as a back-up. The selected actinide compound is carbide in the design studies, but nitride remains a possible candidate.

2. Unit size: 2400 MWth.

3. Power density: 100 MW/m^3.

4. Natural convection passive decay heat removal approach, which should be combined with active means (low power circulators) in a well-balanced mix to be refined, but this does not exclude alternative options (search for conduction paths, heavy gas injection, etc.) on which some effort still has to be devoted.

5. Direct Brayton cycle option remains the reference but considerations of the indirect super-critical CO_2 BOP cycle with an equivalent cycle efficiency have also been included.

The conclusion from this work is that encouraging results on the feasibility of the GFR have been obtained. In particular, the proposed concepts are characterized by attractive features regarding safety, which take advantage of the attractiveness of helium in terms of neutronic quasi-transparency and the enhancement of the Doppler effect in connection with the candidate materials selected for the fuel and structural materials. Work is still needed to refine the safety approach and the development of challenging fuels remains a key issue.

Planned Activities

Effort will be completed on the point design characterization studies of Year 3 and the results will be documented. These reports will form the chapters of the System Design Report (SDR) on the GFR design and safety approach.

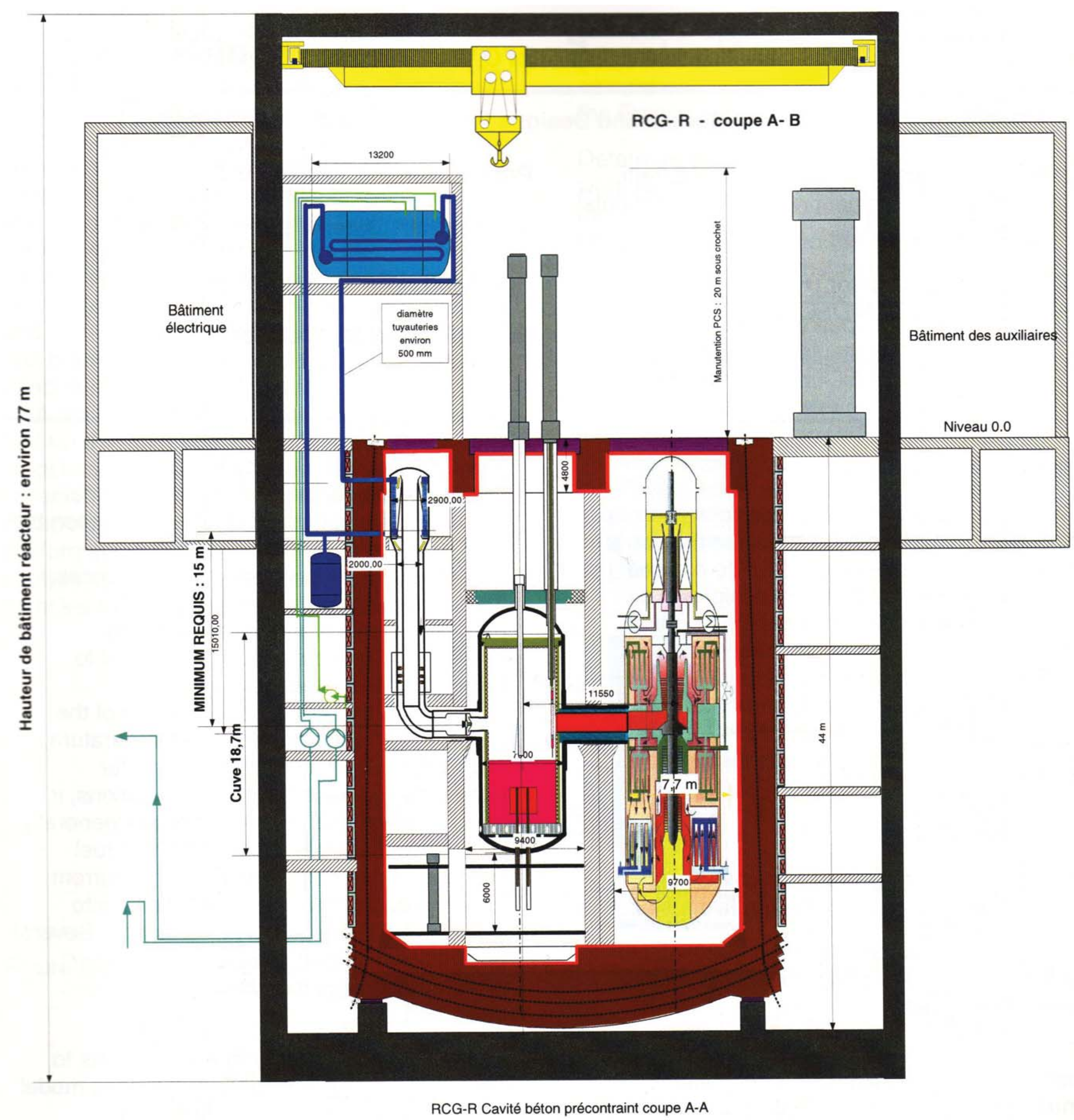


Figure C-1. 600 MWth Modular GFR Concept.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): D. Petti, Idaho National Laboratory (INL)

Principal Investigator (France): M. Phélip, DEN/DEC/SESC CEA

Collaborators: Massachusetts Institute of Technology (MIT)

Project Number: 2001-003-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

Research Objective

The objective of this I-NERI project was to develop improved fuel behavior models for gas reactor coated-particle fuels and to explore improved coated-particle fuel designs that could be used reliably at very high burnups and, potentially, in Gas-Cooled Fast Reactors. Project participants included the INL, CEA, and the Massachusetts Institute of Technology (MIT).

Research Progress

To accomplish the project objectives, work was organized into the following five tasks:

Task 1 - Information relative to material property databases and existing fuel models was exchanged.

Task 2 - An integrated fuel model was developed that includes the effects of multi-dimensional failure mechanisms and phenomena not included beforehand in the models.

Task 3 - Deterministic fuel performance calculations were performed to evaluate the capacity of classical tristructural isotropic (TRISO) fuel to reach extended burnups and thereby establish requirements for fuel materials.

Task 4 - The feasibility of using particle fuel in a fast neutron environment was investigated.

Task 5 - An irradiation testing strategy for prototype fuel particles was developed.

The CEA and INL exchanged their databases on coated particle fuel material property correlations. Comparison between U.S. and European data revealed many similarities and a few important differences. These correlations are used in model predictions of fuel performance during irradiation. Such predictions are useful to understand the interplay of important phenomena that could occur outside of the existing irradiation envelope of temperature, burnup, and fast neutron fluence. After reviewing and assessing the correlations, it was observed that property data are generally lacking for materials exposed to high fuel burnups and neutron fluences. This current lack of data will introduce uncertainty into model predictions of fuel performance. Several key material properties that affect fuel performance were identified and briefly described.

The INL continued, from earlier efforts, to develop an integrated fuel performance model called PARFUME with the objective of physically describing both the mechanical and physicochemical behavior of particle fuel under irradiation. In addition to the traditional pressure vessel failure mode, the model includes multi-dimensional failure mechanisms. These mechanisms include particle failure due to shrinkage cracks in the inner pyrolytic carbon (IPyC) layer, partial debonding between the IPyC and SiC layers, particle asphericity, and kernel migration. A statistical approach is used to simulate detailed finite element calculations and allows for changes in fuel design attributes (e.g., thickness of layers, dimensions of kernel) as well as changes in important material properties which increase the flexibility of the code. Time-dependent thermal modeling capabilities for either spherical or cylindrical fuel elements and for individual fuel particles are included in PARFUME. The thermal model accounts for changes in fission gas as well as shrinkage and swelling of the particle layers and kernel with the potential for formation of a gap between the buffer and IPyC layer. This effect is illustrated in Figures C-2 and C-3. The CEA has developed a finite element, particle fuel simulation model called ATLAS. This model and the material properties with constitutive relationships have been incorporated into a more general software platform called Pleiades. Pleiades is able to analyze various fuel geometries from single particles to fuel elements and is able to account for the statistical variability in coated particle fuel. Preliminary benchmark calculations show good agreement between the French and U.S. models. Deterministic fuel performance calculations were performed to evaluate the ability of particle fuel to reach extended burnups. These calculations illuminated the requirements that the fuel be able to withstand the stress levels and internal chemical environment that would be developed as a consequence of extended fuel life. For these evaluations, the INL developed, partially with internal funding, a fission product chemistry and transport module and incorporated it into PARFUME. This module calculates CO production, shown as the INL model in Figure C-4, release of gaseous fission products into particle void volume, and release to birth ratios for selected isotopes.

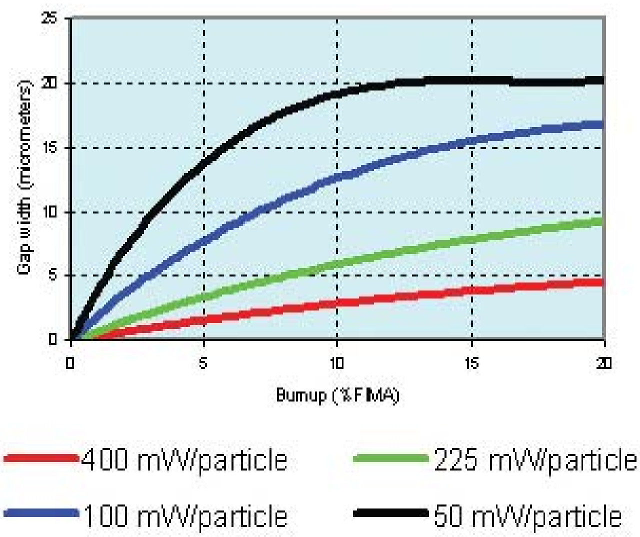


Figure C-2. Gap Development in a Prismatic Block Core as a Function of Burnup and Particle Power.

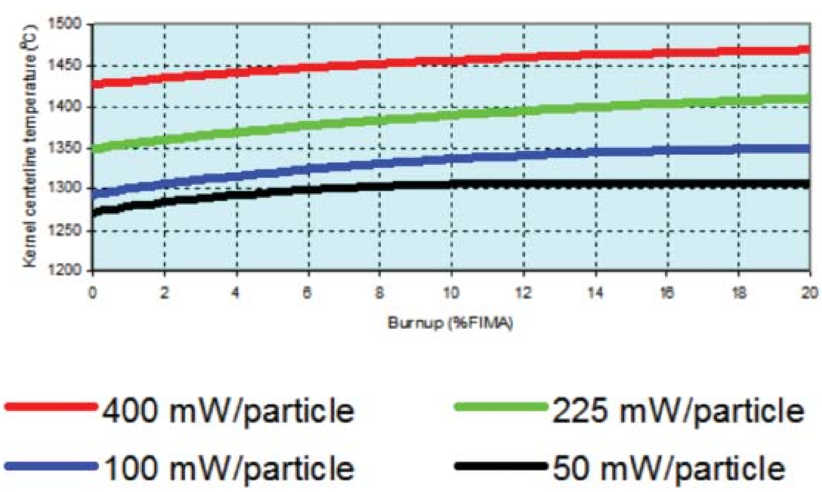


Figure C-3. Kernel Centerline Temperatures in a Prismatic Block Core as a Function of Burnup and Particle Power.

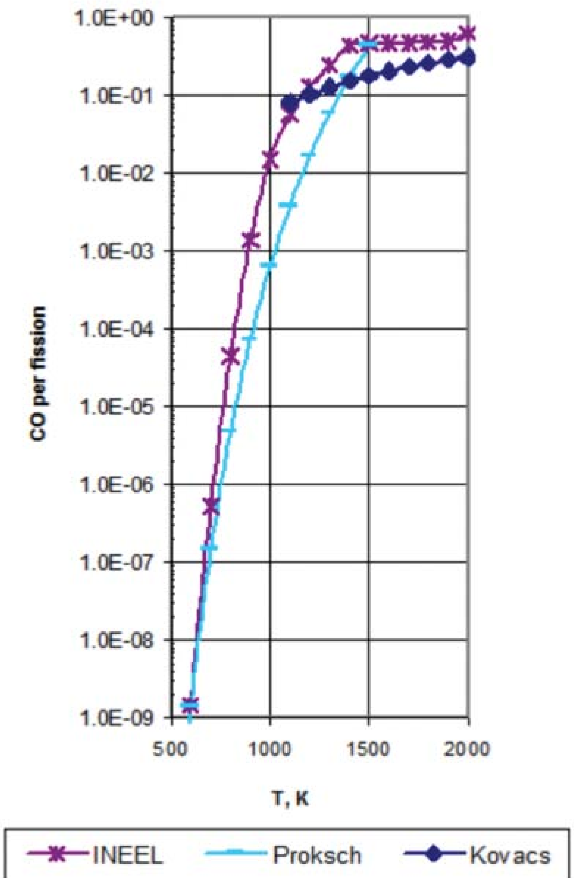


Figure C-4. INL Model Predictions of CO Yield per Fission vs. Temperature for a Case with Pure UO_2 Fuel Compared with German (Proksch) and Historical U.S. (Kovacs) at t = 573 Days (approximately 50MWd/kg).

An extensive review of the literature was performed to understand the physical mechanisms for fission product transport in PyC and SiC. Mechanisms include: vapor transport via Knudsen diffusion for gaseous fission products and illustrated for Kr in Figure C-5, intercalation of alkali and alkali-earth fission products like Cs and Sr in the PyC layers, grain boundary diffusion, surface diffusion, and bulk diffusion. Diffusivities for Ag, Xe, Cs, and Sr have also been gathered from the literature. In addition, scoping calculations were performed using a diffusion and trapping code called TMAP to model fission product transport from the particles. The code can model diffusion and trapping of multiple species and can model diffusion in the presence of a temperature gradient (the so-called Soret Effect). The code also has a thermal model that has been used to determine the temperature distribution and thermal gradient in each of the layers of the coated particle. Sensitivity studies have been conducted to look at thermal diffusion effects which are most important in the low density buffer where large thermal gradients could be expected depending on the power density in the fuel particle. The INL also investigated the effects of SiC layer thinning which may result from interactions with fission products. Preliminary results indicated that widening of the thinned area more strongly increases particle failure probability than does deepening of the thinned area. A metallic Pd – SiC interaction model was developed and, when combined with the SiC layer thinning evaluation, will form an integrated model.

MIT performed diffusion couple experiments to study Ag and Pd transport through SiC. Results indicate that Knudsen pressure-driven diffusion is the most likely mechanism for silver transport. This finding would imply transport via nanoporosity or nanocracks in the SiC. Knudsen and viscous pressure-driven diffusion calculations have been performed to examine transport through sub-micrometer sized pores or cracks in the SiC layer.

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

Potential irradiation of prototype fuel particles in the ATR was examined. It was determined that the ATR would provide a near optimum balance of burnup accumulation and fast neutron fluence for irradiation testing of particle fuel. The CEA has investigated particle fuel irradiation in the French Material Testing Reactor, OSIRIS. Initial experiments utilizing historical German fuel and newly manufactured French fuel have been planned and are being implemented under a separate European program.

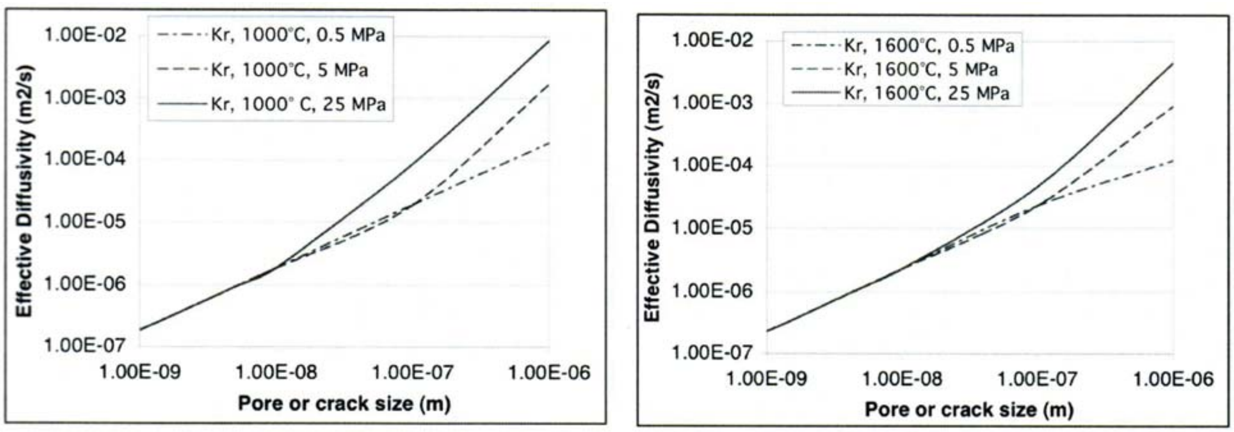


Figure C-5. Effective Diffusivities for Knudsen and Viscous Diffusion.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): R. Klann, Argonne National Laboratory (ANL)

Principal Investigator (France): J.-P. Hudelot, DRN/DER/SPEx/LPE CEA

Collaborators: University of Michigan

Project Number: 2001-006-F

Project Start Date: September 30, 2001

Project End Date: September 30, 2004

Research Objective

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

Research Progress

The collaborative project is defined by five major tasks: reactor modifications, reactor modeling, sample fabrication, experiments, and data analysis.

Task 1 – Reactor Modifications

Modifications to the MINERVE facility were completed previously. No additional modifications were performed during 2004.

Task 2 – Reactor Modeling

The analytic effort is being performed using separate suites of reactor analysis codes in the U.S. and in France. The effort will allow improvement of the codes.

The Material Specification Report for the MINERVE reactor has been issued and serves as a reference for the calculational models. It was also translated into English and issued as an ANL report. Monte Carlo and deterministic models have been created to calculate control rod reactivity worths, axial and radial power profiles, spectral indices, and ^{238}U modified conversion ratio for the R1-UO_2 and R1-MOX configurations. The deterministic model is also used to calculate the reactivity worth of UO_2 and borated calibration samples in the R1-UO_2 and R1-MOX configurations.

The Monte Carlo model uses the MCNP-4C code system with the continuous energy cross sections of the ENDFB-VI library and fully describes the neutronic region of interest of the MINERVE reactor.

The deterministic model is based on the REBUS code system. A radial view of the R1-UO_2 model is shown in Figure C-6. The model uses an XYZ geometry with approximately 200×200×100 mesh cells. The self-shielded cross sections are provided by the one-dimensional transport code system WIMS-ANL 5.07 and collapsed to 7 groups.

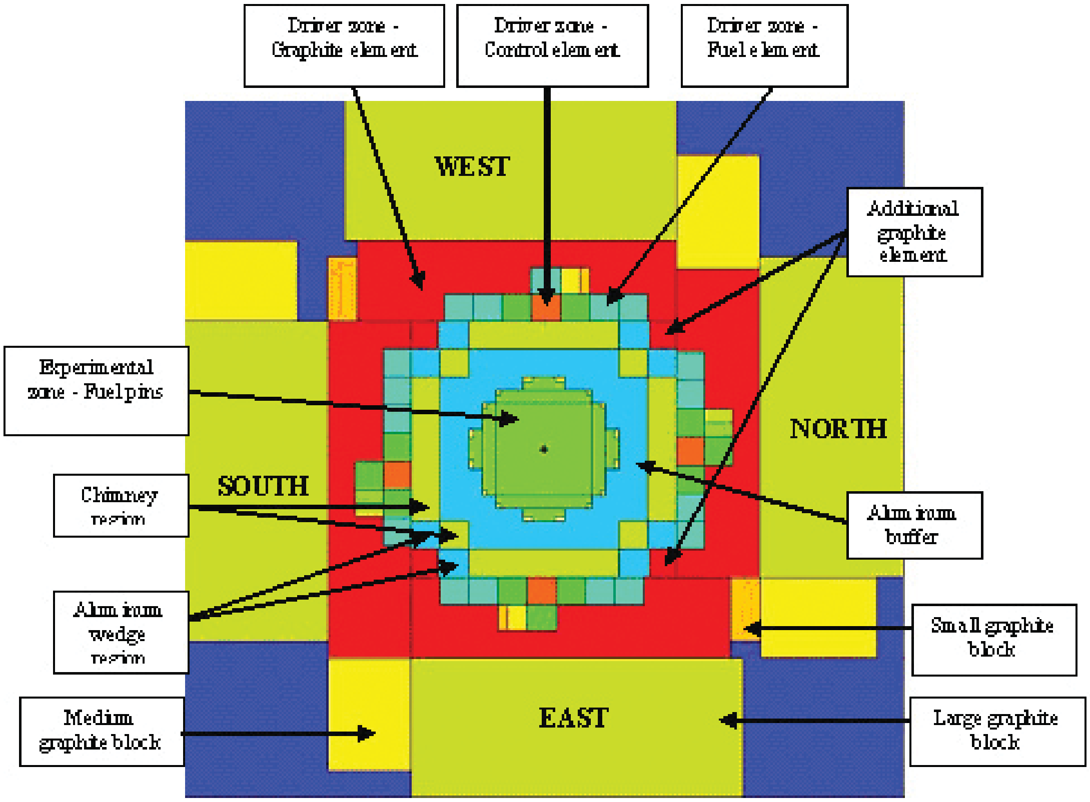


Figure C-6. Radial View of the REBUS Model of the $RIU0_2$ Configuration.

Task 3 – Sample Preparations

The OSMOSE program requires the fabrication of 21 oxide samples containing separated actinides (^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm, ^{245}Cm). The samples consist of assembled fuel pellets containing the isotopes of interest and a double zircaloy cladding.

The OSMOSE oven installation and checkout was completed in early January 2004. The safety authority of the Atalante facility of CEA Marcoule gave the final authorization in the middle of January 2004 to operate the OSMOSE oven. The oven has operated normally, except from July to October when it was shut down because of a failure of the power supply connector.

The analysis and isotopes preparations continued in 2004. In particular, several purifications routes have been undertaken: purification of ^{241}Am by oxalic precipitation, preparation of ^{239}Pu from metal standard Metallic Plutonium #2 (MP2), and purification of ^{234}U from ^{238}Pu (shown in Figure C-7). The effort in preparing isotopes will be maintained continuously in order to allow the on-time fabrication of the OSMOSE samples.

In the OSMOSE oven, fuel pellets were fabricated for 5 sets of samples: $UnatO_2$, UO_2 + ThO_2, UO_2 + $^{234}UO_2$, and two samples of UO_2 + $^{237}NpO_2$. In every case, the pellets showed mean densities better than the specification of 95% of theoretical density. ^{239}Pu and URT pellet fabrication is underway. The entire set of 21 oxide OSMOSE samples is planned to be completed by the end of 2005.

The equipment for cladding was also tested. The helium filling equipment is now ready for use in checking the quality of the welding of the OSMOSE pins (He leak test). During these studies, the dimensional characteristics of the welding cap required modification. The change in these specifications has been reviewed by CEA and accepted.

Currently, the two laser welders in the glovebox and in the hot cell are operational after a major repair by the manufacturer. Qualification of the welding equipment is expected in November 2004.

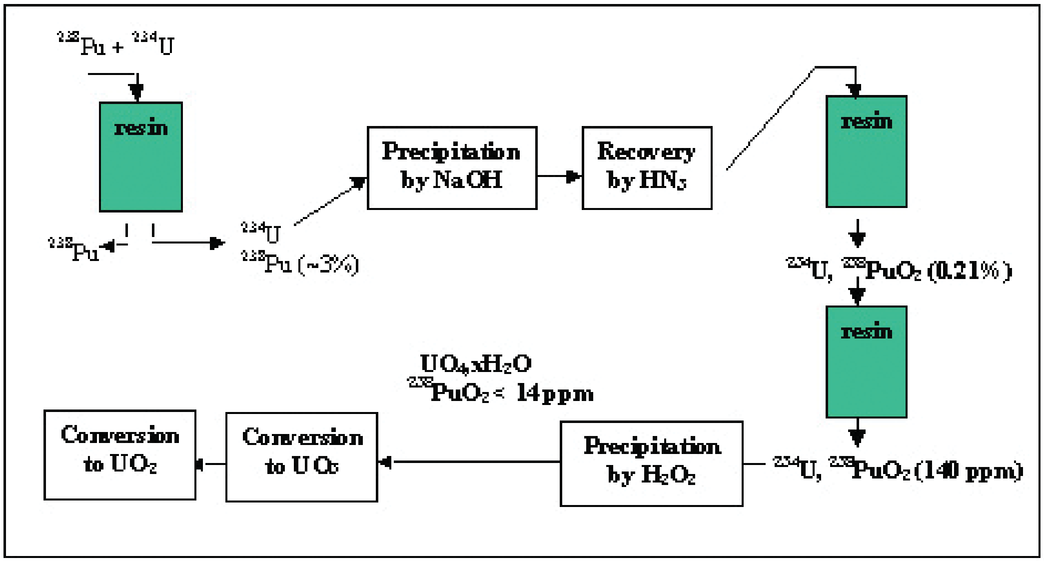


Figure C-7. Purification of ^{234}U from ^{238}Pu.

Task 4 – Experiments

The objective of the measurements is to characterize the neutron flux, spectrum, and power distribution of the reactor to qualify computational models and neutron cross sections.

The goal of the calibration measurements is to demonstrate the oscillation technique on known and calibrated samples, and to support the development of the analytic technique for reactivity-worth measurements of separated actinides and cross-section evaluations.

The first part of the year was devoted to completing the HTC program (a French research program on high purnup nuclear fuels) in the R1-MOX lattice and to processing experimental data. Axial profile measurements performed by gamma scanning and by ^{235}U and ^{237}Np fission chambers were compared with each other to prove their excellent agreement. The axial bucklings were determined with special attention to the statistical treatment of the data and determination of experimental uncertainties.

Spectral indices measurements were treated in three different ways, considering the three different sets of data available. The results were compared and synthesized in a technical report by CEA Cadarache. Radial power profile measurements were re-interpreted with an improved methodology for treatment of the uncertainties, resulting in reduced and more accurate uncertainties. Reactivity worths of the control rods were also treated with a special attention to the treatment of uncertainties from the nuclear data, the numerical treatment of the Inhour equation, and the experimental data.

The second part of the year was devoted to the experiments performed during the VALMONT program in the R1-UO_2 lattice. Calibration curves obtained by the oscillation technique with calibration samples were measured in the R1-UO_2 lattice. Axial power profile was measured by integral gamma spectroscopy on three UO_2 fuel pins located in the central cell of R1-UO_2 and near the central cell. After statistical treatment, axial bucklings were determined. The modified conversion ratio of ^{238}U was also measured on the same three UO_2 fuel pins as for the axial profile. This experiment was performed by single peak gamma spectroscopy on ^{239}Np and ^{143}Ce gamma rays. A very good consistency between the results was observed, with an accuracy of about 2%.

Task 5 – Data Analysis

During year 2004, the data analysis tasks addressed the analysis and reduction of data for each series of measurements, and the comparison with calculated results for the R1-UO_2 and R1-MOX core configurations.

Spectral indices measurements ($^{239}Pu/^{235}U$, $^{241}Pu/^{239}Pu$, and $^{237}Np/^{239}Pu$), performed in the central oscillation channel, agree within 1% with the calculated values using MCNP and the ENDFB-VI data library except for the $^{237}Np/^{239}Pu$ spectral index in the R1-UO_2 configuration, which agrees within two standard deviations.

^{238}U modified conversion ratios have been calculated with MCNP for the R1-MOX configuration. The measured and calculated modified conversion ratios agree within two standard deviations.

Axial fission rate distributions have been calculated using the REBUS and MCNP models. Figure C-8 shows the axial fission rate distribution obtained by the ^{235}U and ^{237}Np fission chamber measurement in the oscillation channel and the corresponding distributions calculated with MCNP. The axial bucklings are estimated on a region carefully based on parametric studies.

For the R1-UO_2 configuration, the axial bucklings calculated with REBUS and MCNP agree with the experimental values within one standard deviation, except for the ^{237}Np profile. For the R1-MOX configuration, calculated and experimental values agree within two standard deviations, except for the ^{237}Np profile.

Both axial and spectral indices measurements show discrepancies between calculation and experimental results when using ^{237}Np that may result from inaccuracy of the ENDFB-VI cross section.

Radial fission rate distributions have been calculated using the REBUS and MCNP code systems and compared with experimental values for the R1-MOX configuration. The MCNP model predicts the experimental values well for the pins away from the MOX/uranium oxide (UOX) interface. The REBUS models underestimated the power in the UOX pins and overestimated the power in the MOX pins.

Reactivity worth of control rods has been calculated with REBUS and MCNP. Experimental values deduced from the point kinetics need to be corrected to account for spatial and energy effects to be used as a reference for the calculated values.

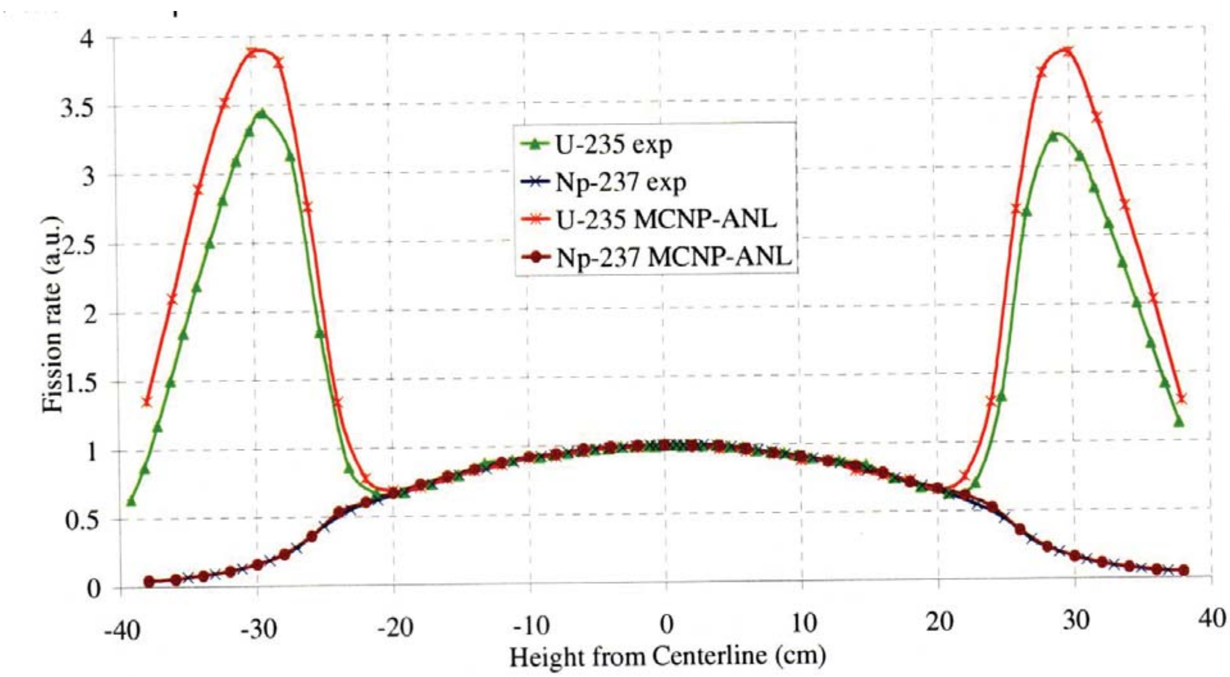


Figure C-8. Axial Fission Rate Distribution for the R1-MOX Configuration.

The reactivity worth of the calibration samples have been calculated with REBUS. Figure C-9 shows the experimental and calculational results for the R1-UO_2 configuration and the linear relationship. The UO_2 samples are very well predicted by the calculation, whereas the borated calibration samples are not as well predicted. The same results are observed for the R1-MOX configuration. The poor agreement between the experimental signal issued from the oscillations of the borated calibration samples and their calculated reactivity is thought to come from uncertainties in the composition of the borated samples and a possible migration of the boron to the periphery of the sample during the sintering of the fabrication process, inducing self-shielding effects. New borated calibration samples with a well-known composition are being fabricated and will allow confirmation of these conclusions.

Planned Activities

The OSMOSE project is continuing in FY 2005 within the framework of the Generation IV Advanced Reactor Program. The main objective in 2005 is to begin performing oscillation measurements with the R1-UO_2 reactor configuration using the OSMOSE separated actinide samples. To support this objective, the remaining OSMOSE samples will be fabricated. In addition, estimates of the reactivity-worth of the samples will be calculated as part of the pre-analysis campaign. Post-measurement analysis will begin toward the end of 2005, and 2006 will be devoted to performing the same series of measurements with the OSMOSE samples in the R1-MOX configuration.

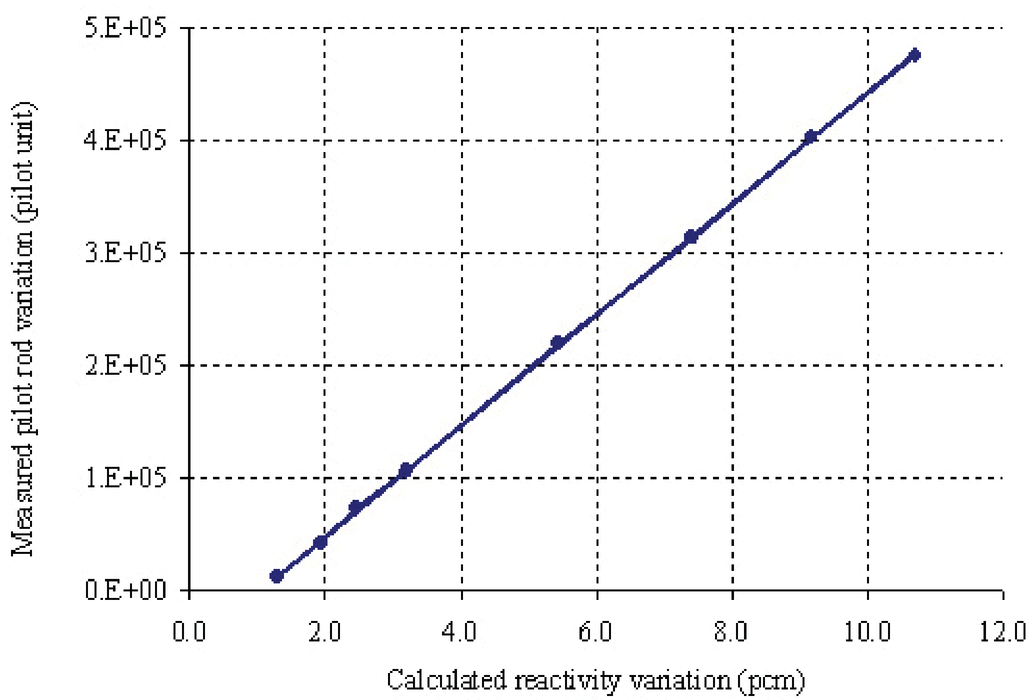


Figure C-9. Reactivity-Worth of UO_2 Calibration Samples in R1-UO_2 Configuration.

International Nuclear Energy Research Initiative

Nano-Composited Steels for Nuclear Applications

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

Principal Investigator (France): A. Alamo, Commissariat a' l'Energie Atomique (CEA)

Collaborators: University of California, Santa Barbara (UCSB)

Project Number: 2001-007-F

Project Start Date: October 1, 2001

Project End Date: September 30, 2005

Research Objective

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

Research Progress

Summary

Substantial progress has been achieved in the three years of this project. Advanced microstructural analysis techniques have been applied to develop our understanding of how and when the desirable nanometer-sized oxide clusters form in the mechanically alloyed materials. New insight into the differences and similarities of the reference ODS alloys MA957 and 12YWT was obtained and an improved mechanical property database on these materials was developed. Alloy processing and fabrication variables were examined in an extensive investigation that employed the different mechanical alloying and alloy consolidation methods available at the three institutions. Most significantly, this work has led to a recipe for reproducibly fabricating an alloy with the desirable microstructure containing a high density of nanometer-sized oxide clusters to provide high temperature strength. This was the primary goal of the research. Further refinement of this recipe, the verification of the material's mechanical properties and its stability under irradiation and high temperature thermal aging, will be carried out during the no-cost extension period (see below).

Current Results

A primary tool of the microstructural characterization effort has been atom probe tomography (APT) at ORNL. This instrument provides the near atomic scale imaging required to observe the very small (~2 to 5 nm diameter) oxide clusters. A typical example is shown in Figure C-10, where 3-dimensional atom maps are shown for the primary alloying elements in one of this project's developmental alloy heats designated U14YWT1150. This is an alloy with 14 wt-% Cr and 0.25 wt-% Y_2O_3 that was ball-milled and consolidated at UCSB using hot isostatic pressing (HIP) at 1150°C. The atom probe specimen contained several fine oxide clusters and a grain boundary. As shown in Figures C-10(b) and (c), chromium (Cr) and tungsten (W), respectively, were found to be slightly enriched at the grain boundary. Carbon and nitrogen contents were very low, but these elements also had a tendency to segregate to the grain boundary, shown in Figures C-10(d) and (e). However, there was no evidence of carbide/nitride formation, or of the presence of any other second phases associated with the alloy W (e.g., Fe_2W) or Cr (e.g., σ-phase) contents. As shown in Figures C-10(f) through (j), the Y, Ti, and O did not segregate to the grain boundaries; these elements were either homogeneously distributed or associated with the fine oxide clusters. Three of these features are indicated in Figure C-10(j). These appear to be very similar in composition to those previously observed by APT in reference ODS alloys MA957 and 12YWT.

Although the clusters were not homogeneously distributed throughout the alloy heat shown in Figure C-10, additional work discussed in the full report describes the processing that is currently being employed to obtain the more desirable homogeneous distribution. An example of APT data from an examination of alloy heat OE14YWT-SM3 is shown in Figure C-11. This heat was ball-milled and hot-extruded at ORNL. The APT data was obtained on a new version of the instrument known as the local electrode atom probe (LEAP), which permits collection of much larger data sets. The larger volume of material makes the clusters somewhat more difficult to observe in Figure C-11. Small angle neutron scattering has also been extensively used to monitor the evolution of the oxide particles as a function of processing time and temperature; this is discussed in more detail in the full report.

Although mechanical testing of irradiated specimens from the developmental heats will be carried out during the no-cost extension, initial results have been obtained by the CEA for the reference ODS alloy MA957. These data are compared to similar conventional martensitic alloys in Figure C-12. The effect of irradiation on the yield strength is shown in Figure C-12(A) and on ductility in Figure C-12(B). The initial yield strength of MA957 is greater than either of the other alloys, but the radiation-induced increase in yield strength is less for MA957. This reduction in hardening is consistent with the observation that the MA957 maintains greater ductility than the other alloys. The MA957 results are an encouraging indicator that the radiation performance of the developmental heats perform similarly.

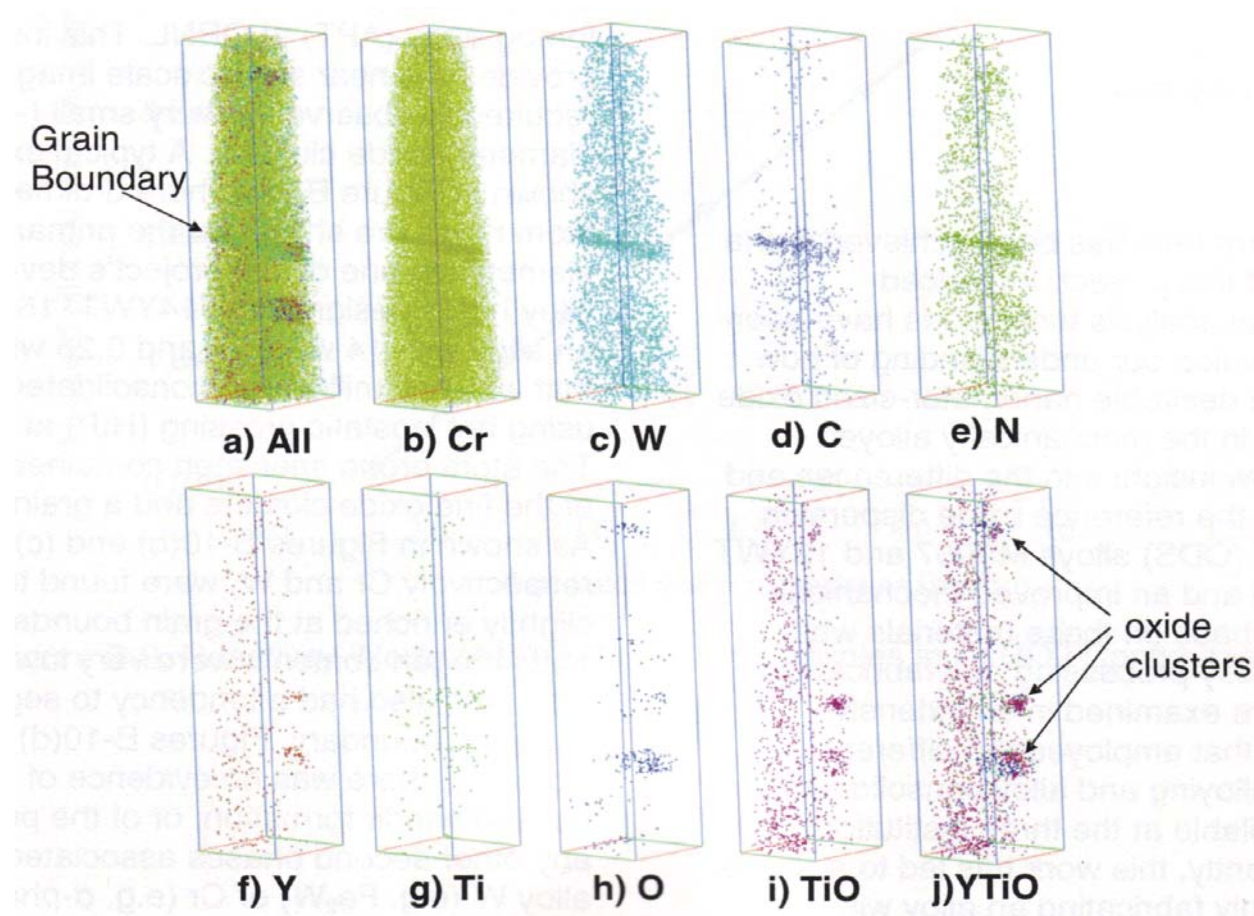


Figure C-10. Atom Maps for U14YWT 1150 Created Using APT.

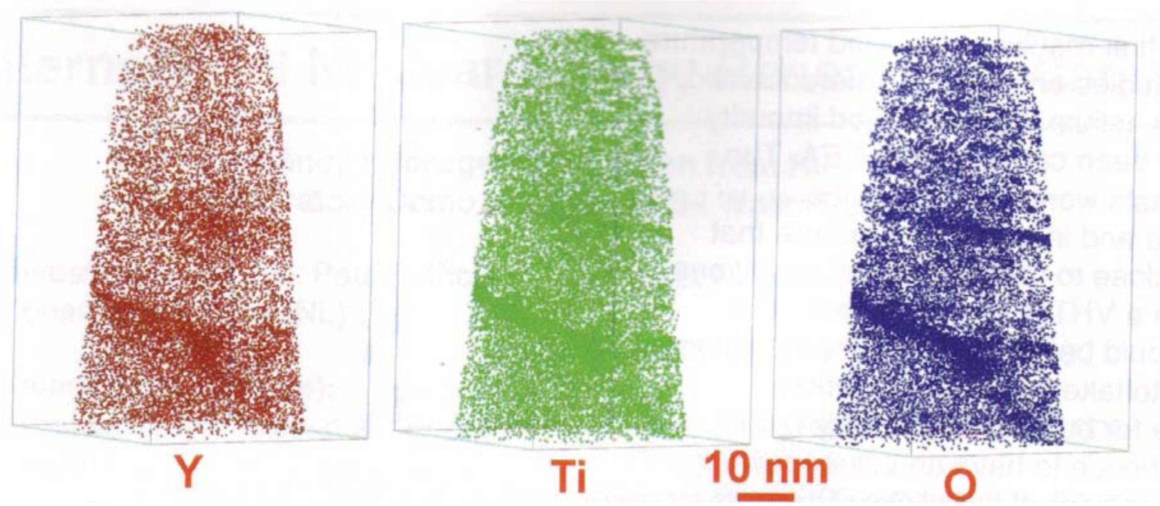


Figure C-11. The Y, Ti, and O Element Maps of Particles Observed in the OE14YWT-SM3 Alloy from the LEAP Analysis.

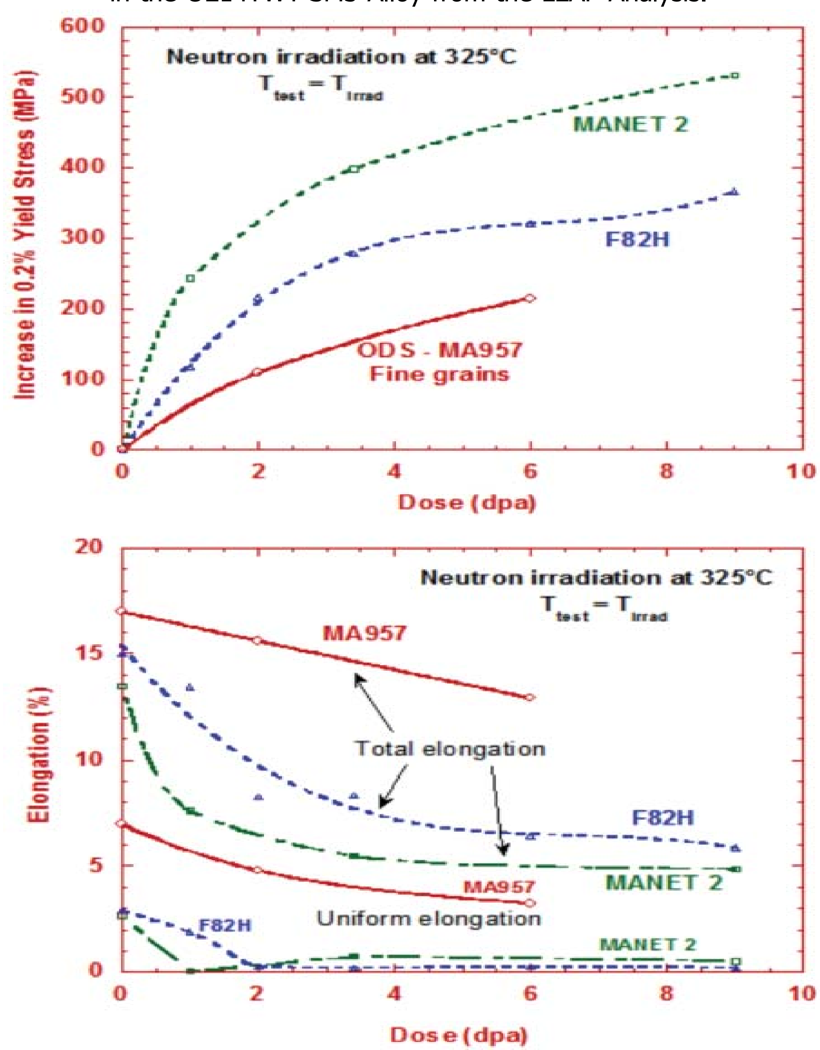


Figure C-12. Radiation-Induced Changes in Mechanical Properties for ODS AlloyMA957, Reduced Activation F82H, and Conventional MANET 2 Steels.
Yield Strength is shown in (A) and Uniform and Total Elongation in (B).

Finally, the first results of elevated temperature corrosion studies on the ODS reference alloys in air and in helium with controlled impurity levels have been carried out at CEA. The corrosion tests were performed at a temperature and in a gas atmosphere that should be close to the operational conditions expected in a VHTR. The temperature of ~950°C should be high enough for the oxide dispersion to take part in the reactions responsible for building the surface oxide scale, and hence to have an influence on the corrosion behavior of the alloys. The tests included three commercial ODS alloys: MA957, PM1000, and PM2000; the best reference ODS material, 12YWT; and the high-chromium conventional alloy Hastelloy X. Despite its low Cr content, the 12YWT alloy builds a chromia scale, which exhibits good properties. The corrosion resistance of 12YWT compares well with Hastelloy X, which is a good candidate material for the VHTR environment. This satisfactory short-term oxidation behavior at the low Po_2 prevailing in a VHTR is particularly promising since this nanocomposited steel could be appropriate for in-core structures.

Planned Activities

This project was scheduled to end in FY 2004; however, some continuing work will be carried out under a no-cost extension. In particular, specimens of the reference and developmental alloys that have been included in long-term irradiation experiments in the High Flux Isotope Reactor will be tested in the coming year. Because of the extended irradiation time and the reactor operating schedule, it was not possible to complete these irradiations within the three years of the original project. Further unirradiated mechanical testing on the developmental alloy 14YWT will also be carried out at ORNL, and additional microstructural characterization is planned by both ORNL and UCSB.

International Nuclear Energy Research Initiative

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

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

Principal Investigator (France): S. Goldstein, Commissariat a' l'Energie Atomique (CEA)

Collaborators: General Atomics (GA)

Project Number: 2002-001-F

Project Start Date: October 1, 2002

Project End Date: September 30, 2005

Research Objective

The objective of this I-NERI project is to perform a laboratory-scale demonstration of the sulfur-iodine (S-I) water-splitting cycle and to assess the potential of this cycle for application to nuclear hydrogen production. The project is designing, constructing, and testing the three major component reaction sections that make up the S-I cycle. The CEA is designing and testing the prime (Bunsen) reaction section, General Atomics is developing and testing the HI decomposition section, and Sandia National Laboratories is developing and testing the H_2SO_4 decomposition section. Activities for this period include initial program coordination and information exchange, the development of models and analyses that will support the design of the component sections, and preliminary designs for the component reaction sections. The sections are being designed to facilitate integration into a closed loop in a later stage.

Research Progress

Bunsen Section—Commissariat à l'Energie Atomique

The experimental studies on the Bunsen section of the S-I cycle are ongoing at the CEA. A laboratory-scale Bunsen reactor has been assembled and initial tests have been completed. To obtain partial pressure data on the vapor-liquid equilibrium of the $HI/I_2/H_2O$ system, a glass cell with sapphire and quartz windows has been designed, built, and assembled in a thermoregulated oven (Figure C-13). The analytical diagnostics for this system have been validated and calibrated with pure vapor components. Measurements on the binary systems HI/H_2O, I_2/H_2O, and I_2/HI have also been performed, as have preliminary total pressure measurements. Measurements on ternary systems are planned next.

HI Decomposition Section — General Atomics

Two alternative designs have been developed for the HI decomposition section. The first method involves the simultaneous concentration and decomposition of HI in a reactive distillation column. The second method for HI concentration and decomposition utilizes H_3PO_4 to first remove the I_2 from the HI_x stream (80:10:10 wt % for I_2:HI:H_2O), and subsequently distill pure HI prior to decomposition. This process is referred to as extractive distillation and is being developed as a backup process to the reactive distillation because reactive distillation has not been experimentally demonstrated.

In reactive distillation, the separation of HI from I_2 is combined with HI decomposition in one step. Roth and Knoche (1989) proposed a reactive distillation flow sheet that was adopted by GA as the baseline process. This year, the baseline was updated to an alternative flow sheet developed at CEA because, in this flow sheet, there is a significant reduction in the amount of HI recycled back to the Bunsen reactor.

The reactive distillation column apparatus (Figure C-14) is enclosed by an internally-insulated pressure vessel which maintains pressure around the glass column and ensures adiabatic conditions for distillation. Each section of the vessel has a sight glass port to allow viewing of critical sections of the glass column during operation. In addition, feed-throughs for process streams and wiring for instrumentation are incorporated into the flanges to allow for process control of the experiments. Experimental runs to verify the viability of the reactive distillation process will be conducted in FY 2005.

The design of the experimental apparatus for the laboratory demonstration of the extractive distillation is essentially complete. Gas-phase decomposition has been selected for the current glass-system demonstration, utilizing a membrane separator for the product recovery due to pressure limits. Should extractive distillation be selected for the metal-system demonstration, liquid-phase decomposition will be used, as the metal system will be operated at the nominal pressures and temperatures proposed for a production plant.

Sulfuric Acid Decomposition—Sandia National Laboratories

A sulfuric acid processing unit that concentrates, boils, superheats, and decomposes the acid has been designed and constructed. Preliminary shakeout tests with water have been successfully performed to demonstrate the integrity of the system. The design is highly modular and allows for replacing individual heating, condensing, and reacting unit operations without disrupting other units. The apparatus is constructed of alloys that could potentially be used in a pilot plant.

The apparatus consists of two sections that are contained in adjacent fume hoods, as shown on the left and right sides of Figure C-15.

The heating section, in the left fume hood, consists of three basic units: a boiler, a superheater, and a catalytic decomposer. Concentrated liquid acid is pumped into the top of the boiler where the acid vaporizes as it falls downward through a column of acid-resistant ceramic pellets. The acid vapors are then heated to 650°C in the superheater. At this temperature the vapor decomposes to SO_3 and H_2O. This stream then passes through a catalyst-packed decomposer where the SO_3 decomposes to SO_2 and O_2. Several catalysts will be tested, but the initial catalyst is 1 wt% platinum on zirconia pellets. In the cooling section, shown in the right fume hood in Figure C-15, acid vapor, SO_2, and O_2 pass through a coiled Incoloy 880H tube that is inside a chilled-water bath. The electrical conductivity of the collected liquid is measured as it flows through the conductivity cell. From this measurement, the unreacted acid concentration is determined. The oxygen concentration in the gas stream is measured as the gases flow past an oxygen probe. Both measurements are performed in real time.

The instruments described above cannot operate at the high temperatures of the experiment. However, an optical measurement such as Fourier Transform Infrared (FTIR) spectroscopy has the potential to provide in situ H_2O, SO_2, and SO_3 concentration data. Therefore, an optical cell to perform such in situ measurements is being developed. The only limiting constraint is finding IR-transparent windows that are resistant to acid corrosion at high temperatures. Researchers are therefore developing windows with very thin gold-coatings that should be IR-transparent and acid resistant.

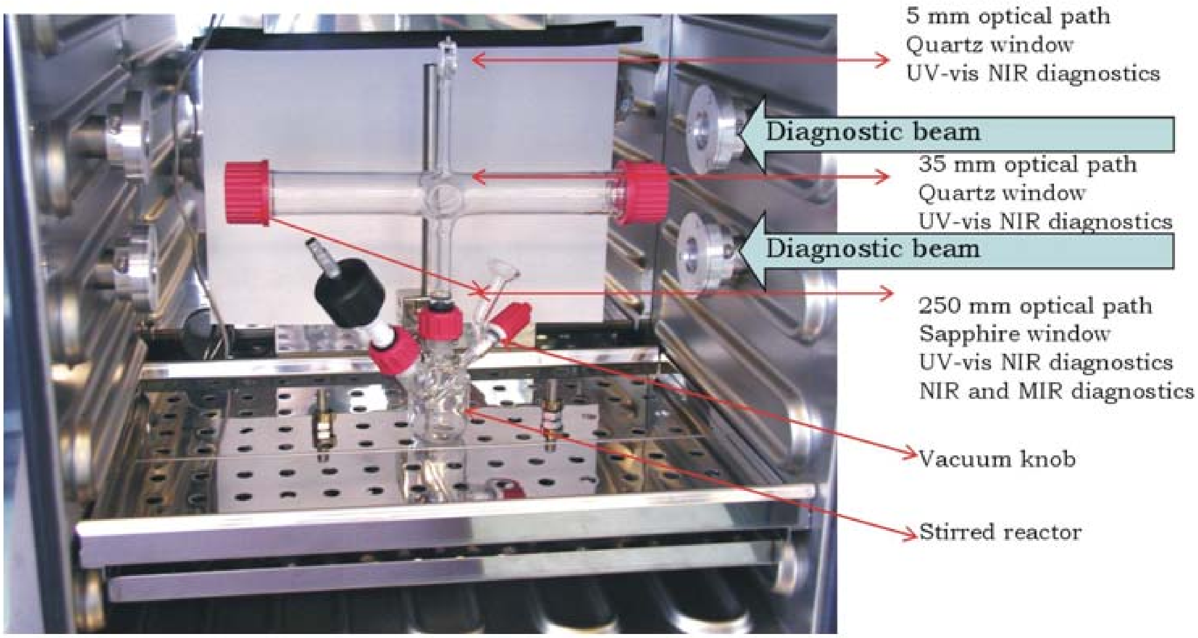


Figure C-13. Apparatus for I_2 Measurements of the Partial Pressures of HI, I_2, and H_2O.

Figure C-14. Pressure Vessel for Containment of Reactive Distillation Experiment.

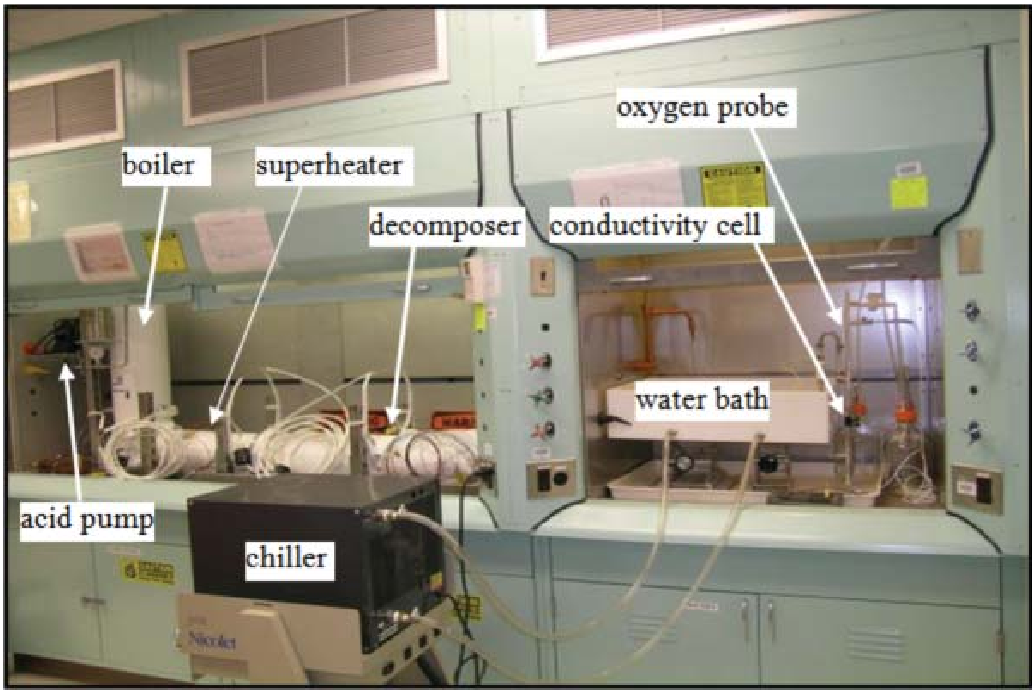


Figure C-15. Sulfuric Acid Heating and Cooling Sections
in the Left and Right Fume Hoods, Respectively.

International Nuclear Energy Research Initiative

Hydrogen Process High Temperature Heat Source Coupling Technology

Principal Investigator (U.S.): C. Park, Idaho National Laboratory (INL)

Principal Investigator (France): P. Billot, Commissariat a' l'Energie Atomique (CEA)-Saclay

Project Number: 2004-001-F

Project Start Date: August 1, 2004

Project End Date: August 31, 2007

Project Abstract

The objective of this project is to develop the technology necessary to couple a high temperature heat source to hydrogen production processes, including both the S-I cycle and the high-temperature electrolysis (HTE) process.

Tasks

1. Evaluate heat transmission and exchangers, including technical and industrial feasibility, flexibility of coupling schemes, and conversion energy losses for the design solutions.
2. Analyze design solution schemes to consider optimized technological options, including component connections for these interfaces: reactor/intermediate heat exchanger/high temperature step/medium temperature step/low temperature step.
3. Develop and utilize an engineering tool to evaluate the complete heat balances for the different schemes, optimizing the use of energy (electricity and/or heat) in relation to hydrogen production. This simulation tool will permit an understanding of the process behavior during normal operation, transient, and accidental condition. The information from the models and simulations will provide the data needed to perform availability and safety analysis, as well as information and interface data for the economic evaluation and cost estimation work teams.

International Nuclear Energy Research Initiative

OSMOSE - An Experimental Program for Improving Neutronic Predictions of Advanced Nuclear Fuels

Principal Investigator (U.S.): R. Klann, Argonne National Laboratory (ANL)

Principal Investigator (France): J. Hudelot, Commissariat a` l'Energie Atomique (CEA)-Cadarache

Collaborator: University of Michigan (UM)

Project Number: 2004-002-F

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

The objective of this collaborative program with the French CEA is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs and to provide the experimental data for improving the basic nuclear data files. These data will support advanced reactors designed for transmutation of waste or plutonium (Pu) burning, sub-critical systems such as found in advanced accelerator applications, and the waste disposal and treatment program in the area of criticality safety. This program is very generic in the sense that it will measure these reaction rates over a broad range of isotopes and spectra, and will be used to provide guidance to all nuclear data programs in the world.

The design of nuclear systems has shifted over the years from a "test and build" approach to a much more analytical methodology based on the many advances in computational techniques and nuclear data. To a large extent, current reactors can be calculated almost as well as they can be measured. This is due in particular to the high quality nuclear data available for the few major isotopes which dominate the neutronics of these systems. Nevertheless, most future nuclear systems concepts and advanced fuels development programs currently under way use significant quantities of minor actinides to address modern day issues such as proliferation resistance and low cost. For example, proliferation resistant reactors and fuels are typically based on ^{232}Th and ^{233}U. High burnup fuels contain large quantities of Americium and Curium. Systems designed for Plutonium and Minor Actinide burning are very sensitive to uncertainties in Americium and Curium data. There are also several other programs where the minor actinide data are essential. These include the Accelerator Transmutation of Waste concepts and Burnup Credit programs.

The need for better nuclear data has been stressed by various organizations throughout the world and results of studies have been published which demonstrate that current data are inadequate for designing the projects under consideration. In particular, a Working Party of the OECD has been concerned with identifying these needs and has produced a detailed High Priority Request List for Nuclear Data. The first step in obtaining better nuclear data consists of measuring accurate integral data and comparing it to integrated energy dependent data. This comparison provides a direct assessment of the effect of deficiencies in the differential data. Several U.S. and international programs have indicated a strong desire to obtain accurate integral reaction rate data for improving the major and minor actinide cross sections. Specifically, these include: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm. Data on the major actinides (i.e., ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) are reasonably well known and available in the Evaluated Nuclear Data Files (JEF, Japanese Evaluated Nuclear Data Library [JENDL], ENDF-B). However, information on the minor actinides (i.e., ^{232}Th, ^{233}U, ^{237}Np, ^{238}Pu, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm) is less well known and considered to be relatively poor in some cases, having to rely on models and extrapolation of few data points. This is mainly due to the difficulty of

obtaining relatively pure samples of sufficient quantity (up to about one gram) to perform reliable reaction rate measurements.

The CEA has also recognized the need for better data and has launched an ambitious program aimed at measuring the integral absorption rate parameters in an experimental facility located at the Cadarache Research Center. A complete analytical program is associated with the experimental program and aims at understanding and resolving potential discrepancies between calculated and measured values. The final objective of the program is to reduce the uncertainties in predictive capabilities to a level acceptable to core designers and government regulators.

ANL has expertise in these areas. In the past, ANL teams have developed very accurate experimental techniques which will greatly enhance the content of the experimental program. Furthermore, current ANL staff have participated in the development of the French experimental and analytical program, and have contributed to the computational tools used by the French teams.

CEA is interested in collaborating with ANL in the experimental design, measurement, and analysis tasks. In addition, CEA is interested in getting specific minor actinide isotopes from DOE that are not easily obtainable in France. In exchange for ANL's expertise and DOE's isotope supply, CEA will make all facilities and experimental results available at no charge. While the exact cost of the French program is not available, it is estimated to be at least U.S. $2 million per year. Thus, ANL's participation in the French program at a modest level ($300K per year) and DOE's one-time supply of sample materials would very strongly leverage the funds available in the U.S. for preparing the technical basis for future nuclear systems and advanced fuels development. In addition, other programs, such as criticality safety, will benefit from this information.

In this program, ANL staff will participate in the experimental measurements made in the MINERVE reactor at Cadarache, and all the data will be available to the U.S. in exchange for this participation.

This project has three critical outcomes:

1. High quality experimental data representative of the major and minor actinides will be made available to the U.S. programs.
2. The U.S. neutronics and criticality safety codes will be validated for reactivity effects from the major and minor actinides.
3. Potential deficiencies in U.S. nuclear data and analysis tools will be identified.

In addition to the critical outcomes from the project, there are general benefits to the nuclear program in the U.S. Specifically, by cross-calibration with measurements in different neutron spectra, the uncertainties in the data will be better known and deficiencies in the cross-section data will become apparent. This will lead to a better understanding of the available cross-section data and of which areas need further development and research. This project also intends to involve a graduate student in the measurement and analysis tasks. By introducing young experimentalists to the project through key involvement in tasks, expertise is developed within the U.S. This is vital since there are very few remaining experimentalists in this area.

International Nuclear Energy Research Initiative

Thermal-hydraulic Analyses and Experiments for GCR Safety

Principal Investigator (U.S.): D. McEligot, Idaho National Laboratory (INL)

Principal Investigator (France): D. Tenchine, Commissariat a' l'Energie Atomique (CEA)

Collaborators: Argonne National Laboratory (ANL), Iowa State University

Project Number: 2004-003-F

Project Start Date: January 1, 2005

Project End Date: January 31, 2008

Project Abstract

The objective of this collaborative experimental and computational research is to provide benchmark data for the assessment and improvement of thermal-hydraulic codes proposed for evaluating decay heat removal concepts and designs in the Gas-Cooled Reactor (GCR) programs of the international Generation IV Initiative. These reactors feature complex geometries and a wide range of temperatures, leading to significant variations of the gas thermodynamic and transport properties, plus effects of buoyancy during loss-of-flow and loss-of-coolant scenarios, and during reduced power operations. The complex geometries proposed have included non-circular fuel channels, high temperature exit regions, inlet regions for heavy gas injection, plenum regions, decay heat removal heat exchangers, regenerative heat exchangers, intermediate heat exchangers, and reactor cavities with cooling panels.

Existing system safety codes provide reasonable predictions for high Reynolds-number flows, but their correlations can give misleading results for low Reynolds-number gas flows with buoyancy, as in accident scenarios, even with simple circular tubes. Conceptually, CFD codes with turbulence models can yield predictions for improvement of correlations and preliminary design. However, recent assessments have shown that most turbulence models used in general purpose codes give unreliable, optimistic predictions for these cases. To avoid this problem and to improve predictive capabilities, further benchmark data are needed for complex geometries. These bases can be obtained from direct numerical simulations (DNS) or large eddy simulations (LES), after validation with measurements or from experiments.

Under the thermal-hydraulic experiments task for the GFR systems design and integration, INL is currently evaluating needs for thermal-hydraulics experiments in support of predictions for GFR decay heat removal schemes. INL is also extending the Relap/Athena codes to treat flows in GCRs and SCWRs. Recent I-NERI and NERI projects have developed LES and DNS codes for low Reynolds-number, strongly-heated, buoyant gas flows in circular tubes to serve as benchmarks in those situations. INL has employed its unique MIR flow system for velocity/turbulence data in scaled fuel channels for a VHTGR concept. CEA is extending the CATHARE code to treat flows in GCR circuits. For GCR thermal-hydraulic studies, CEA is improving CFD codes such as TRIO_U and CAST3M. CEA INL/PRO-04-01848 is also developing global CFD models to describe GCR systems, including reactor core, reactor vessel, reactor pit, and containment, allowing the simulation of nominal and accidental regimes with coupled thermal-hydraulics, neutronics, and heat transfer including radiation. Under Generation IV Modeling improvement, ANL is currently developing global CFD models of a VHTR concept to identify key fluid flow and heat transfer phenomena (in-vessel and reactor cavity cool-down system [RCCS]) for limiting accident scenarios during passive shutdown heat removal. The studies also assess the analytical capabilities available in the commercial CFD code Star-CD for this application and potential applicability to the VHTR of the ANL Natural Convection Shutdown Heat Removal Test Facility (NSTF) for future VHTR experimental efforts. NSTF is a full-scale simulation facility of a Reactor Vessel Auxilliary Cooling System (RVACS) which is

similar to the RCCS system of VHTR. Additional ANL efforts are examining turbulence modeling options for a wide variety of systems.

Outcomes of this proposed research will be (1) validated LES and Reynolds-averaged Navier-Stokes (RANS) techniques for gas flows with property variation and buoyancy effects through complex geometries important in GCR development, (2) computational and experimental benchmark data for assessing existing and future CFD codes, (3) improved quantitative understanding of the limitations of current and proposed system safety codes, and (4) user-friendly LES and RANS codes for these geometries.

International Nuclear Energy Research Initiative

SiC/SiC for Control Rod Structures for Next Generation Nuclear Plants

Principal Investigator (U.S.): W. Windes, Idaho National Laboratory (INL)

Principal Investigator (France): P. Billot, University of Bordeaux

Collaborator: Pacific Northwest National Laboratory (PNNL)

Project Number: 2004-004-F

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

This project falls under Generation IV materials R&D. The proposed research will develop tubular geometry SiC/SiC composite material for control rod structures with equal or better mechanical, thermal, and radiation damage-resistant properties compared to present flat plate SiC/SiC composites. Material synthesis methods will be developed to optimize properties of SiC/SiC based on Nicalon Type-S fibers, chemical vapor infiltrated silicon carbide (CVI-SiC) matrix, and either pyrocarbon or multilayered C/SiC interfaces. This work will generate a property database for the optimized materials using standardized American Society for Testing and Materials (ASTM) test methods for strength, toughness, and thermal conductivity. Tests will be conducted on tubes of SiC/SiC prepared in this program and compared to flat plate materials made by current state-of-the-art methods. The project will deliver a state-of-the-art tubular SiC/SiC composite material together with a synthesis method for the same.

International Nuclear Energy Research Initiative

Assessment of Existing Physics Experiments Relevant to VHTR Designs

Principal Investigator (U.S.): T. Taiwo, Argonne National Laboratory (ANL)

Principal Investigator (France): R. Jacqmin, Commissariat a` l'Energie Atomique (CEA)

Collaborators: Idaho National Laboratory (INL)

Project Number: 2004-005-F

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

A program element of the Generation IV Design and Evaluation Methods area is aimed at identifying system design and safety analysis needs and advancing analysis capabilities to meet these needs. Important activities that have been identified include the qualification of analysis methodologies and tools, and the identification of experiments and benchmark tests that could be used for this purpose. Additionally, it was recommended to leverage efforts through international collaborations and participation in ongoing international activities.

This project will identify and assess experimental data and benchmark tests applicable to the qualification of Generation IV system design and analysis physics tools. During FY 2005, the team will interface with U.S. and international groups to identify and assess existing data that could be used for the qualification and quality assurance of computer codes and databases for reactor physics analysis of the VHTR. This activity is expected to support subsequent efforts (in FY 2006 and beyond) to document the benchmark specifications and measured results in a standard format for use in VHTR software quality assurance efforts. That effort will include:

- Evaluation of the adequacy of existing critical experiments and nuclear data
- Definition of target accuracies for pertinent core parameters
- Sensitivity studies for assessing relevance of experiments to VHTR
- Identification of additional integral experiments and/or nuclear data evaluation and measurements that are required
- Joint detailed analysis of selected physics experiment(s).

International Nuclear Energy Research Initiative

GFR Physics Experiments in the CEA-Cadarache MASURCA Facility

Principal Investigator (U.S.): T. Taiwo, Argonne National Laboratory (ANL)

Principal Investigator (France): R. Jacqmin, Commissariat a' l'Energie Atomique (CEA)

Project Number: 2004-006-F

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

GFR designs are being developed to meet Generation IV goals of sustainability, economics, safety and reliability, and proliferation resistance and physical protection as part of an I-NERI project led by the CEA-Cadarache nuclear center in France and ANL in the U.S. The U.S. and French organizations are developing GFR concepts employing different coolants and fuel assembly geometries. The GFR designs include block-type, pin-type, pebble-bed, and dual-particle cores using helium or CO_2 as coolant. In addition to the design studies, planning is underway at CEA-Cadarache for experiments that will investigate the core physics issues relevant to Generation IV GFR designs that were not addressed in previous Gas-Cooled Reactor experiments. This effort, designated Experimental Neutron Investigation on Gas-reactors at MASURCA (ENIGMA), has the objectives of defining MASURCA configurations that are similar in their neutronic characteristics to the candidate GFR designs and extending the validation domain of the neutronics tools to design and licensing calculations or future GFRs.

ANL has participated in the initial planning phase for ENIGMA by evaluating the "representativity" of proposed experiments (i.e., their neutronic similarity to corresponding design concepts). The objective of the current proposal is to extend this collaboration by having ANL personnel participate in the ENIGMA project and evaluate experimental results with the intention of improving analytical models for GFRs. The scope of work for future collaborations include:

1. ANL will contribute personnel to the experimental teams for Phases 1 and 2. Phase 1 activities will be conducted in 2005-2006, and will include core characterization measurements, central substitution worths, and reflector studies for representative configurations of the demonstration plant–the Experimental and Technology Demonstration Reactor (ETDR). The ETDR is a 20-50 MWth demonstration reactor to be built in Cadarache around 2010. Phase 2 activities will study the ETDR mock-up.
2. Perform neutronic sensitivity studies to support the definition and justification of the experimental program. Contribute to the definition and design of experiments and selection of integral parameters that will be measured to characterize the GFR systems.
3. Analysis of experimental results with the goal of improving reactor analysis tools for GFR design.

International Nuclear Energy Research Initiative

Evaluation of Materials for Gas-Cooled Fast Reactors

Principal Investigator (U.S.): K. Weaver, Idaho National Laboratory (INL)

Principal Investigator (France): J. Seran, Commissariat a' l'Energie Atomique (CEA)

Collaborators: University of Wisconsin, University of Michigan (UM), Argonne National Laboratory (ANL), Pacific Northwest National Laboratory (PNNL)

Project Number: 2004-007-F

Project Start Date: August 1, 2004

Project End Date: August 31, 2007

Project Abstract

Both France and the United States have a shared interest in the development of advanced reactor systems that employ inert gas as a coolant. Currently, insufficient physical property data exist to qualify candidate materials for GFR designs. The goal of this project is to establish candidate metallic and ceramic materials for GFR designs and to evaluate their mechanical properties, dimensional stability, and corrosion resistance.

The first goal of this project is to improve high-temperature creep strength and resistance to environmental attack by optimizing grain boundary structural orientations and alloy compositions. Thermal-mechanical treatment will be performed on GFR candidate alloys to maximize the fraction of low-energy boundaries. Additionally, minor alloy modifications will be attempted. Following treatment, the changes to microstructure will be characterized. For optimized alloys, creep testing in controlled purity gas environments like He, He+O, and simulated High-Temperature Gas-Cooled Reactor (HTGR), with complex impurities such as H_2, H_2O, CO, CO_2, and CH_4, will be performed.

The second goal of the project is to characterize radiation resistance of candidate GFR metallic materials. Candidate metallic materials for the GFR have not typically been used for high-dose core components. Therefore, radiation response of these alloys will be characterized by examining the changes in microstructure in samples irradiated with high-energy ions and, when available, neutrons from a test reactor.

The third goal of the project is to start studies on the environmental compatibility of ceramic materials in controlled purity gas environments (e.g., He, He+O, and simulated HTGR with complex impurities).

International Nuclear Energy Research Initiative

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

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

Principal Investigator (France): J. Rouault, Commissariat a` l'Energie Atomique (CEA)

Collaborators: General Atomics (GA), Brookhaven National Laboratory (BNL), Idaho National Laboratory (INL), Massachusetts Institute of Technology (MIT)

Project Number: 2004-008-F

Project Start Date: October 1, 2004

Project End Date: September 31, 2007

Project Abstract

The objective of this Generation IV project is to continue the France-U.S. effort to develop and design High-Temperature GFR/Hardened Spectrum Reactors with a high degree of safety and an integrated fuel cycle. It will produce the report on GFR Preliminary Viability defined by the Generation IV GFR R&D Plan. CEA-Cadarache and Argonne National Laboratory, together with partners from both French and U.S. industry, sister national laboratories, and academia, are currently collaborating on the ongoing I-NERI project FY 2001–2002, "Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum." The cooperative project has reached the stage where it is being integrated into the Generation IV/GIF effort. This I-NERI project is in accordance with the International Collaboration Plan drafted by the provisional GFR System Design and Safety Management Board at the March 2-3, 2004 meeting to respond to the Generation IV Nuclear Energy Systems GFR R&D Plan.

The effort is being carried out in coordination with the current ongoing U.S./France I-NERI GFR development project between CEA-Cadarache and ANL which will reach the completion phase in February 2005. The following are the main collaborative tasks:

1. GFR Exploratory Studies - Core Design (calendar year [CY] 2005)
2. GFR Exploratory Studies-System Designs (CY 2005)
3. GFR Safety Studies (CY 2005-CY 2007)
4. GFR Pre-Conceptual Design (CY 2006-CY 2007)

2001-002-F

5. GFR Physics Experiments in MASURCA (CY 2005-CY 2007)
6. Code System benchmarking (CY 2006-CY 2007)
7. GFR/EDTR Mission Report (CY 2005-CY 2007)

Project Organization

ANL Tasks: 1-7

GA Tasks: 2, 3, 4

BNL Tasks: 2, 3, 5

MIT Tasks: 2, 3, 5

INL Tasks: 4, 6, 7

CEA Tasks: 1-7

International Nuclear Energy Research Initiative

Development of Fuels for the Gas-Cooled Fast Reactor

Principal Investigator (U.S.): M. Meyer, Idaho National Laboratory (INL)

Principal Investigator (France): N. Chauvin, Commissariat a' l'Energie Atomique (CEA)-Cadarache

Collaborators: Joint Research Center Institute for Transuranium Elements (ITE), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL)

Project Number: 2004-009-F

Project Start Date: October 1, 2004

Project End Date: September 30, 2007

Project Abstract

GFR fuel-operating parameters and physical requirements are outside of the current experimental nuclear fuel database. Many basic viability issues will need to be experimentally addressed to demonstrate the feasibility of proposed GFR fuels.

Two basic fuel types appear to be viable for GFR service: refractory matrix dispersions and refractory metal or ceramic clad pin-type fuels. This project seeks to develop fuels of these types suitable for GFR service and demonstrate feasibility of these fuels through analysis of fuel requirements, simulation of fuel behavior using fuel performance models, fabrication of fuel specimens, characterization of microstructure and properties, and scoping fuel irradiation testing. Ion irradiation testing of materials will be conducted to simulate material behavior at high irradiation doses in short times. The GFR-F1 test in ATR (and the FUTURIX-MI test in Phénix) also addresses basic issues regarding the irradiation behavior of the 'exotic' refractory materials required for GFR fuel service in a neutron-only environment. Ultimately, proof-of-concept for GFR fuel can only be demonstrated through irradiation testing of fissile-bearing specimens. The GFR-F2 scoping fuel irradiation test in the ATR is planned as an integral fuel behavior test that will give the first true indication of fuel feasibility.

The proposed work will be conducted within the GFR Fuel Project (i.e., with participation from GIF partners France, UK, Japan, and South Korea). Data collected as a result of this work will be leveraged to the extent possible to provide data relevant to small modular reactor and LWR inert matrix fuel development efforts.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): K. Weaver, Idaho National Laboratory (INL)

Principal Investigator (France): N. Devictor and J. Rouault, Commissariat a' l'Energie Atomique (CEA)

Collaborator: Massachusetts Institute of Technology (MIT)

Project Number: 2004-010-F

Project Start Date: August 1, 2004

Project End Date: August 31, 2007

Project Abstract

GFRs are contenders of international interest for advanced nuclear power service. It is well recognized, however, that particular attention must be paid to reliable decay heat removal if GFRs are to meet the high expectations for safety assurance established for new reactor designs. Probabilistic Risk Assessment (PRA) has matured over the last thirty years and is expected to play a key role in all aspects of system design and safety. The use of PRA will allow the designers to take advantage of lessons learned from the vast array of PRA applications to LWRs and other reactor types.

There are two major issues that need to be addressed in order to take full advantage of PRA capabilities. First, since currently-operating LWRs do not employ passive systems, PRA models for such systems will have to be developed for advanced reactors. Second, the use of PRA in design implies that there are probabilistic goals that can be used to determine which design is "good enough." Although there are activities by the U.S. Nuclear Regulatory Commission and the International Atomic Energy Agency to establish such probabilistic goals, the current licensing framework is largely "deterministic" and is not expected to change substantially in the near future. This state of affairs raises the issue of whether the design should satisfy the deterministic criteria or the probabilistic goals, especially when a particular design option meets the probabilistic goals but fails the deterministic criteria.

While the proposed project will address both of the issues mentioned above, its focus will be the formulation of the conceptual design of GFR decay heat removal systems that are effective in normal modes of operation as well as normal shutdown, refueling, and, especially, post-loss-of-coolant accident (LOCA) modes, under a range of scenarios including ATWS and station blackout. We will explore the possibility of having safety systems that would be initially active in order to promote economic feasibility. These active systems would be followed by passive, reliable systems that would take over after a pre-determined time for continued core cooling.

International Nuclear Energy Research Initiative

Thermochemical Hydrogen Production Process Analysis

Principal Investigator (U.S.): M. Lewis, Argonne National Laboratory (ANL)

Principal Investigator (France): J. Borgard and S. Goldstein, Commissariat a' l'Energie Atomique (CEA)

Project Number: 2004-011-F

Project Start Date: October 1, 2004

Project End Date: September 31, 2007

Project Abstract

ANL has recently initiated a U.S. DOE-NE project to develop and apply a methodology for a consistent assessment of thermochemical hydrogen production cycle efficiencies. Published cycle efficiencies suffer from inconsistent evaluation methodologies and, potentially, unreliable thermodynamic data. For example, published efficiencies for the most well known sulfur-iodine cycle vary from 28% to greater than 55%. These differences can be attributed to various factors, such as changing the calculational method from essentially hand calculations in the 1970s to sophisticated computerized simulation programs currently available, improvements in process design and technologies, better heat matching, and different peak temperatures.

Researchers have not always defined their efficiency estimates in terms of high or low heating value, or closed or open loop. The effect of cogeneration should be explicitly defined for comparison purposes. For the UT-3 calcium-bromine cycle, efficiencies are estimated to increase by 10% when cogeneration is included. Similar increases are predicted for the sulfur-iodine cycle. Cogeneration has a huge impact on efficiency. Different thermodynamic databases have been used in the past and reasonable, but not necessarily transparent, estimates have been used when data are not available. These differences make it difficult to determine the best efficiency estimate for the well-studied sulfur-iodine cycle and almost impossible to fairly judge the merits of other potentially promising hydrogen production methods. The DOE Nuclear Hydrogen R&D Plan recognizes that a standardized process assessment methodology is needed as the program selects the most promising hydrogen production technologies to pursue.

CEA, in Saclay, France has been collaborating with the U.S. on the development of sulfur-iodine thermochemical hydrogen production processes. Several papers concerning calculating the upper bound for the efficiency of the sulfur-iodine cycle have been published recently. These papers have considered the inadequacy of the thermodynamic data for the H_2O-HI-I_2-H_2 reactions and how these data impact overall efficiency.

Appendix D

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

International Nuclear Energy Research Initiative

Project No.	Title
2002-008-K	Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in Advanced Light Water Reactors (ALWRs)
2002-010-K	The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena
2002-016-K	Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Super-critical Reactors
2002-020-K	Development of Enhanced Reactor Operation Strategy Through Improved Sensing and Control at Nuclear Power Plants
2002-021-K	Condition Monitoring Through Advanced Sensor and Computational Technology
2002-022-K	In-Vessel Retention (I&II)
2003-002-K	Passive Safety Optimization in Liquid Sodium-Cooled Reactors
2003-008-K	Developing and Evaluating Candidate Materials for Generation IV Super-Critical Water-Cooled Reactors
2003-013-K	Development of Safety Analysis Codes and Experimental Validation for a Very-High-Temperature Gas-Cooled Reactor
2003-020-K	Advanced Corrosion-Resistant Zirconium Alloys for High Burnup and Generation IV Applications
2003-024-K	Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte
2004-001-K	Screening of Gas-Cooled Reactor Thermal-Hydraulic and Safety Analysis Tools and Experiment Database
2004-002-K	Investigation of Heat Transfer in Super-Critical Fluids for Application to the Generation IV Super-Critical Water-Cooled Reactor (SCWR)
2004-003-K	Development of Advanced Suite of Deterministic Codes for VHTR Physics Analysis
2004-004-K	Development of Voloxidation Process for Treatment of LWR Spent Fuel
2004-005-K	Development and Test of Cladding Materials for Lead-Alloy Cooled Transmutation Reactors
2004-006-K	Alternative Methods for Treatment of TRISO Fuels

U.S./Republic of Korea Collaborators

U.S. National Laboraties

Argonne
Idaho
Oak Ridge

U.S. Universities

University of California, Santa Barbara
Univerisity of Illinois, Chicago
Iowa State University
University of Maryland
University of Michigan
Ohio State University
Pennsylvania State University
Purdue University
University of Wisconsin

U.S. Industry

Westinghouse Electric

International

Cheju University
Chosun University
Chungnam National University
Hanyang University
Korea Advanced Institute of Science and Technology (KAIST)
Korea Atomic Energy Research Institute (KAERI)
Korea Hydro and Nuclear Power Company
Korean Electric Power Research Institute
Republic of Korea (ROK) Maritime University
Seoul National University
Tusan National University

International Nuclear Energy Research Initiative

Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in Advanced Light Water Reactors (ALWRs)

Principal Investigator (U.S.): M. Corradini, University of Wisconsin (UW)

Principal Investigator (Korea): K. Bang, Republic of Korea Maritime University (KMU)

Collaborator: Argonne National Laboratory

Project Number: 2002-008-K

Project Start Date: January 1, 2002

Project End Date: December 31, 2004

Research Objectives

Our objectives for this research are to:

Task I – Measure the cool-down behavior of the melt-water mixing zone by thermal mapping of this multiphase, multi-component system (ANL lead).

Task II – Measure the flow regime and interfacial area behavior of the melt-water multiphase, multi-component mixture by the use of innovative real-time x-ray imaging (UW lead).

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

Task IV – Integrate the various models for long-term debris coolability into an overall approach.

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

Research Progress

Tasks I and II: Transient Thermal Mapping of the Mixing Zone and X-ray Imaging (ANL-UW)

In the design of the next generation of nuclear reactors and in the safety assessment of currently operating nuclear power plants, it is necessary to evaluate the risk from a severe accident and to identify the key strategies to follow in order to mitigate possible consequences. In the unlikely event of a severe accident involving core melt, it is important to identify the processes that would allow the molten core material to cool down and resolidify, to reliably remove core decay heat, and to bring core debris to a safe and stable state, thereby achieving "core coolability."

For current nuclear plants, the safety approach taken by plant operators and the NRC is to:

- Provide alternative sources of water to arrest progression of the degraded core accident
- Develop accident management procedures to maximize reactor pressure vessel integrity
- Provide long-term ultimate heat sink paths to remove decay heat from containment
- Delay any potential containment failure beyond 24 hours for off-site emergency actions.

For next-generation plants seeking final design certification from the NRC (e.g., AP1000), the approach has been to improve the reliability of each of these actions to reduce the probability of progressing further into a severe accident with required off-site emergency preparedness actions. Nevertheless, safety analyses of next-generation plants indicate that all four types of actions will be necessary to minimize off-site radiological dose. For example, NRC03-202 indicates that in-vessel retention is not assured, which would eventually cause containment basemat melt-through and off-site radiological consequences requiring a range of emergency actions (e.g., evacuation).

In contrast to this safety approach, next-generation nuclear plant safety in some developed countries (e.g., Europe) and developing countries (e.g., Korea) have taken a more proactive posture and have a goal to eliminate off-site radiological consequences and the need for off-site emergency actions. For example, the planned nuclear plant for Finland, Framatome European Pressurized Water Reactor (EPR), will incorporate an ex-vessel coolability system with the objective to preclude containment failure and the need for substantial off-site emergency action plans.

If the next-generation nuclear plants developed by U.S. nuclear vendors (e.g., AP1000, Economic and Simplified Boiling Water Reactor [ESBWR]) are to be competitive worldwide, the ability to preclude containment failure and off-site emergency actions will need to be incorporated into their future reactor designs. The OECD, in collaboration with the NRC and other reactor safety agencies worldwide, is creating a program to develop a technical basis for ex-vessel coolability concepts independent of any specific reactor design.

We would propose that the Department of Energy collaborate with the NRC in developing and participating in this program in 2005.

Current Technical Basis on Ex-Vessel Coolability

In a severe accident (i.e., "beyond design basis"), core-melt material has the potential to be discharged from the core region and eventually fail the reactor pressure vessel lower plenum wall. For example, analyses in NRC03-202 indicated the likelihood of reactor vessel failure given a core-melt relocating to the lower plenum was over 1-in-3. Given this scenario, the molten core material pours from the reactor pressure vessel, and accumulates as a molten pool in the reactor cavity below. This molten material, usually called corium, is composed of mainly uranium-oxide and zirconium-oxide, as well as limited quantities of zirconium and stainless steel from the melting of the reactor core internals and lower plenum. The molten corium can thermally attack the concrete underneath and decompose it, producing gases which agitate the pool, enhancing heat release to the boundaries as fission product decay heat and chemical reactions continue to add mass and energy to containment. It is the production of the gases that pressurize the containment and the attendant erosion can melt through the containment basemat.

To achieve "core coolability" in the reactor containment cavity and eliminate any threats to containment integrity by overpressurization or melt-through, a number of approaches have been proposed (e.g., water flooding from above or injection of water from below). The effectiveness of these techniques to achieve ultimate coolability involves the mixing of high-temperature melt with water, via boiling processes and/or injection.

Because this process occurs at large scales and with materials whose physical properties are not well determined, the phenomenology involved is not completely understood. In addition, many of the current, most widely used models were not specifically developed to simulate this phenomenology and do not always predict the experimental observations. Various attempts have been made to reproduce the problem experimentally by using either prototypic or simulant materials. Some of these are integral experiments that try to reproduce the entire scenario to pinpoint all the processes involved (e.g., EPRI/DOE/NRC supported melt attack and coolability [MACE] tests), while others are separate effect studies focused on the more detailed analysis of very specific phenomena (e.g., OECD-supported MCCI program).

A concept of core-melt quenching and long-term coolability by the bottom injection of water into the melt was originally proposed for application to the EPR. A number of tests employing simulant melts (coolability of melt by water injection experiment [COMET]) were conducted to demonstrate the viability of this concept. The concept appears promising for the development of innovative safety technologies for next-generation LWRs. However, fundamental data on the transport phenomena involved are needed for further refinement of the concept and, potentially, for regulatory acceptance and industry implementation.

As part of the DOE I-NERI project with Korea, Argonne National Laboratory and the University of Wisconsin-Madison have been conducting experiments to provide fundamental data on the COMET concept. These experiments are focusing on the effects of two important variables for which no data is currently available, namely the ambient pressure and the presence of non-condensable gas co-injected from below with the water.

Ex-Vessel Reactor Cavity Flooding Concepts

In the absence of a dedicated device for core-melt retention and heat removal (i.e., core catcher) the only option for achieving ex-vessel coolability is to add water to the reactor cavity so as to remove the decay heat and enthalpy of the corium by vaporizing the water mixed with the corium melt. Three different concepts for water addition have been proposed and experimentally tested:

1. Core molten material falling into a preflooded cavity and quenching (melt jet quenching [FARO] tests)
2. Pouring of water onto the top of core molten materials and quench (MACE tests)
3. Injection of water into the bottom of molten core pool and quench (COMET tests).

Based on experimental results and evaluation to date, the COMET concept (i.e., bottom injection of water) appears to be the most effective approach.

As noted previously, ANL and UW-Madison have conducted a series of experiments, as part of the DOE Bilateral I-NERI project with Korea, to provide fundamental data needed for refinement of the COMET concept. Key findings from these experiments are as follows:

- The heat removal rate from the melt increases with increasing water injection rate. Beyond some limit, the heat removal rate appears to be independent of the water injection rate. This finding suggests that there exists an optimum water addition rate for melt quenching (Figure D-1).
- For an ambient pressure range relevant to the LWR containments (e.g., 1 - 5 bars), the system pressure has little effect on the heat removal rate from the water and melt.
- The presence of a noncondensible gas in the injected water does not seem to impair the heat transfer from the melt. This finding suggests that the noncondensible gases produced from core-concrete interactions will not impact the injected water heat removal capability.
- Modest ambient pressures and noncondensible gas flows stabilize the quench process.

These findings are based on experiments which employed a metallic melt (500-800 K) rather than the prototypic oxidic melt (2000-2500 K). However, analyses seem to indicate that the overall heat transfer process is controlled by the melt-water mixing driven by the steam being produced. This strongly suggests that the above findings would apply to the case of water injection into the corium melt in the ex-vessel situation. This would need to be confirmed by conducting experiments with a prototypic melt with corium constituents.

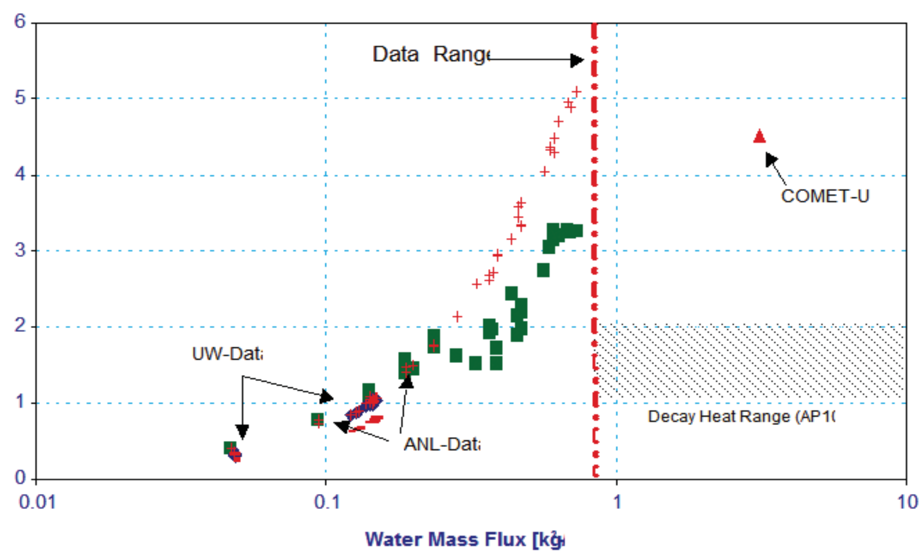


Figure D-1. Volumetric heat removal rate as a function of mass flux. The data indicates ideal heat exchange at low mass fluxes and drops off as the mass flux is increased. The heat removal rate is sufficient to remove decay heat from an AP1000.

Planned Activities

Over the years, a large amount of R&D effort has focused on the ex-vessel coolability issue. Most of the studies centered on individual phenomena relevant to basic issues such as heat transfer or materials interactions. It is now necessary to address this issue holistically with a view toward the implementation of proposed accident management strategies and the objective to preclude containment failure by possible ex-vessel mitigative measures. Possible action plan items are:

- Assess the various water addition concepts for each of the advanced plant cavity designs (i.e., AP1000 and ESBWR) with respect to effectiveness and potential implementation (e.g., failure of in-vessel retention would need to be included for a pre-flooded cavity case).
- Develop an approach to optimize the available water addition concepts for these reactor designs, including minor modifications to the existing cavity designs.
- Identify experimental needs for the above two items. Specifically, evaluate the tests that would be required with water injection into a corium melt with sustained heating.

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

The focus of this task is to conduct separate effect tests for key physical models. These include (1) visualization of melt-coolant mixing using transparent stimulant material, (2) film boiling on spheres in high temperature, (3) two-phase heat transfer in porous media, and (4) development of simultaneous measurement technique for temperature and flow field. High temperature film boiling experimentation has been carried out up to 2000 K of sphere temperature. Direct measurement of heat transfer coefficient is not possible due to the difficulty of instrumentation for high temperature, but the estimation of heat transfer coefficient for saturated film boiling can be made by estimating the steam bubble volume and release period in the visual images. The experimental parameters are: sphere diameters of 19 mm and 12.7 mm, water temperature of 50~100^{o}C, and sphere temperature of 500~1700^{o}C. The saturated flow film boiling heat transfer coefficient ranges 70~200 W/m^{2}K for the sphere temperature 750~1950 K and it increases slightly as the sphere temperature increases. The existing heat transfer correlations for saturated flow film boiling overestimate the present data. A new correlation has been proposed and this correlation can be extended to the temperature range beyond 2000 K.

Film boiling of spheres in a porous environment has been experimentally studied and compared with the dryout heat flux data for top flooding. The film boiling heat transfer coefficients were measured for water temperature 80~100^{o}C and sphere temperature 500~700^{o}C. The porosity was 0.45. It is observed that the effect of water subcooling is small for the porous medium case because the instrumented sphere is actually experiencing heated water below even in the case of subcooled water initially. Using the measured heat transfer coefficients, the volumetric heat transfer coefficient can be calculated for saturated water at 50~150 kW/m^{3}K. Also the heat removal capacity per unit volume in this porous medium was calculated and compared with the dryout heat flux data for porous medium of 3 mm diameter spheres reported by Barleon et al. (1984)(Figure D-2). The present data are for 22.2 mm-diameter porous medium and it is higher than the dryout heat flux compared. But the cooling rate in film boiling can be lower than the dryout heat flux if the same diameter spheres are used. This will be conducted in the next year.

Task IV: Integration of Analytical Models (KMU lead)

The progress in the first year was to check the adaptability of commercial code CFX4.4 to melt coolability problem. The results showed positive signs in some numerical calculations but revealed certain limits in handling three or more phases. The fatal problem is that it cannot deal with the interphase transfer between two continuous media. Only continuous-disperse pair is possible in calculating interphase transfer. The only way to make it possible for continuous-continuous pair is to assume one fluid disperse with a certain diameter, which may cause a large error to result in solution failure. The CFX5.6, an updated version of 4.4, adopts a coupled algorithm and possibly saves computing time by faster convergence. In addition, a solution for the mixture model becomes possible by user input subroutines. The model for drag and heat transfer coefficient must be set by the users. These models will be established by further study. During this year, some sample problems were set up for computational work. Only the results for a sample computational domain, which is similar to but not exactly the same as the UW experimental setup, are shown in this report. The geometry used for computation is a cylinder 60 cm high with a 10 cm diameter. The whole container is assumed to be 'the computational domain'. The lead melt of 690 K in temperature is filled up to 30 cm in height. The inlet is located at the center of the bottom face and its dimension is 2 mm in diameter. A water injection rate of 10 g/s has been used. Extensive analyses were not carried out at this point due to the time-consuming nature inherent to this kind of work.

Some other geometry and melt-coolant pairs have been tried for comparison. One of these trials is the R22-water pair, which is used for visualization purposes at KMU. More computational work for accurate results will be performed next year after setting up appropriate transfer models to be adapted to the CFX5.6 code.

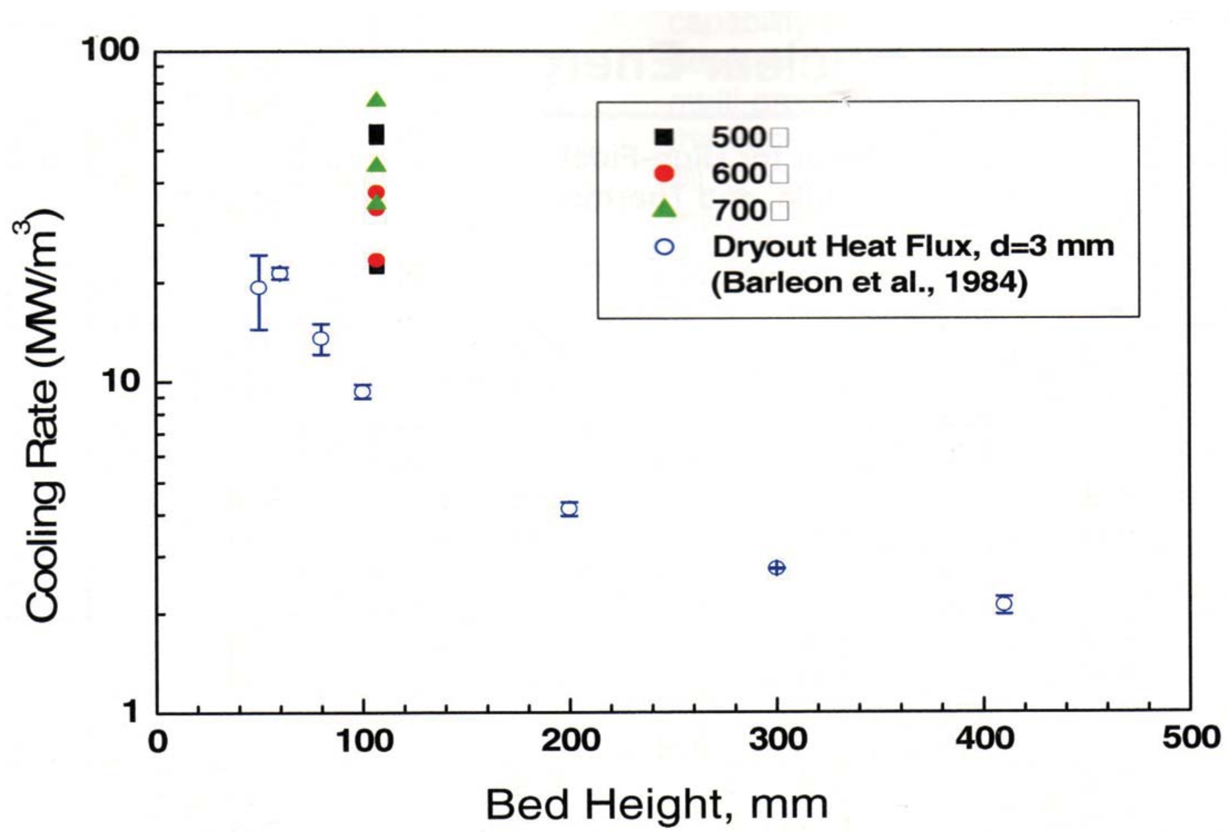


Figure D-2. Film boiling heat removal per unit volume from spheres in a porous environment compared to dryout heat flux data of Barleon et al., 1984.

International Nuclear Energy Research Initiative

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

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

Principal Investigator (Korea): K. Kim, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Purdue University, Seoul National University (SNU)

Project Number: 2002-010-K

Project Start Date: January 1, 2002

Project End Date: December 31, 2004

Research Objectives

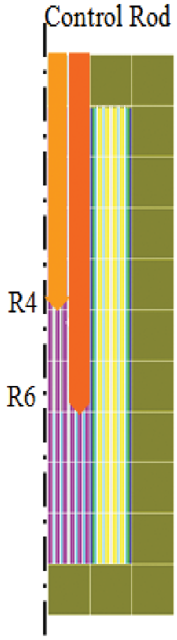
A comprehensive high-fidelity reactor core modeling capability is being developed for detailed analysis of current and advanced reactor designs. The work involves the coupling of advanced numerical models such as CFD for thermal-hydraulic calculations, whole core discrete integral transport for neutronics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

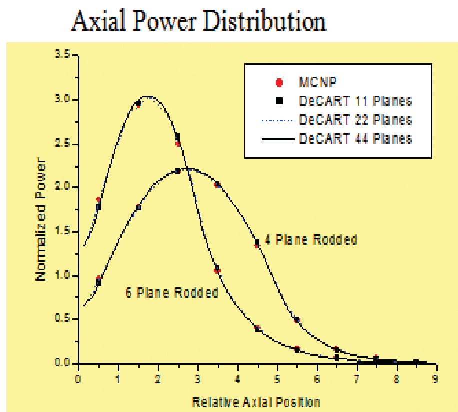
In each of the phenomenological areas, including reactor physics, thermal-hydraulics, and thermo-mechanics, the objective was to develop and demonstrate the ability to calculate accurately the individual key phenomena. The integration of these high-fidelity models into a robust computational tool, along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

Research Progress

The reactor physics module, being developed at KAERI and Seoul National University (SNU), is a whole core transport code Deterministic Core Analysis based on Ray Tracing (DeCART), based on the method of characteristics. This code generates sub-pin level power distributions by representing local heterogeneity explicitly without homogenization, using a multigroup cross-section library directly without group condensation and incorporating pin-wise thermal hydraulic feedback. As part of the efforts to verify this module, a Monte Carlo computational scheme with pin-by-pin thermal hydraulic feedback capability has also been developed by partners at Seoul National University. Comparisons between conventional methods on benchmark problems, fuel assemblies, and whole core calculations indicate excellent performance in terms of accuracy and computational time.

An extensive verification program for the steady-state version of DeCART has been conducted and verification of the transient capability has been performed. For the steady-state DeCART, benchmark problems included the C5G7MOX 2-D and 3-D and Modified Rodded Variations, as well as the VENUS2 3-D critical benchmark. Eigenvalue errors of less than 100 pcm were observed and pin power errors were less than a few percent for the C5G7MOX rodded problem. Figure D-3 shows excellent comparisons between DeCART and the standard Monte Carlo code, MCNP, for the C5G7MOX rodded problem. Consistent comparisons to the Monte Carlo code with thermal-hydraulic feedback developed at SNU, known as MCCARD, have also been performed for a variety of problems and conditions. Figure D-4 shows one such result for a 3-D assembly at hot zero and full power conditions. Most recent developments have focused on implementation of a transient capability into DeCART, implementation of a depletion capability, and updating the DeCART multi-group cross-section library. Regarding the transient capability of DeCART, a transient MOC solver was newly implemented as one of the solver development efforts and a series of cross-comparisons with VARIANT-K and PARCS was successfully performed for the verification of the DeCART transient solver. A simple comparison of DeCART with the HELIOS lattice physics code for the depletion capability was performed for the verification, which showed excellent agreements. Regarding the cross-section library update, a new procedure to generate resonance integrals and subgroup parameters employing the MCCARD Monte Carlo was developed.





Number of Planes	Plane Thickness cm	Eigenvalue Error, pcm		3-D Pin Power RMS Error, %		Maximum Absolute 3-D Pin Power Error		Maximum Relative 3-D Pin Power Error, %	
		R4	R6	R4	R6	R4	R6	R4	R6
11	21.42	-1.5	-25.7	2.95	4.68	0.140	0.229	6.67	12.12
22	10.71	-20.8	-76.4	0.97	1.87	0.061	0.118	3.71	3.39
44	5.355	-23.7	-82.7	0.84	1.68	0.049	0.100	3.84	3.74
88	2.6775	-23.7	-81.9	0.84	1.67	0.048	0.099	3.80	3.79

Figure D-3. Rodded C5G7MOX Benchmark Problem.

Core State	MCCARD(M)	DeCART(D)	
	k-eff	k-eff	D-M pcm
HZP	1.44423	1.44405	-9
HFP	1.42666	1.42753	43

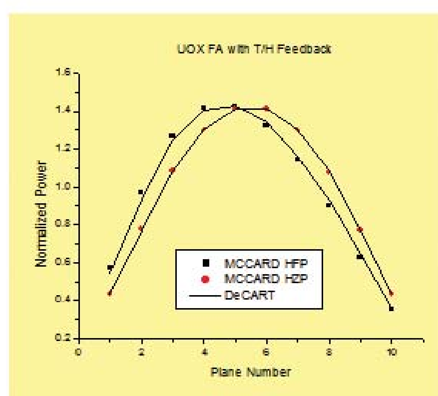


Figure D-4. 3-D Assembly with Power and Flow.

Thermal-hydraulic analysis has been focused on the use of high-fidelity computational fluid dynamics capabilities available in several commercial CFD software programs, with specific focus being applied to the STAR-CD and CFD-ACE codes. While it is recognized that these CFD codes have the theoretical capability of simulating events with fine detail in a reactor application, a demonstration of their ability to predict observed flows in rod bundle geometry was considered critical for their ultimate inclusion in the integrated code system.

Project efforts in thermal-hydraulics at ANL and KAERI have thus focused on experimental validation of these codes, with particular emphasis on demonstrating their ability to predict turbulent flow and associated heat transfer in rod bundle flows. During the reporting period, the project has concentrated on the evaluation of these models in heated bundles, using the experimental data of Krauss and Mayer (Nucl. Eng. & Design, 180, 1998). Figure D-5 illustrates the ability of various turbulence models to predict flow characteristics, including sub-channel mean velocities, turbulent flow, and temperature distributions. Although completely accurate prediction of turbulent structures in the subchannel is yet to be demonstrated, prediction of heat transfer coefficients and temperature distributions considered critical for these applications was quite good for forced flow. Improvement in calculational results based on higher order turbulence models, such as the Reynolds Stress Model (RSM), can be seen in Figure D-5. Additional calculations for assemblies with spacer grids have been performed, illustrating the ability of the CFD codes to predict complex flow patterns down stream of the mixing vane (see Figure D-6).

KAERI analysis using CFD-ACE, ANL analysis using STAR-CD

- Fairly good agreement for mean velocity, temperature, turbulence intensity
- Improved predictions with non-linear models that account for anisotropy

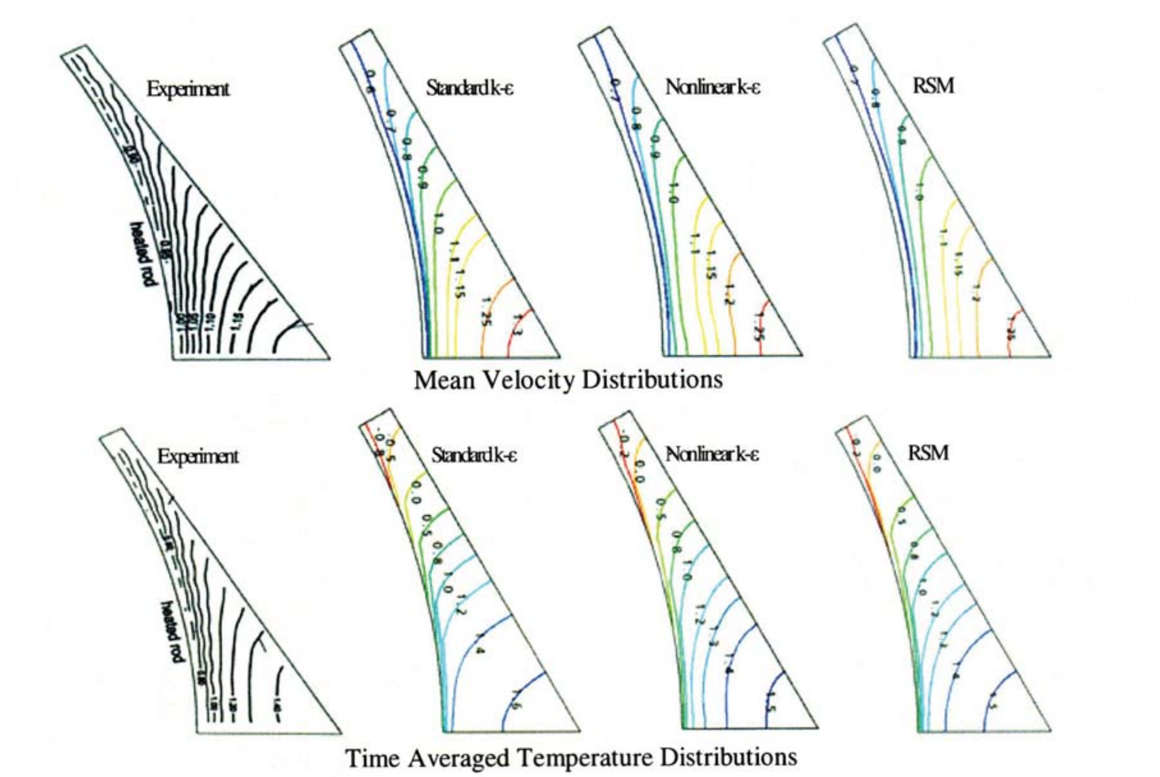


Figure D-5. Turbulent Heat Transfer in Rod Bundles.

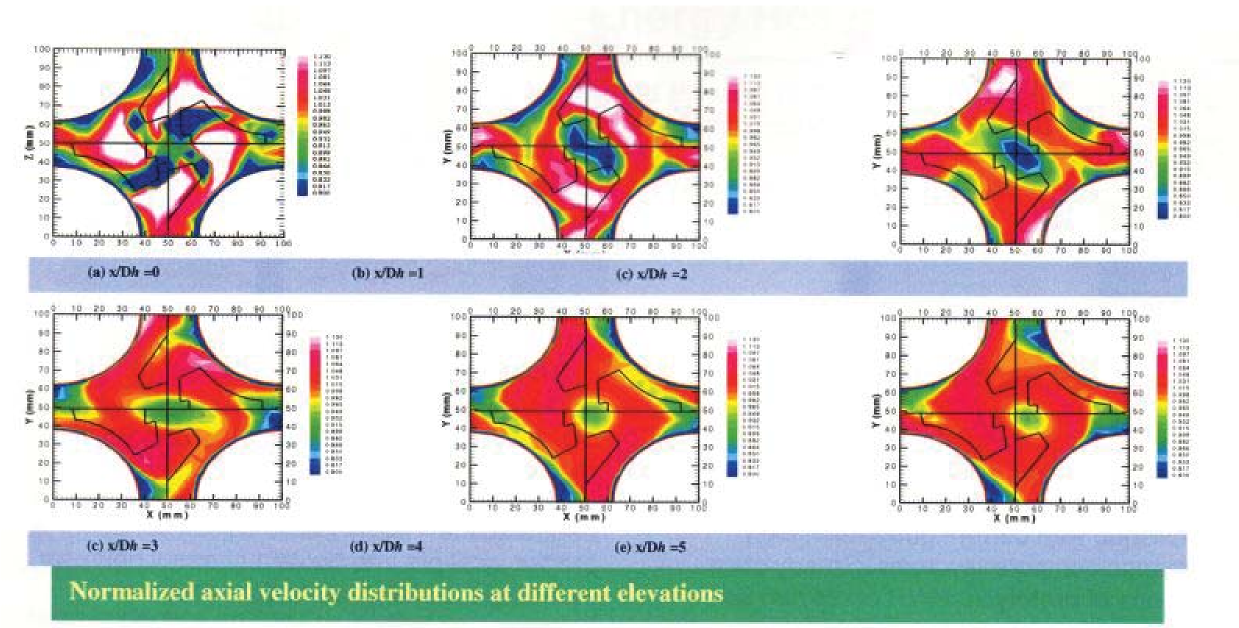


Figure D-6. Pressurized Water Reactor (PWR) Fuel Assembly with Spacer Grids.

The third phenomenological element of the integrated code relates to the modeling of thermo-mechanical response. One of the objectives of the I-NERI project is the coupling of the CFD models with ANL's structure finite element code NEPTUNE for integrated whole-core analyses. The conceptual methodology of the coupling was investigated by using the dynamic grid deformation of the STAR-CD CFD code. The results of the current study indicate the concept of utilizing the dynamic grid formation will provide an efficient coupling method between the thermo-mechanical and thermal-hydraulic calculations in the numerical reactor tool.

The ultimate objective of this effort is to produce an integrated analysis capability utilizing the multi-physics models, having demonstrated the validity of the individual components. General purpose coupling schemes have been developed and intercompared for quality assurance. Strategies for exchange of relevant information among the modules have been investigated and they demonstrated the importance of proper mapping between the vastly different neutronics and thermal hydraulics grids. From the convergence standpoint, selection of optimal timing for exchange of information among models has been examined.

Demonstration of the integrated analysis capability has been initiated for a series of sample problems. The testing sequence included single-pin models, multipin models, fuel assembly models, multi-assembly models and, most recently, whole-core models. The whole core model for a small PWR consisted of three different types of fuel assemblies with varying positions of guide tubes, control rods, and fuel pins with burnable poison. The ¼ core DeCART model had 3½ million flat flux regions, 45 energy groups, and 8 azimuthal and 4 polar angles in 90°. The 1/8 core STAR-CD model used 64 million cells. DeCART/STAR-CD calculations were performed on the ANL Beowulf cluster, JAZZ. For DeCART, 12 processors were used, while 57 processors were used for STAR-CD. Figure D-7 illustrates the results for pin power and temperatures, illustrating the increase in power around control guide tubes and in pins along the core periphery where the pins are near two reflector assemblies.

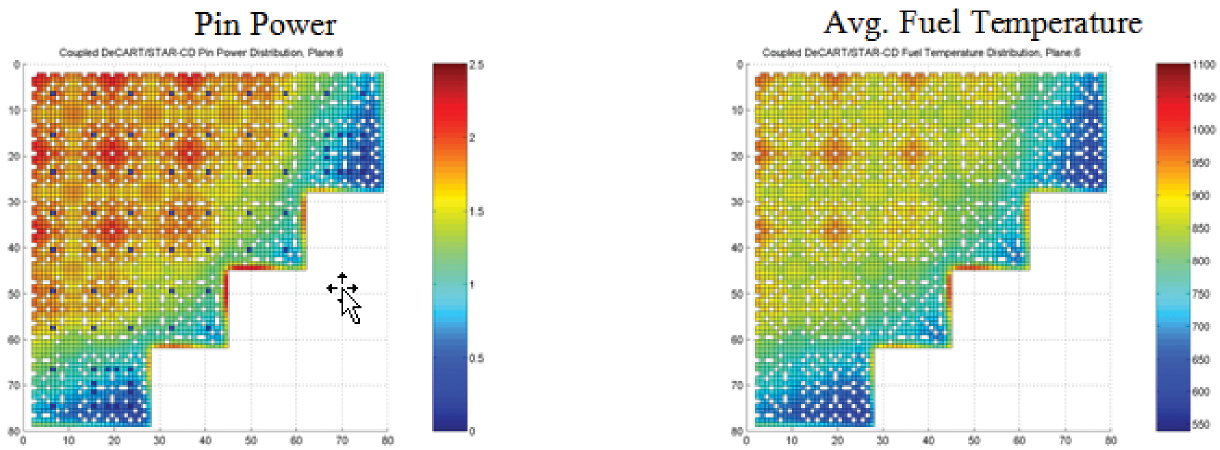


Figure D-7. Small PWR Coupled Results: Temperature and Pin Power.

Computational results on the JAZZ multiprocessor Linux cluster at Argonne confirm that whole core applications can be accomplished with current generation high-performance computer systems. Steady-state calculations of prototypic PWR cores can be performed on a teraflop class machine in less than a day. Extrapolation of these estimates suggest that transient calculations for relevant scenarios can also be accomplished in computing times of several tens of hours on such machines. Scalability of results to date also indicates that expected availability of more powerful machines will result in proportional reduction in computing time, with the expectation that such whole core high fidelity calculations will soon be possible in times measured in hours, rather than days or weeks.

Planned Activities

Final activities during this last year of the project are focused on verification of transient neutronics capabilities, implementation of depletion capability, continued validation of CFD models, integration of CFD with thermo-mechanics, and performance of integrated calculations. The integrated calculations are focused on improving the numerical iteration strategy for the multi-physics problem to reduce the overall computing time for steady-state and transient problems. The sample whole-core problem currently being examined is the thermal-hydraulic and neutronic behavior of a PWR that may be subject to local subcooled nucleate boiling crud formation and deposition and the neutronic effect known as Axial Offset Anomaly. This problem requires the capability to perform high fidelity, coupled neutronic/thermal-hydraulic calculations to predict local pin and coolant channel behavior, including the ability to predict azimuthal behaviors on individual pins. The Numerical Nuclear Reactor is one of the few codes with the required capabilities.

International Nuclear Energy Research Initiative

Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Super-critical Reactors

Principal Investigator (U.S.): D. McEligot, Idaho National Laboratory (INL)

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

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

Project Number: 2002-016-K

Project Start Date: December 11, 2001

Project End Date: October 31, 2005

Research Objective

The ultimate goal of this Korean/U.S. laboratory/university collaboration of coupled fundamental computational and experimental studies is the improvement of predictive methods for Generation IV reactor systems (e.g., Super-Critical-Pressure Water Reactors [SCWRs]) and associated AFCI and NHI activities. The general objective is to develop the supporting knowledge of advanced computational techniques needed for the technology development of the concepts and their passive safety systems. The resulting specific objectives are to develop and to extend DNS, LES, and differential second moment closure (DSM) techniques to treat super-critical property variation and complex geometries, thereby providing capabilities to:

- Assess predictive capabilities of current codes for SCWRs, VHTGRs, etc.
- Provide bases to improve nuclear reactor thermal-hydraulics safety and subchannel codes
- Provide computational capabilities where current codes and correlations are inadequate
- Give predictions for Generation IV conceptual and preliminary designs for:
  - Full power operation (LES, DSM and Reynolds-averaged Navier-Stokes approaches)
  - reduced power operation
  - transient safety scenarios
- Ultimately, handle detailed thermal-hydraulic flow problems for final Generation IV designs for improved performance, efficiency, reliability, and safety, and reduced costs and waste.

This project provides basic thermal fluid science knowledge to develop increased understanding for the behavior of superheated and super-critical systems at high temperatures, for application and improvement of modern computation and modeling methods, and for incorporation of enhanced safety features.

Research Progress

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

Variations of fluid properties along and across heated flows are important in SCWRs, VHTGRs, and Gas-Cooled Fast Reactors (GFRs), all Generation IV reactor systems concepts. Significant differences and uncertainties have been found between thermal-hydraulic correlations for these conditions. Improved computational techniques and supporting measurements are needed to assist the developers of codes for reactor design and systems safety analyses to treat the property variations and their effects reliably for some operating conditions and hypothesized accident scenarios of these reactors. The geometries of the reactor cooling channels of some SCWR concepts are demonstrated in Figure D-8. Most of these geometries are more complex than those that have been used to generate the empirical correlations employed in the thermal-hydraulic codes. Advanced computational techniques may be applied but measurements with realistic geometries are needed to assess the reliability and accuracy of their predictions.

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

DNS employs no turbulence modeling; it solves the unsteady governing equations directly. Consequently, along with measurements, it can serve as a benchmark for assessing the capabilities of LES, DSM, and general RANS techniques. It also can be applied for predictions of heat transfer at low flow rates in reduced power operations and transient safety scenarios—such as loss-of-coolant or loss-of-flow accidents—in SCWRs, GFRs, and VHTRs. For SCWRs it can handle sensitive situations which are difficult to treat properly with correlations or with many turbulence models. Once validated, LES and DSM techniques can be applied for predictions at higher flow rates, such as near

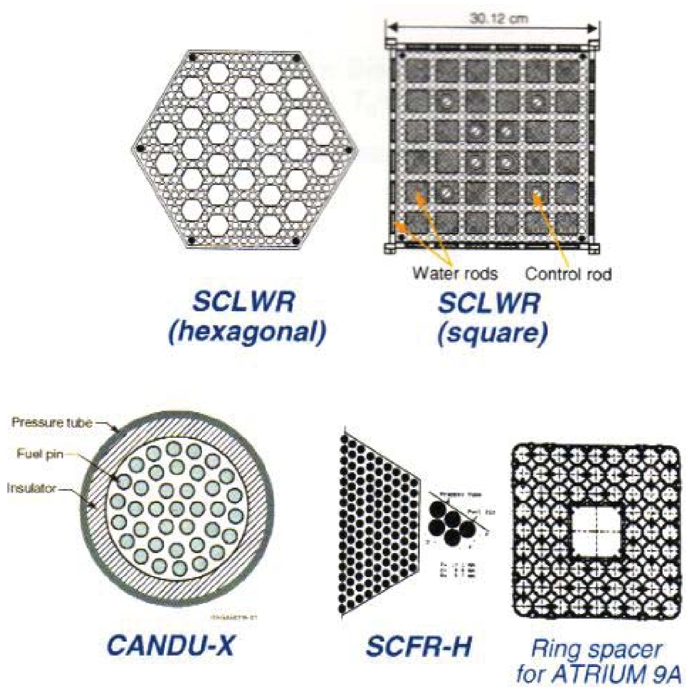


Figure D-8. Proposed Designs for Fuel Assemblies in Some Super-Critical Water-Cooled Reactor Concepts.

normal full-power operating conditions, for these Generation IV reactor concepts. The flow facility developed at SNU provides a means to measure heat transfer to super-critical fluids for assessment of the effects of their property variations, and the miniaturized multi-sensor probes from the University of Maryland will permit measuring the turbulence, which is modeled by the codes. The INL experiment models the complex geometry of coolant passages in an SCWR concept to provide benchmark data.

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

The following was accomplished during this reporting period:

- Developed digital flow visualization for turbulent DNS code for annular geometry with super-critical flow (Figure D-9) and continued analysis of turbulent super-critical flow in annular geometries with heat transfer.
- Extended turbulent, super-critical LES code to developing flows and continued extending turbulent, super-critical LES code to complex geometries.
- Applied DSM code to examine turbulence models with property variation and examined the modeling of turbulent heat flux in the pseudocritical region.
- Prepared for LDV and PIV measurements simulating flow in a Super-Critical Water-Cooled Reactor in a unique large Matched-Index-of-Refraction flow system and obtained initial LDV data (Figure D-10).
- Developed and calibrated a two-sensor miniaturized hot-wire probe (Figure D-11) for use in upcoming super-critical heat transfer experiments at SNU and trained students in the use of the probes for measurement of temperature and velocities in a super-critical fluid. Designed and constructed a traversing mechanism to move the probe inside a high pressure CO_2 flow facility.
- Obtained the first measurements of heat transfer to super-critical flow in small square and triangular tubes (Figure D-12) and measured heat transfer and pressure drop to super-critical CO_2 with a small circular tube for 59 sets of conditions.

Since January 2002, the project partners have had 30 archival papers published or in press, 44 conference presentations, and 12 invited presentations relating to this collaborative Korea/US I-NERI project. They also had over 120 publications and presentations on other topics.

Planned Activities

During the remainder of the third phase:

- Continue simulation studies on the immersed boundary technique for complex geometries and the ribbed annulus case and make comparisons with DNS and experimental results obtained by Korean colleagues for super-critical flows of CO_2 under high heating conditions
- Conduct and document LDV (and possibly PIV) measurements of the velocities and turbulence with a model simulating features of a complex Generation IV SCWR coolant channel with grid spacers
- Address optimization of the hot-wire calibration parameters in the pseudocritical temperature region and prepare a journal article on hot- and cold-wire response in super-critical CO_2 flow
- Submit at least five topical reports and the final technical report
- Related technical papers will be published and technical presentations will be made.

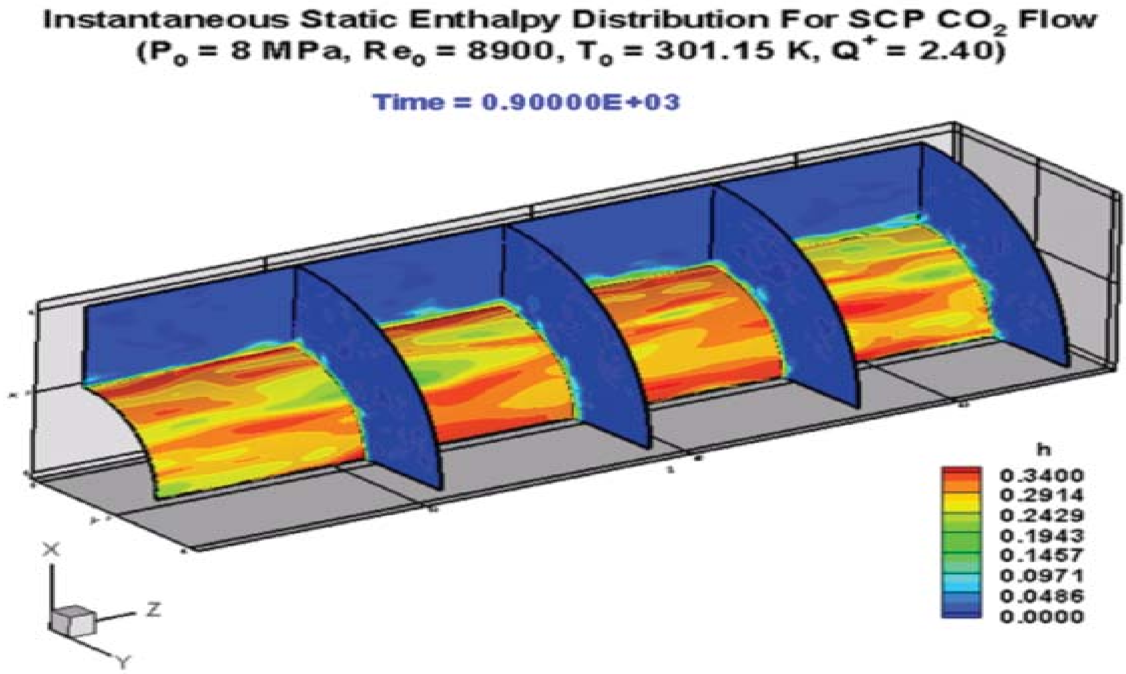


Figure D-9. Direct numerical simulation of pseudocritical annular flow along a heated rod (h_{pc} = 0.0489). The gas-like region forms a very thin insulating layer near the heated surface.

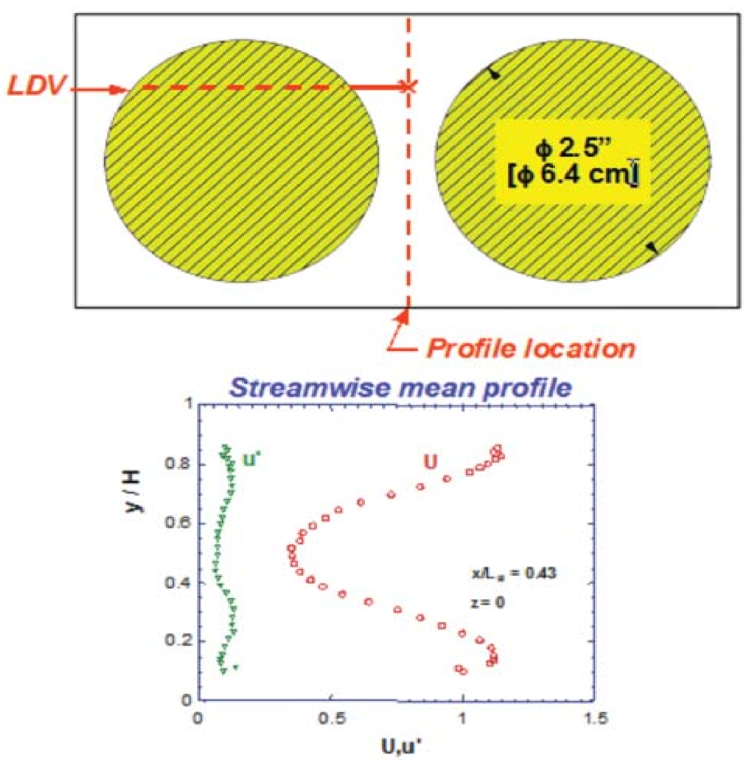


Figure D-10. Initial laser Doppler velocimeter measurements
of flow features in a large-scale idealized model of an SCWR fuel assembly.
Matching of the refractive indices of the fluid and rod allows undistorted access to the profile plane.

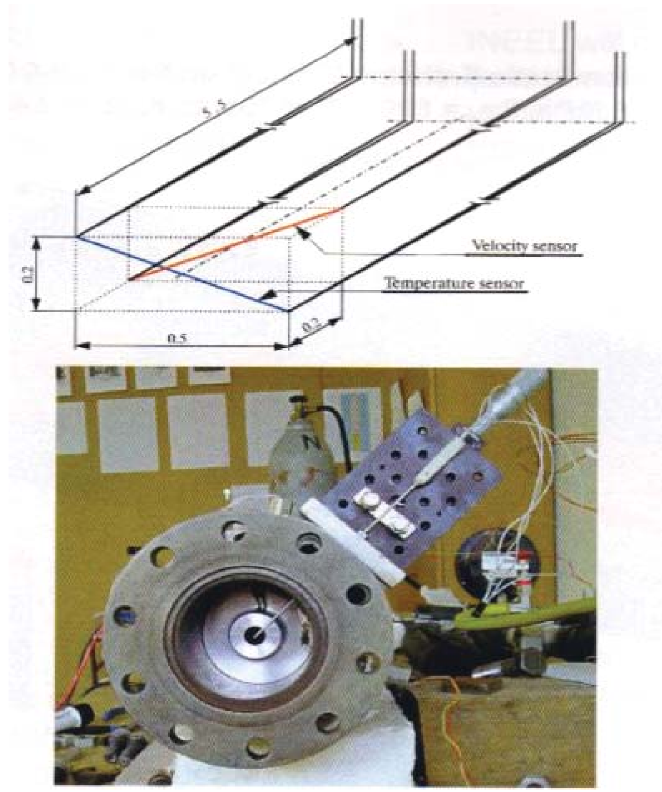


Figure D-11. (a) Schematic view of miniature hot-wire probe (dimensions in mm) and
(b) photograph of probe mounted in front of nozzle for calibration in heated super-critical flow.

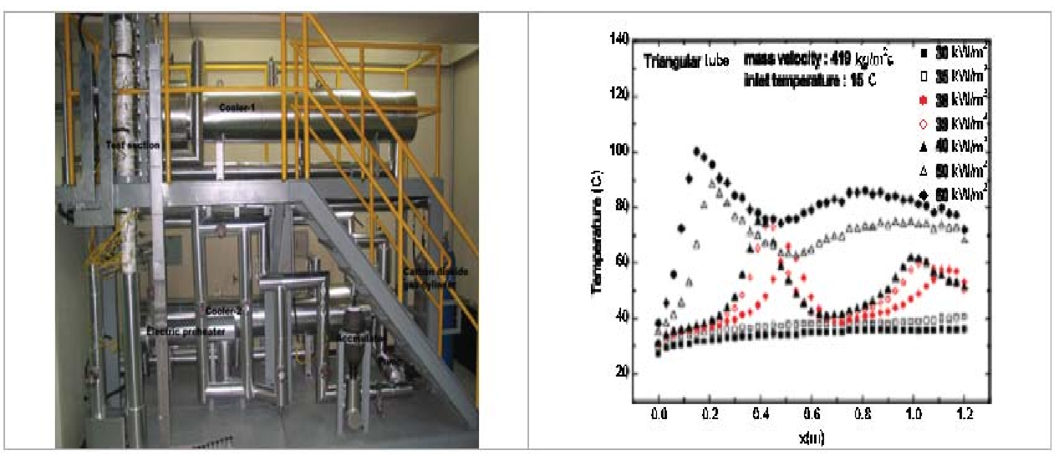


Figure D-12. (a) SNU facility for measurements of heat transfer to super-critical carbon dioxide and
(b) first measurements for small triangular tubes (hydraulic diameter = 9.7 mm). As with circular tubes,
"deterioration" is sensitive to the surface heat flux.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): D. Holcomb, Oak Ridge National Laboratory (ORNL)

Principal Investigator (Korea): M. Na, Chosun University

Collaborators: Ohio State University (OSU), Korea Atomic Energy Research Institute (KAERI), Cheju National University

Project Number: 2002-020-K

Project Start Date: December 11, 2001

Project End Date: December 31, 2004

Research Objectives

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

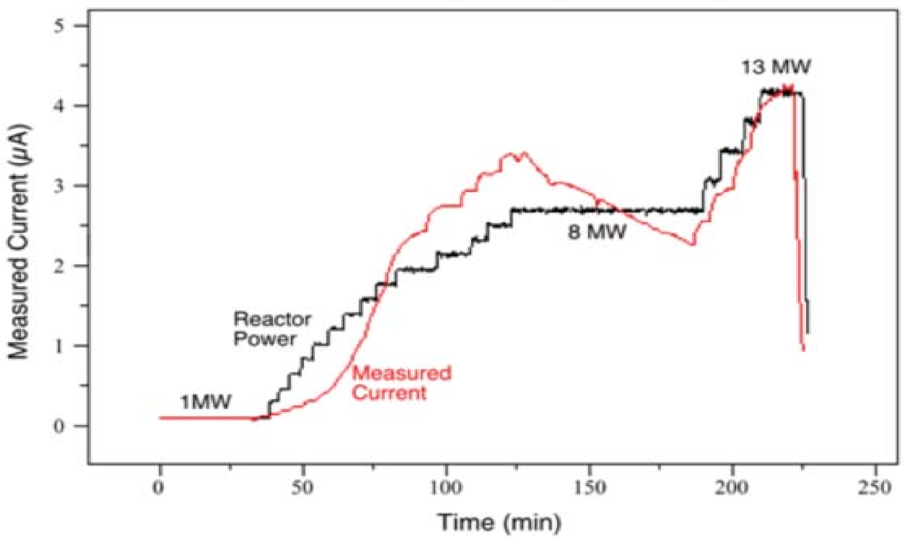
The project consists of three major tasks. The objectives of the first task are to evaluate the basis for current reactor operation strategies, including assessment of the state-of-the art for primary system measurement; investigate the effects of measurement limitations on the operational performance of existing nuclear power plants; and identify potential operational/safety improvements resulting from improved measurement and control. The objective of the second task is to develop and demonstrate three advanced sensors: a solid-state in-core flux monitor (SSFM) applicable to high temperature reactors (Figure D-13), a first-principles and thus drift free temperature measurement system (Johnson noise based), and the component technologies required to make magnetic flowmeters function on the primary side of pressurized water reactors. The objective of the third task is to take advantage of the benefits of improved sensors by devising advanced reactor operational strategies that optimize core performance and permit reduced operational margins.

Research Progress

All of the major project tasks have either been accomplished or are nearing completion. Task I (Evaluation of the Basis for Current Reactor Operation Strategies) is now complete. Task I was comprised of three subtasks: 1) assessment of the state of the art for primary system measurements, 2) investigation of the effects of measurement limitations on operational performance of existing nuclear power plants, and 3) identification of potential operations/safety improvements. Reports on these subtasks were issued as part of the second year annual project report.

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

The SSFM development and evaluation task is now essentially complete. Conceptually, the SSFM is based on employing a polycrystalline aluminum nitride compact as a flux-sensitive resistor. ORNL has now designed, fabricated, and delivered (to OSU and KAERI) three generations of prototype sensors. The design refinements have allowed using standard semiconductor fabrication technology as the basis for device assembly enabling repeatable, inexpensive devices. KAERI completed its work with the SSFM packaging and testing and has produced a report on its efforts. OSU is continuing its investigations into the detailed performance of the SSFM devices. While the developed sensors function in rough accord with theoretical expectations, as anticipated with any novel sensor device concepts, the SSFM exhibits some undesirable attributes: high temperature current leakage and sensor polarization. The sensor technology will require further refinement before deployment in NPP applications.



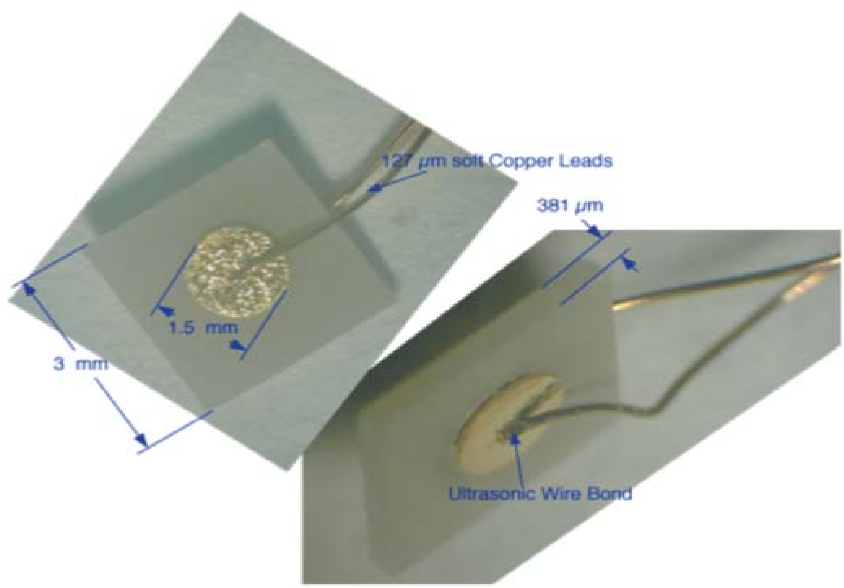


Figure D-13. Flux Monitor Compacts and the Measured SSFM Current Compared to the Hanaro Reactor Power.

The central reasoning underlying the Johnson noise thermometry (JNT) Subtask 2.2 is that all available versions of temperature measurement technology drift under the harsh environment of a nuclear power plant. Knowledge of the plant thermal condition is both a principal performance and safety requirement for both current reactors as well as all of the proposed future reactors. While the device being developed and demonstrated in this project is specifically intended to provide a first-principles measurement of temperature within the primary coolant loop of PWRs, Johnson noise is a more general technology equally applicable to direct in-core temperature measurement. The underlying technology being developed under the current program can be applied to the reactors of the Generation IV and Nuclear Hydrogen initiatives. Changes in material characteristics over time do not affect the fundamental physics of how Johnson noise is produced. Because of the fundamental nature of JNT, calibration is not needed. The U.S. Nuclear Regulatory Commission mandates periodic calibration of primary coolant system temperature measurement instruments. Without an assured measurement uncertainty, safety margins would be required to be increased leading to reduced plant thermal efficiency. In high-temperature reactor designs, available sensors change calibration very rapidly making their use impractical. Calibration activities are time consuming, costly, and increase exposure risk to technicians.

The main effort during FY 2004 for the JNT subtask was refinement and evaluation of the developed measurement channels. Additionally, KAERI has developed a digital signal processing stand-alone unit incorporating the JNT logic removing the requirement for using a computer in the measurement. The developed JNT hardware represents a significant technical achievement as these units are by far the most advanced Johnson noise measurement systems ever actually built. Attempts to measure temperatures at nuclear power plants using Johnson noise thermometry have now been going on for more than 30 years and the use of Johnson noise for temperature measurement is now more than 50 years old. Johnson noise thermometry is not yet used at any nuclear plant and, more generally, Johnson noise temperature measurement instrumentation is not currently commercially available. Simply put, Johnson noise thermometry, while conceptually highly appealing has proven very difficult to implement in practice in a sufficiently robust and cost-effective manner as to be viable. The hardware developed under this task exhibits all of the necessary component technologies for in-plant application. Note, however, that this project specifically excluded radiation-tolerant electronics implementation as outside the focus on demonstrating the combination of advanced digital signal processing algorithms with a dual-mode Johnson noise and resistance temperature measurement device. Radiation-tolerant electronics would need to be developed before direct application of the new JNT hardware into operating nuclear power plants.

Subtask 3.3 has made significant strides in developing and demonstrating the component materials technology to enable magnetic flowmeter deployment within primary loop piping of PWRs. Primary loop flow measurements are used to determine the core heat rate in PWRs and as such are a basic safety indication. These measurements are conventionally made using flowmeters based on differential pressure. Differential pressure-based flowmeters have significant fundamental accuracy limitations as well as having failure modes difficult to diagnose while in service. Magnetic flowmeters offer a potential solution to these limitations. Magnetic flowmeters are highly accurate, respond linearly, and are obstructionless (no fouling; consume no pumping power). Also, the transmitter for magnetic flowmeters can be located remotely (up to hundreds of feet) from the point of measurement, thus reducing environmental exposure. The major limitation to the immediate application of magnetic flowmeters to nuclear power plants is the radiation sensitivity of the non-conductive inner pipe liner. Ceramic pipe liners are currently available for pipe diameters up to 30 cm. However, for larger pipes, only radiation sensitive materials such as Teflon™ or rubber are available. Ceramic pipe liners are not currently available for larger diameter pipes due to manufacturing and material limitations. ORNL has produced a series of magnetic flowmeter ceramic liner test components and supplied them to both KAERI and OSU for testing. The test pieces have been irradiated to 1.8 MGy (equivalent to 15 full-power years) and their mechanical performance has been evaluated. The test components show no evidence of degradation in material strength or mechanical integrity after irradiation.

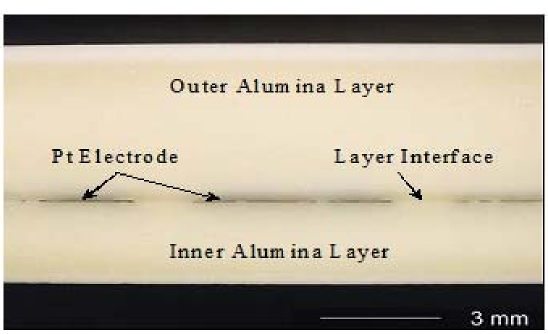
The flowmeter liner consists of an alumina tube having an embedded electrode layer near the inside diameter surface shrink-fitted into stainless steel primary piping. Platinum ink is used to form the internal electrode in the alumina tube. This is accomplished by using a two-step gelcasting procedure to form the tube. A thin-walled tube is first gelcast and electroded on the outer diameter surface. A second, thicker tube is then gelcast against the electroded surface of the first tube, resulting in an embedded electrode. The Pt electrode is an interconnected but discontinuous layer in the laminated structure. The sintering process used to densify the alumina ceramic hermetically seals the electrode inside the tube wall. The strength tests combined with the examination of the laminated alumina samples with embedded electrodes confirm that the concept of a gelcast laminated alumina ceramic liner is a feasible approach to producing a large-scale, ceramic-lined magnetic flowmeter (Figure D-14).

Task 3 (implementing improved measurements into reactor control algorithms) is also nearing completion. Under Task 3, a three-dimensional reactor core kinetics model integrated with a thermo-hydraulic model of a reactor core has been developed and implemented. This developed reactor core model was used to design the protection and monitoring algorithms that focus on Departure from Nucleate Boiling Ratio (DNBR) and local power density (LPD), and also to design advanced control algorithms. Two kinds of methodologies were applied to develop the protection and monitoring algorithms: model-based method and data-based method. The model-based method is directly achieved by the 3-D reactor core model. The data-based method uses fuzzy neural networks, of which the inputs are measured signals including in-core neutron detector signals, etc. These methods were known to have a larger margin compared to the existing methods. The developed controller has been applied to verify controllability for the load-following operation for the daily load cycle of YGN-3, which is a prototype of Korea Standard Nuclear Power Plants (KSNPs) and is simulated numerically by the reactor core design code MASTER (Multipurpose Analyzer for Static and Transient Effects of Reactor). In addition, Chosun University (CU) and Chungnam National University (CNU) have integrated the MPC controller for load-following operation and the core monitoring algorithms for DNBR and LPD with MASTER. The controller was applied to the integrated power level and to axial power distribution controls for KSNPs.

CU & CNU also designed a robust proportional-integral-derivative (PID) controller for load following operation. More realistic conditions were applied to the methodology used in the robust PID development during this year—overlapped control rod maneuvering was considered. Using the 3-D reactor code MASTER, the relationship between rod movement and reactor power was obtained to model the reactor plant. Three variables were used to establish the model: initial power, initial rod position, and boron concentration. A program was developed which determines the reactor model with these three initial conditions.

Planned Activities

Almost all the technical work of the project is now complete and each of the technology development subtasks has advanced the state-of-the-art in its area. However, the developed technologies are at an advanced prototype level of development and require further refinement for in-plant usage. The focus for future activities is thus on developing partnerships with commercial entities for producing commercial versions of the new technologies.



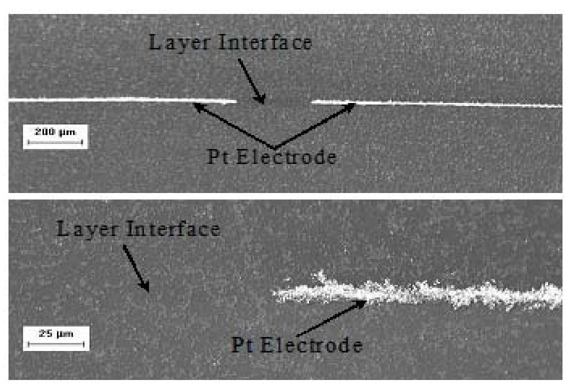


Figure D-14. Optical Microscope Image of the Cross Section of the Laminated Alumina Sample and High Magnification Optical Images of the Pt Electrode and Layer Interface.

International Nuclear Energy Research Initiative

Condition Monitoring Through Advanced Sensor and Computational Technology

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

Principal Investigator (Korea): J. -T. Kim, Korea Atomic Energy Research Institute (KAERI)

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

Project Number: 2002-021-K

Project Start Date: January 1, 2002

Project End Date: December 30, 2004

Research Objective

The overall goal of this joint research project is to develop and demonstrate advanced sensors and computational technology for continuous monitoring of the condition of components, structures, and systems in advanced and next-generation nuclear power plants. This project has conducted condition monitoring tests on check valves and piping elbows using several advanced sensors such as optical fiber sensors, acoustic emission, ultrasonic devices, accelerometers, and chemical sensors. The project team investigates and develops sophisticated signal processing, noise reduction, and pattern recognition techniques and algorithms to differentiate various degradation stages of these two components.

Research Progress

The KAERI Check Valve Test Loop

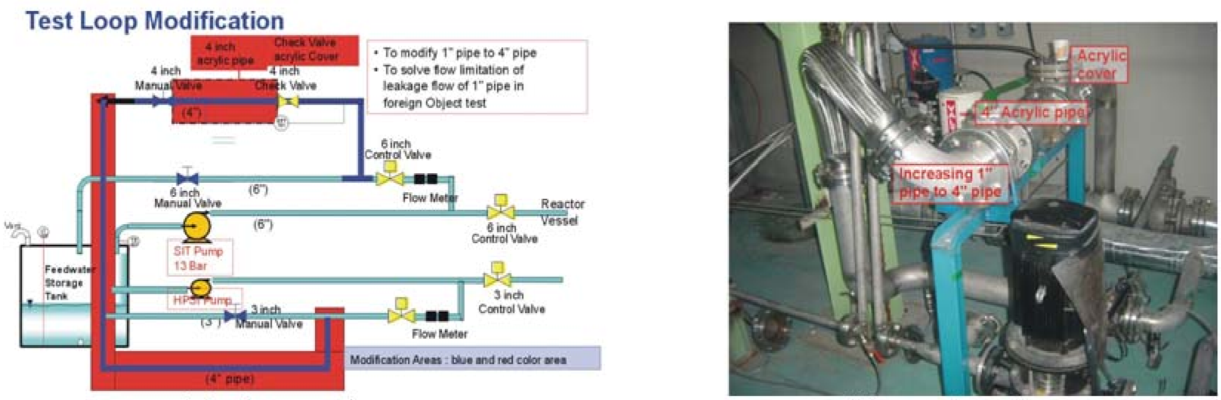
The KAERI check valve test loop was modified to increase the leakage flow rate by using the 4-inch connecting pipe to replace the 1-inch pipe. Figure D-15 shows the schematic diagram and the actual installation of the modified test loop with an increased leakage flow. In March 2004, two sets of condition monitoring tests were performed with the modified test loop for the abnormal configurations with foreign object interference and disc wear.

The ARL Check Valve Test Loop

Condition monitoring tests with the ARL check valve test loop, as shown in Figure D-16, were performed to mirror those of KAERI partners and include the normal and degraded conditions with foreign object interference and disc wear. Experiments were conducted over a sweep in pressure range (1 – 10 bars), including static pressure.

Data Processing Analyses for the Check Valve Test Data

The processing analysis of acoustic emission sensor data from the KAERI check valve test loop was performed. The processed results indicate that the acoustic emission sensor data provide a clear differentiation between the normal and the faulted conditions, and a resolution for different levels of disc wear simulated in the condition monitoring tests. The analyst observed both pulsed and continuous noises that contaminate the sensor data, generated by sources not directly associated with the check valve under test. However, no noise reduction techniques were applied to the data in the analysis process, suggesting that the sensor data from degraded valves may have sufficient strength to support detection and classification even in an operational, non-laboratory setting.



(a) Schematic diagram (b) Actual installation

Figure D-15. Modified Check Valve Test Loop at KAERI.

Figure D-16. ARL Check Valve Test Loop.

With proper analysis, procedures such as Music (multiple signal classification), the sensor data from the faulty condition with disc wear are repeatable and clearly distinguish between normal and abnormal conditions. Figures D-17 and D-18 show the level and frequency differences for the Music pseudo spectrum of a normal check valve and a degraded check valve with disc wear at 9-bar pressure, respectively. Differing degrees of disc wear produce sensor data having reproducible pseudo-spectral peaks at distinct frequencies, when evaluated at a constant operating pressure. The sensor data from degraded valves often show characteristic features, suggesting that the fundamental physical processes induced by component failure might be identifiable from the sensor data created by the degradation.

In conducting the ARL check valve tests, accelerometer data were collected from both the pump (primary noise source) and choke valve (secondary noise source) to allow applying cross-correlation and coherence functions in analyzing accelerometer data from the check valve. Figures D-19 and D-20 show the incoherent spectra for different cases of disc wear and foreign objects at a pressure of 9 bars. The plotted results indicate that the multiple coherence method provides one of the possible data processing procedures to differentiate check valve response of normal and faulty conditions, and may also be able to characterize different types of fault.

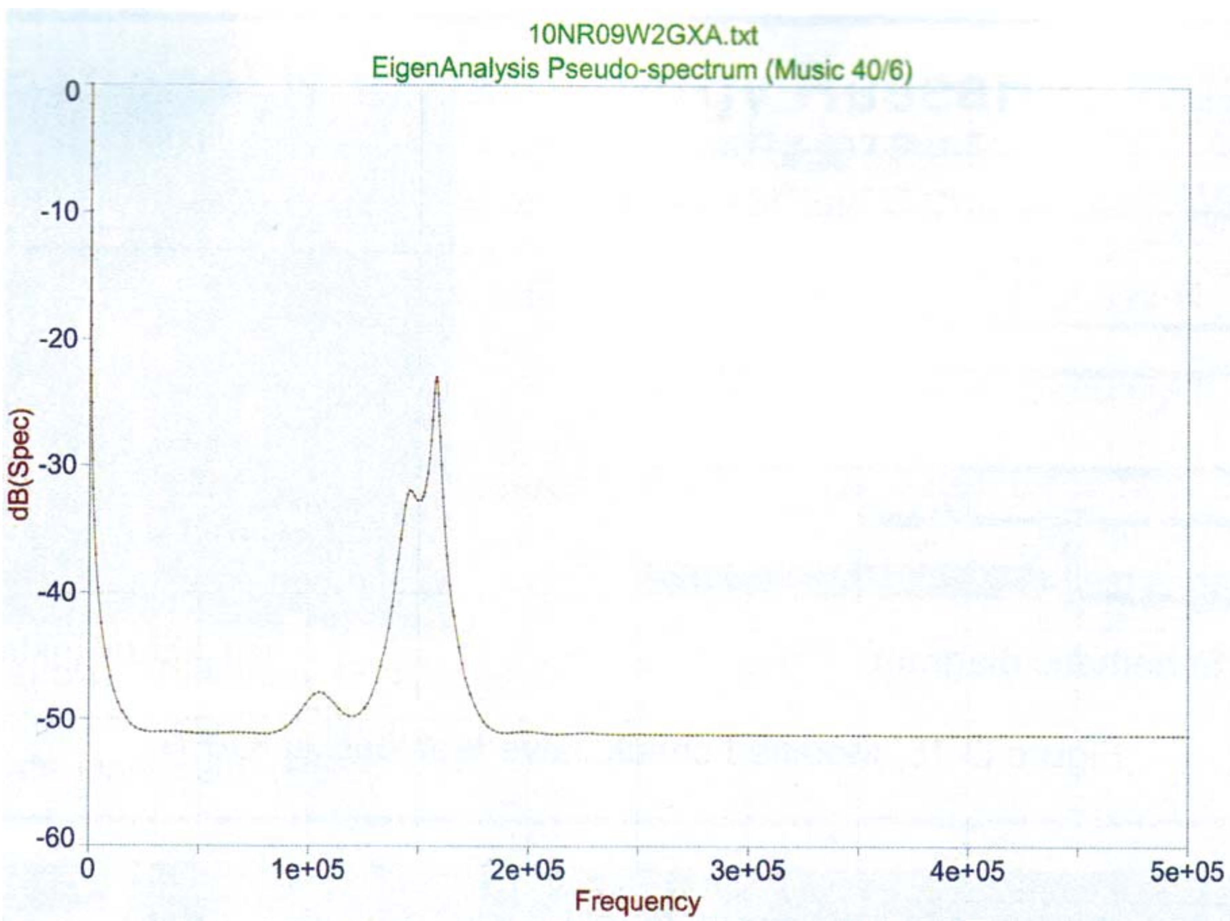


Figure D-17. A Normal Check Valve.

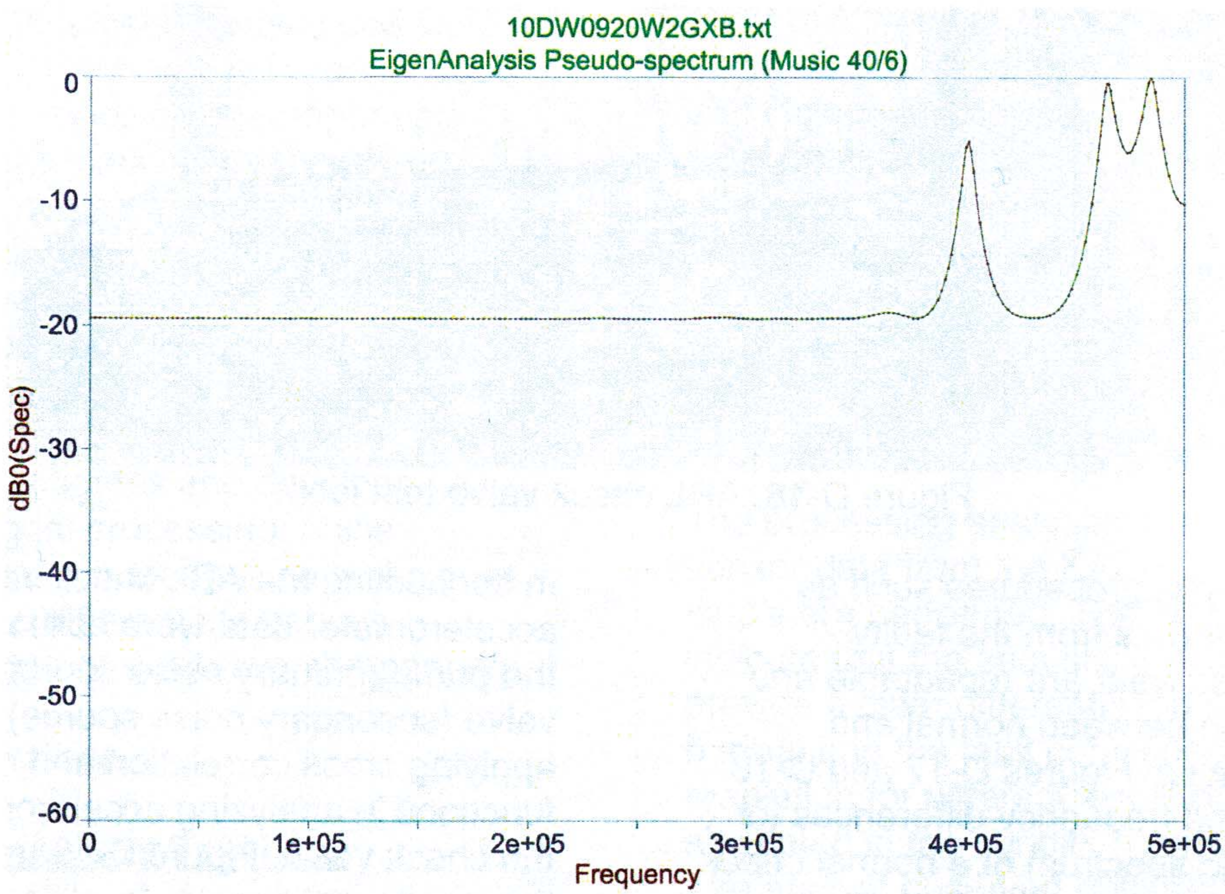


Figure D-18. A Degraded Check Valve with Disc Wear.

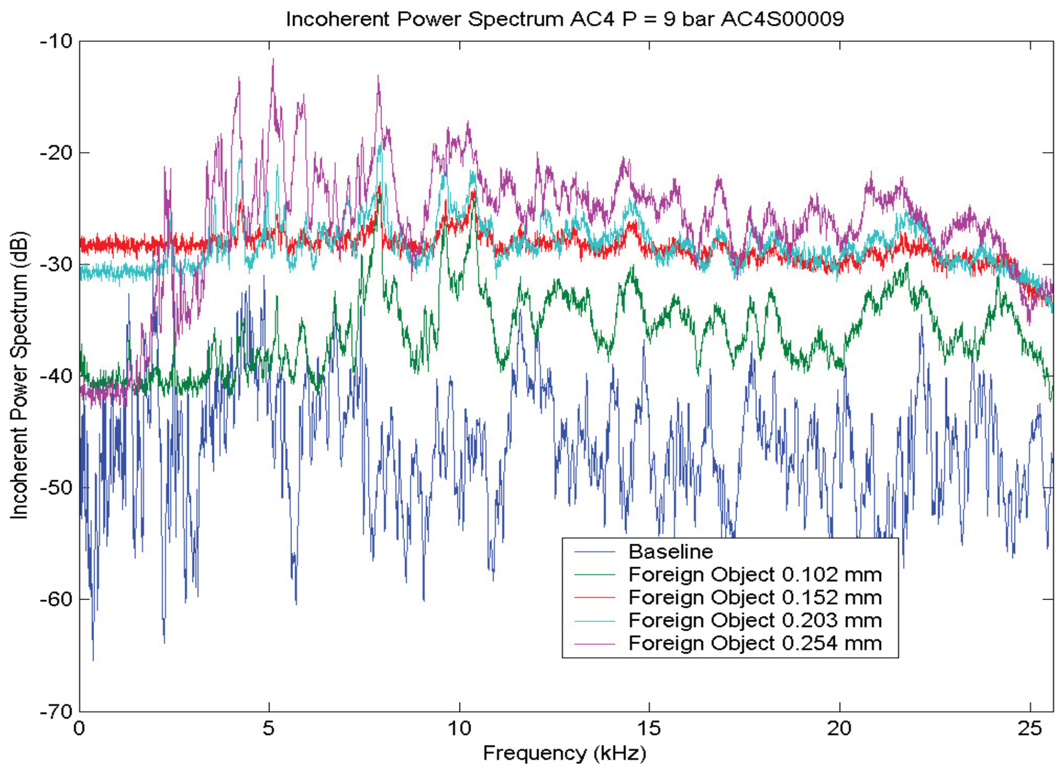


Figure D-19. Incoherent Spectra, Different Disc Wear.

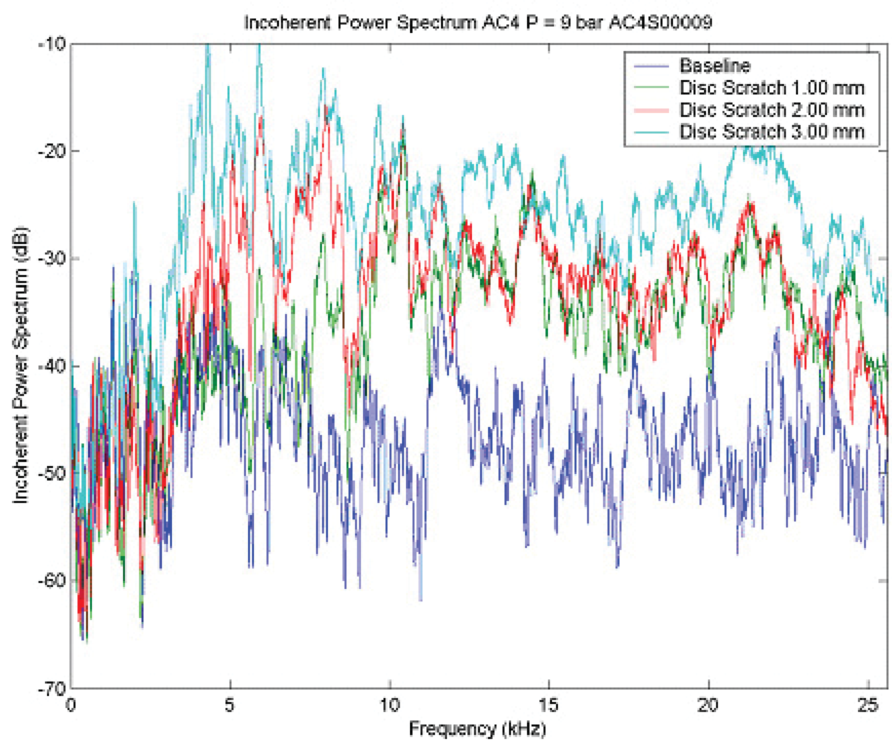


Figure D-20. Incoherent Spectra, Different Foreign Object.

Application of Artificial Neural Network

A trained neural network was applied to examine the sensor data from the check valve tests. The neural network results obtained from disc wear and foreign object failure modes indicate that (1) the neural network algorithm can clearly distinguish between the normal and failure modes; (2) it can also differentiate between disc wear and foreign object failure modes; and (3) it is very difficult to estimate the level of degradation. The results indicated that the maximum error of disc wear failure mode is 26% and errors in most cases are below 5%. In the case of the foreign object failure mode, the maximum error is about 27% and errors in most cases are below 10%. Therefore, the neural network algorithm is suitable for classifying the failure mode in check valves.

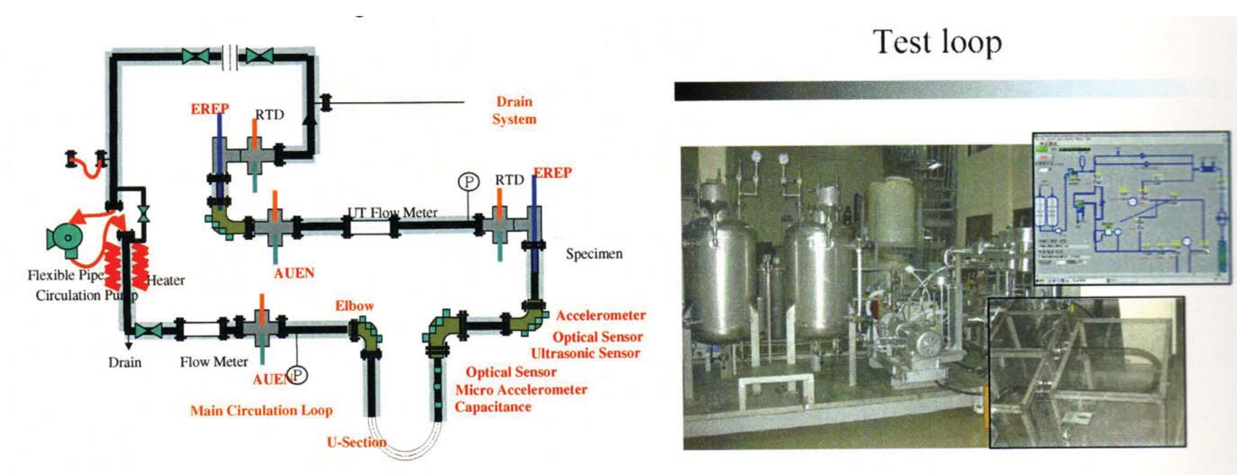
The flow-accelerated corrosion (FAC) test loop at SNU

The FAC test loop was installed at SNU to conduct condition monitoring tests of a secondary piping elbow in a non-safety related environment as a passive component. This test series addresses the degradation of piping systems subjected to corrosion/erosion attacks in a hostile environment of high temperature and pressure and undesirable water chemistry. This test series involves investigating the degradation behavior of the piping elbow in an accelerated corrosion/erosion environment, monitored by AUEN sensor (gold-coated electrode with metal-ceramic brazing seal). A schematic diagram of the piping elbow test loop and its actual installation are shown in Figure D-21.

The electrochemical corrosion potential (ECP) and pH electrodes in the FAC test performed as expected by demonstrating consistent behavior. Figure D-22 shows the responses from these electrodes during the chemical tests.

Planned Activities

To bring this joint research project to a successful completion, the project team plans to prepare the final report to document all findings from the two-condition monitoring test series of check valve and piping elbow. The findings include test data from all sensors, data processing analysis results, and finite element and hydrodynamic analysis results.



(a) Schematic diagram

(b) Actual installation

Figure D-21. Piping Elbow Test Loop.

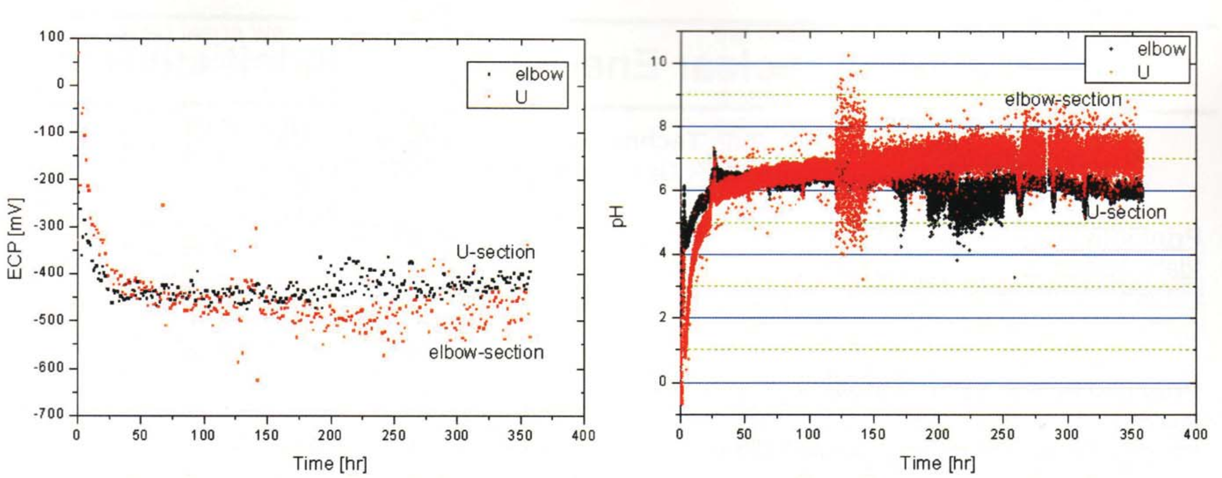


Figure D-22. Monitored ECP and pH Electrodes During the Chemical Test.

International Nuclear Energy Research Initiative

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

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

Principal Investigator (Korea): S. Oh, Korean Electric Power Research Institute (KEPRI)

Collaborators: Westinghouse

Project Number: 2002-022-K (I)

Project Start Date: January 15, 2002

Project End Date: January 15, 2004

Research Objective

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

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

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

Research Progress

UCSB was in charge of Base-Technology tasks and the overall coordination of the project, while Westinghouse and Korea Hydro and Nuclear Power (KHNP) were in charge of implementation tasks for the AP1000 and APR1400, respectively.

On the Base-Technology tasks, UCSB had been occupied principally with the Coolability (Critical Heat Flux [CHF]) issue, and, in particular, searching for mechanisms of CHF enhancement under IVR prototypic conditions. The Thermal Loading (Natural Convection Heat Transfer) issue was also addressed during the second year of the project.

On the Coolability task, testing in the modified (for AP1000 geometric features) full-scale ULPU-2400 Configuration V facility (Figure D-23), and in the small-scale BETA facility, produced a number of important results that were used by Westinghouse to support their IVR case in the AP1000 certification by the U.S. Nuclear Regulatory Commission. These included:

- Demonstrated that coatings normally used (and left on) for protection during shipping of the Reactor Pressure Vessel (RPV) can degrade burnout performance; recommended that they be removed. This was accepted by Westinghouse and it defines the base material for the burnout experiments as RPV grade steel.
- Defined a thermal insulation design that enhances (via streamlining the flow) the limits to coolability. The streamlined geometry, such as employed in ULPU Configuration V, is shown to significantly increase the coolability margins previously defined in ULPU (~1.5 MW/m^2) in connection with the AP600 certification. In the upper region, a critical heat flux of 1.8 to 2.0 MW/m^2 (an increase by ~ 20%) is an appropriate estimate of the expected AP1000 performance, including plant-specific inlet and exit geometries (Figure D-24).
- Obtained important basic understanding on the relative role of hydrodynamics and heater surface nanostructure (including the effects of water chemistry). As a result, researchers are able to connect the ULPU (copper material) performance to reactor (RPV grade steel) and bolster the reliability of the empirical results from ULPU. In particular, we found that a totally clean (de-ionized) coolant is able to promote such a degree of "cleansing" of the heater surface as to have a rather significant deleterious effect on the CHF. At the other extreme, the presence of TSP (tri-sodium-phosphate), a typical dissolved substance in reactor cavity water (an alkaline solution), has an outstanding beneficial effect on CHF. The presence of boric acid, also a normal ingredient here, somewhat diminished this enhancement, but the CHF is still considerably higher than that with pure water. The term "aging" is used to describe the aggregate of these not-yet-fully-understood, molecular-scale phenomena. An aged copper surface such as that employed in ULPU exhibits a similar coolability performance as the bare external surface of the RPV steel.
- Defined boiling/condensation loads for use by Westinghouse in the structural design of the reflecting thermal insulation. Under representative AP1000 exit geometry at the RPV nozzle gallery, the natural circulation flow is dominantly subcooled, modulated by periodic flashing and sweepout events, and associated pressure loss phenomena at the exit.

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

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

On the thermal-loading task, CFD simulations of the natural convection startup process in a fluid layer upon a sudden application of a thermal boundary condition were performed. Numerical results revealed a regular pattern of instability that evolves into natural convection loops. Most interestingly, natural convection experiments, BETA-NC, were performed on nanofilm heaters that allow transient pattern identification through IR thermometry. In the BETA-NC experiments, natural convection develops in response to the application of a constant heat-flux boundary condition on the fluid-layer's bottom surface. Temperature of the bottom surface is measured by a high-speed, high-resolution infrared camera (Figure D-25). Results of the 2-D and 3-D numerical simulations were compared with first-of-a-kind thermal patterns of natural convection heat transfer from the infrared thermometry. The BETA-NC data are essential in evaluating the capability of existing numerical schemes to capture such pattern formation in natural convection. The numerical simulation capability was also applied to examine natural convection heat transfer in metal-rich spherical-segment fluid layer stratified in the bottom layer of the pressure vessel lower head. The effects of Prandtl number (metallic layer separated at the bottom) and segment geometry (as compared to a fluid layer at the top of the oxidic pool) were delineated.

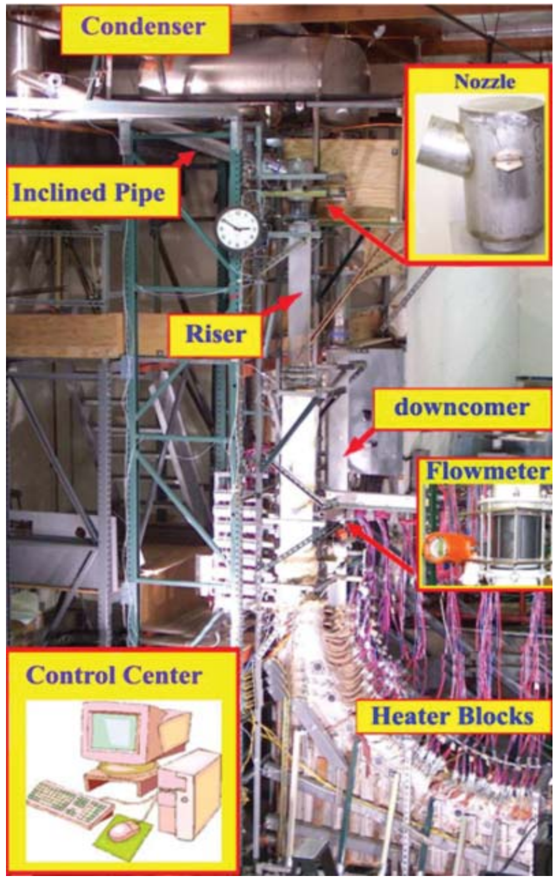


Figure D-23. AP1000-Related ULPU-2400 Configuration V Facility.

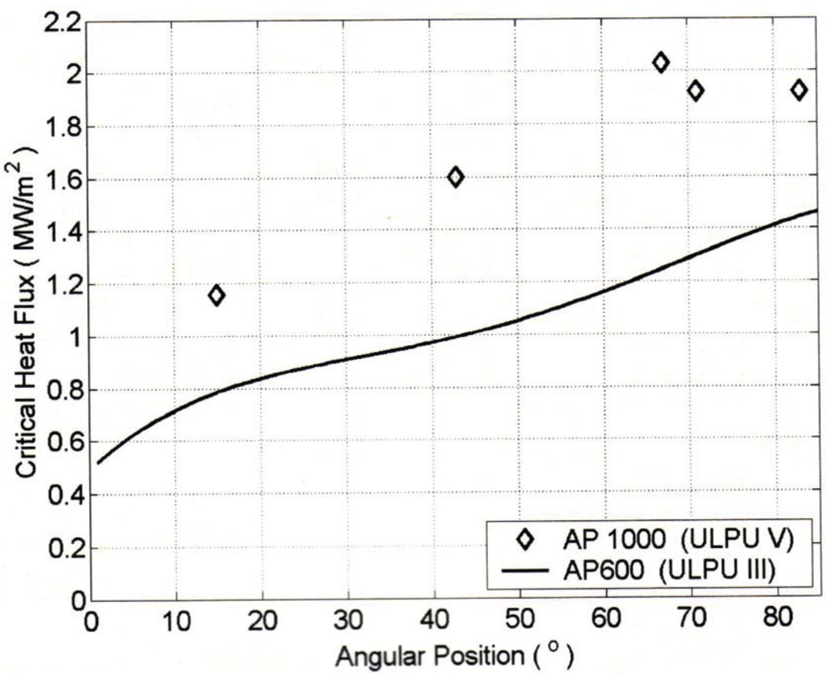


Figure D-24. Limits of Coolability in AP1000-Related ULPU-V.

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

The first potential failure mode is from high heat fluxes generated by a bottom metal layer with a significant fraction of the fission products partitioned into the metal. The lower bound critical heat flux at the bottom point of the reactor vessel lower head for the AP1000 geometry is greater than 800 kW/m^2. The upper bound of the heat flux from the bottom metal layer to the vessel wall is predicted to be less than 500 kW/m^2. Therefore, the heat flux from a bottom metal pool is not expected to exceed the critical heat flux at the bottom of the reactor vessel lower head. The ratio of the maximum heat flux to the minimum critical heat flux q/q_{CHF}, is 0.625. The second potential failure mode is from a thinned metal layer producing high heat fluxes via the focusing effect at the top. For a bounding case of top metal layer thinning due to material interactions, the peak heat flux to the RPV wall is 1720 kW/m^3. The lower bound critical heat flux in this region (angular position ~ 90^o) is 1890 kW/m^3. The q/q_{CHF} is 0.91, demonstrating that margin-to-failure is maintained for this bounding case. The assumptions in this analysis are conservative with respect to increasing the heat loading to the vessel.

The main effort by KHNP was constructing a natural circulation loop based on APR1400 design and conducting the tests. Both air-water and steam-water tests were conducted. Oscillatory flow behavior was observed on some steam-water test runs. The regime map depicting the oscillatory flow was derived. In collaboration with KHNP, researchers at KAIST performed forced flow CHF tests and developed CHF correlations. To perform an integrated IVR performance evaluation for APR1400, four representative scenarios for APR1400 were examined using MAAP. Evaluation was also made using SCDAP code. The integrated IVR performance was evaluated for APR1400 in support of the implementation of in-vessel retention in the APR1400 plant.

Planned Activities

All the tasks of the project were completed in January 2004.

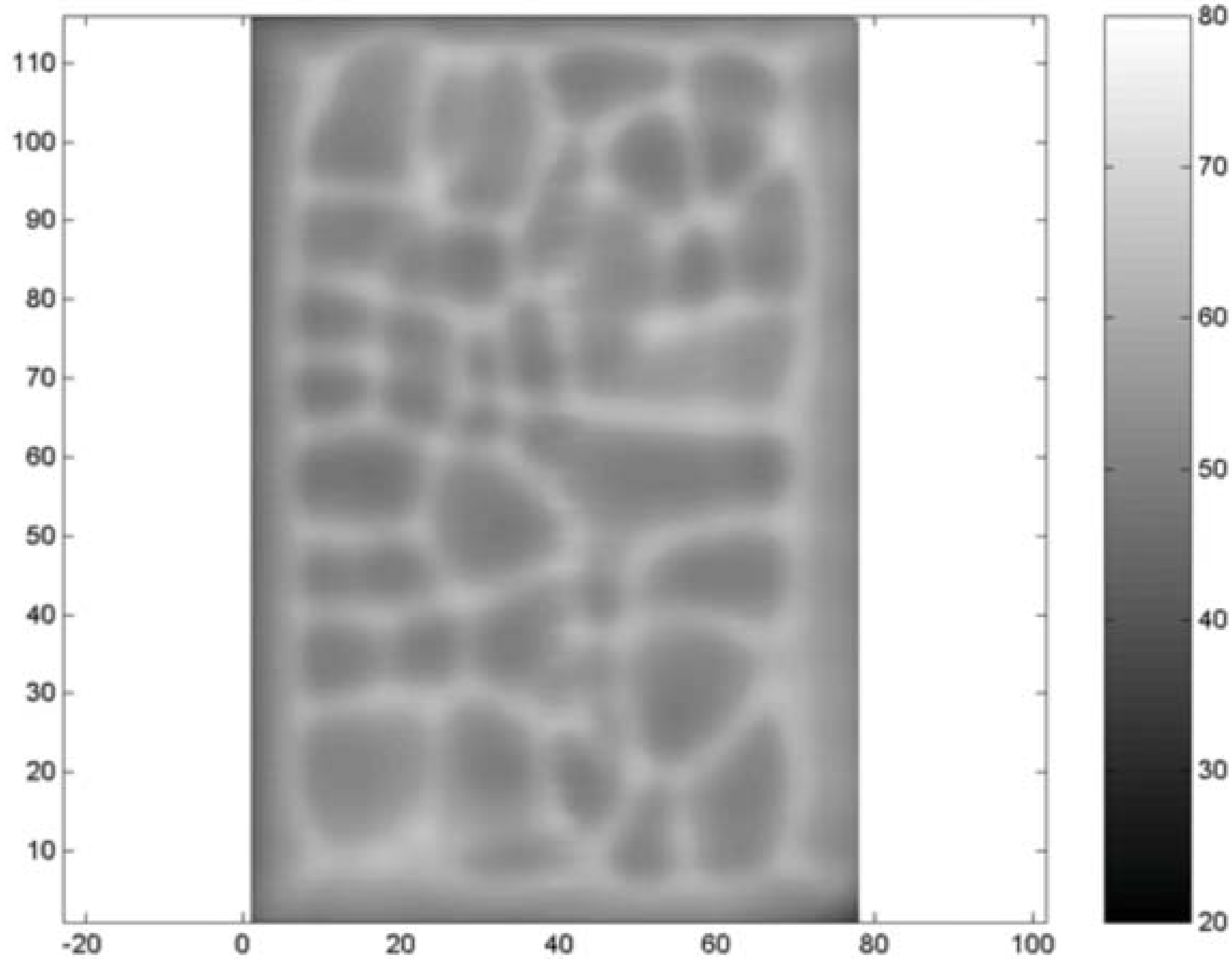


Figure D-25. Infrared Thermometry of Nano-Film Heater Under Natural Convection.

International Nuclear Energy Research Initiative

In-Vessel Retention Strategies for High Power Reactors

Principal Investigator (U.S.): J. Rempe, Idaho National Laboratory (INL)

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

Collaborators: Pennsylvania State University (PSU), Korea Atomic Energy Research Institute (KAERI)

Project Number: 2002-022-K (II)

Project Start Date: January 1, 2002

Project End Date: December 31, 2004

Research Objective

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

Research Progress

This three-year project includes four tasks. In Task 1, which was completed during the first year of this research program, SCDAP/RELAP5-3D© calculations were conducted to define representative bounding late-phase melt conditions. Characteristic parameters from those bounding conditions (thermal loads, pressure, relocated mass, etc.) are used to design an optimized core catcher (in Task 2) and ERVC enhancements (in Task 3). Task 2 and 3 activities, which were initiated in the first year of this project, were completed during the third year of this project. In Task 4, collaborators assess the improved margin obtained with Task 2 and 3 design modifications. Margins are presented such that the impact of these modifications can easily be applied to other reactor designs. As indicated in Figure D-26, key facilities and capabilities of each collaborator were used to complete these tasks. Key tasks completed during this reporting period are highlighted below.

- **Completed Task 2 efforts to quantify the increased margin offered by an in-vessel core catcher.** These efforts help quantify the reduced heat loads to the reactor vessel if the proposed in-vessel core catcher is inserted into the reactor vessel. During this reporting period, several efforts were completed to optimize the proposed core catcher design and provide insights about its performance.
  - KAERI completed four in-vessel core catcher (IVCC) LAVA-GAP tests (LAVA-GAP-4 through LAVA-GAP-7). The LAVA-GAP tests, which use Al_2O_3 to simulate materials relocating from the core, provide insights about the impact of an in-vessel core catcher and candidate coatings on vessel thermal response. Post-test examinations and test instrumentation show that the presence of an insulator coating significantly reduces the thermal loads and attack from relocating materials.

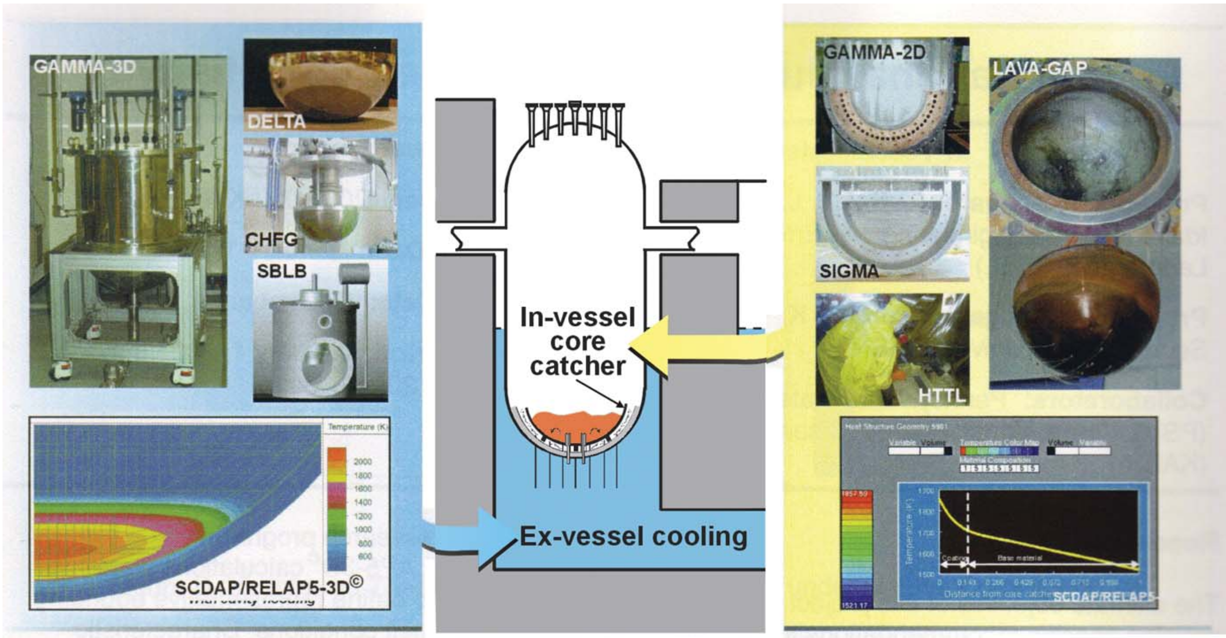


Figure D-26. Key U.S. and Korean experimental facilities and state-of-the-art analytical tools are applied to investigate options that could enhance external reactor vessel cooling and internal core catcher performance.

- SNU conducted SIGMA 2-D and 3-D tests for a circular pool and a spherical pool, respectively.
- SNU conducted GAMMA 1-D and 2-D tests for a rectangular channel and a circular channel, respectively, to account for the effects of test specimen length and gap size on CHF, respectively. Using test results, SNU developed coefficients for a narrow gap cooling model.
- INL completed high-temperature materials interaction tests with plasma-sprayed samples. Results suggest that the core catcher should consist of a stainless steel base material that is thermally sprayed with an Inconel-178 bond coat underneath a ZrO_2 oxide coating.
- INL completed tests investigating the potential for materials interactions between proposed core catcher materials and prototypic materials expected to relocate during a severe accident. Test results from the P-1 test, which was conducted in an argon atmosphere, and the P-2 test, which was conducted in a steam atmosphere, both indicate that the proposed core catcher coatings protected the base material and did not experience any materials interactions with prototypic materials.

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

  - PSU completed transient quenching and steady-state boiling experiments in the SBLB facility for vessels coated with micro-porous coatings, vessels with an enhanced insulation design, and coated vessels with an enhanced insulation design. Results indicate that these enhancements significantly enhance the CHF for boiling on downward facing curved surfaces. PSU finalized CHF correlations showing the separate and combined effects of vessel coatings and enhanced vessel/insulation on the local CHF limits.

  - SNU continued the DELTA 1-D tests to quantify parameters for film boiling analysis with interfacial wavy motion.

  - SNU completed installation of and started conducting tests in the GAMMA 3-D facility. These test results will provide complementary data to results from the SBLB facility.

  - KAERI conducted tests in the HERMES-HALF facility, which is a half-height, half-sector model for evaluating two-phase natural circulation phenomena through the gap between the vessel and insulation, to provide recommendations for the APR1400 insulation, and to correlate the re-circulation flow rate.

- Initiated Task 4 efforts to evaluate the improved margin associated with proposed IVR modifications. These efforts use the bounding endstate from Task 1 to assess the increase in margin for IVR.

  - INL initiated SCDAP/RELAP5-3D© and VESTA analyses efforts to evaluate the impact of various features proposed in this project to increase the IVR margin. INL completed a SCDAP/RELAP5-3D© base case analysis using a recently constructed COUPLE mesh for predicting temperatures in the reactor vessel lower head and in materials that relocate to the lower head. INL compiled VESTA and SCDAP/RELAP5-3D© calculation input requirements and prioritized analysis cases.

  - KAERI initiated SCDAP/RELAP5-3D© and LILAC analyses to implement the effect of the in-vessel core catcher as a measure for the IVR enhancement. KAERI initiated SCDAP/RELAP5-3D© calculations for a Large Break LOCA in the APR1400.

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

Planned Activities

During the remainder of the final year of this project, calculations to assess the increased margin offered by proposed IVR enhancements will be completed. In addition, collaborators will prepare the final report for this project.

International Nuclear Energy Research Initiative

Passive Safety Optimization in Liquid Sodium-Cooled Reactors

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

Principal Investigator (Korea): D. Hahn, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2003-002-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

Research Objective

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

Research Progress

Task 1 provides for development, implementation, and testing of a detailed three-dimensional fuel subassembly thermal-hydraulic model. Detailed modeling is necessary for the accurate quantification of fuel, cladding, coolant, and structure temperatures in passive safety transient analysis, and enables evaluation and selection of reactor design features that promote safe and reliable operation and eliminate core damage in accidents. The new model is specifically designed for computational efficiency and easy integration into the ANL SAS4A/SASSYS-1 computer code and the KAERI SSC-K computer code. This integration feature is necessary for coupling to reactor kinetics, reactivity feedback, and structural mechanics models that simulate reactor design performance. In 2004, ANL and KAERI researchers completed development of the first version of the model. Figure D-27 shows the calculated temperature field in a vertical plane of a subassembly pin bundle, taking into account both axial and lateral mass and heat transfer. In the final year of the project, the model will be validated with experimental test data and used to provide high fidelity temperature predictions for advanced reactor safety simulations.

In Task 2, the advanced modeling capability developed in Task 1 and state-of-the-art modeling capabilities are used to perform integrated safety assessments of passive safety design features that promote safe and reliable operation and eliminate core damage in accidents. The conceptual design of a prototypic metallic-fueled, sodium-cooled reactor plant serves as the framework for computational investigations that quantify the relative safety margins provided by specific advanced design features. Figure D-28 summarizes peak temperatures calculated with SAS4A/SASSYS-1 and SSC-K in 2004 for baseline studies of unprotected transient overpower (UTOP), loss-of-flow (ULOF), and loss-of-heat-sink (ULOHS) accident sequences in the KALIMER-150 reactor design. Overall agreement between the U.S. and Korean codes is very good, with some differences in calculated fuel temperatures due to modeling assumptions. These baseline results show the temperature safety margins for (1) the peak fuel temperature, (2) the peak cladding temperature, (3) the peak coolant temperature, and (4) the peak core outlet temperature in relation to American Society of Mechanical Engineers (ASME) Service Level D limits for KALIMER-150. Future work will include advanced analyses of enhanced design features for comparison to the baseline results.

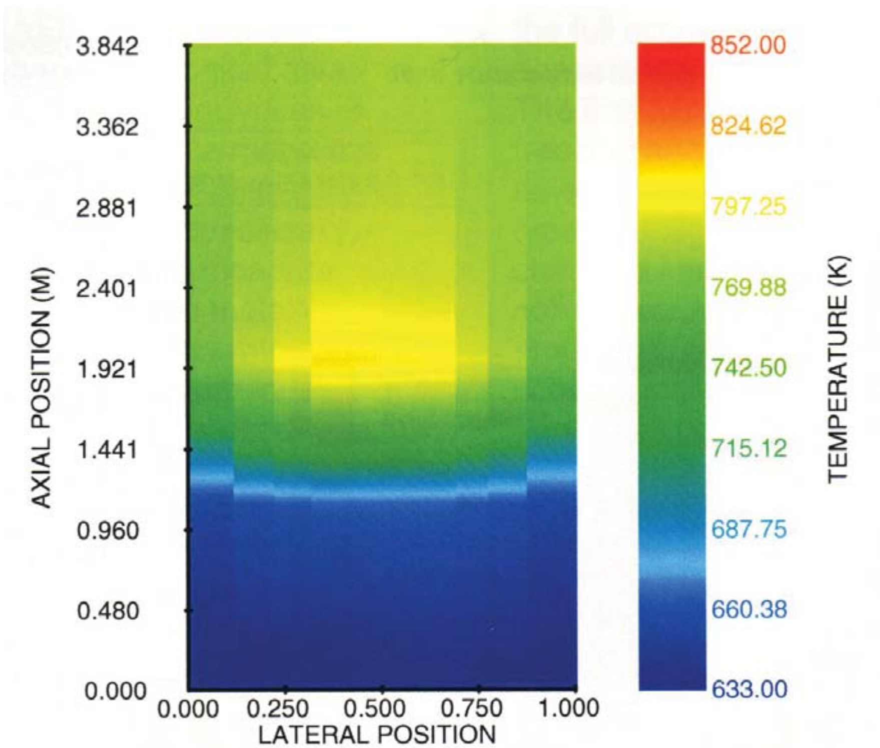


Figure D-27. Task 1 Model Results.

The safety and economics implications of coupling a super-critical carbon dioxide (S-CO_2) power cycle to a sodium-cooled fast reactor are investigated in Task 3. The advantages of the S-CO_2 Brayton cycle include significantly improved cycle efficiency relative to a Rankine steam cycle; reduced plant footprint due to fewer, simpler, and smaller-sized components; and reduced capital and operating costs and plant staffing requirements from plant simplification and elimination of costly Rankine cycle components. Completed studies in Task 3 include analyses to maximize plant efficiency by development of optimized component designs, particularly heat exchanger designs. Figure D-29 shows the results of a cycle efficiency analysis for a system design that employs high-performance heat exchangers for both the intermediate sodium heat transfer in the reactor vessel and also the sodium-to-CO_2 heat exchangers. Future activities will include development of a control strategy for the coupled reactor/S-CO_2 Brayton cycle plant, identification of a set of potential accidents, and assessment of needs for system or equipment modifications to assure and improve safety.

In Task 4, ANL and KAERI develop test plans for measurement of phenomenological data required for quantifying the consequences of beyond-design-basis accidents in a metallic-fueled, sodium cooled fast reactor coupled to a super-critical CO_2 Brayton cycle. Specifically, test plans are being developed to measure data describing freezing of molten metallic fuel, melt relocation and interaction with steel structure, and intermixing of high-pressure CO_2 with sodium. In 2004, work focused on a test plan to measure data on freezing and plugging of molten metallic fuel in structures within the subassembly above and below the core, and in inter-subassembly gaps. To cover the full scope of freezing and plugging phenomena, two types of tests are needed. The first type is concerned with the transient freezing and plugging behavior of molten fuel flowing in coolant channels, and the second type investigates aspects of the inter-metallic chemical interaction between molten fuel and solid steel. A single test apparatus will be employed for the two types of tests. As shown schematically in Figure D-30, the overall test apparatus consists of a fuel melt furnace vessel, melt delivery system, test section, catch pan and containment vessel. The test section for the chemical interaction test is shown in Figure D-31.

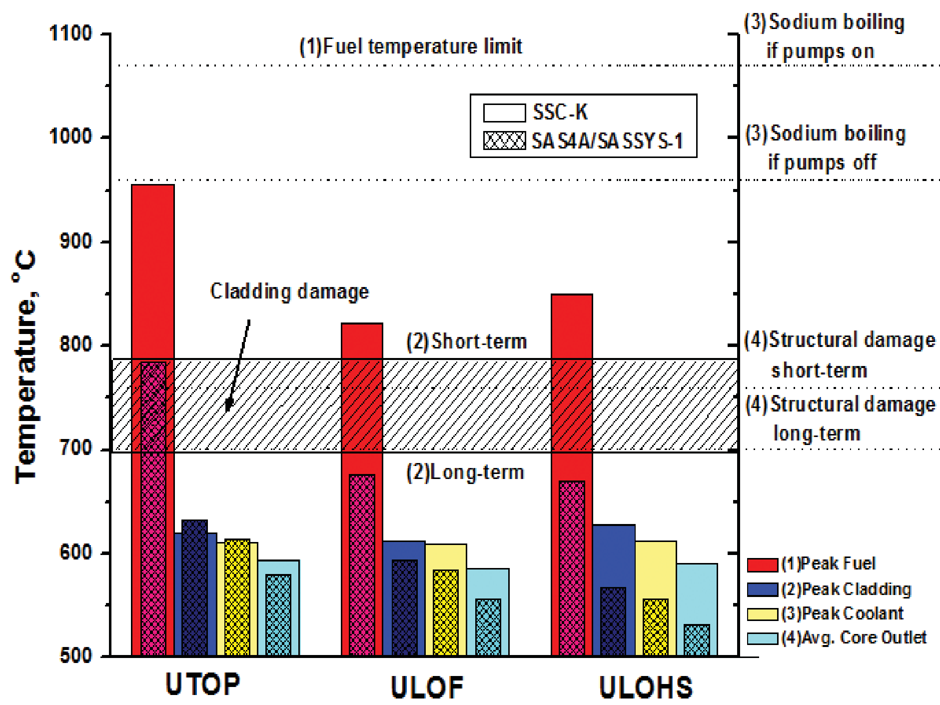


Figure D-28. Task 2 Analysis Results.

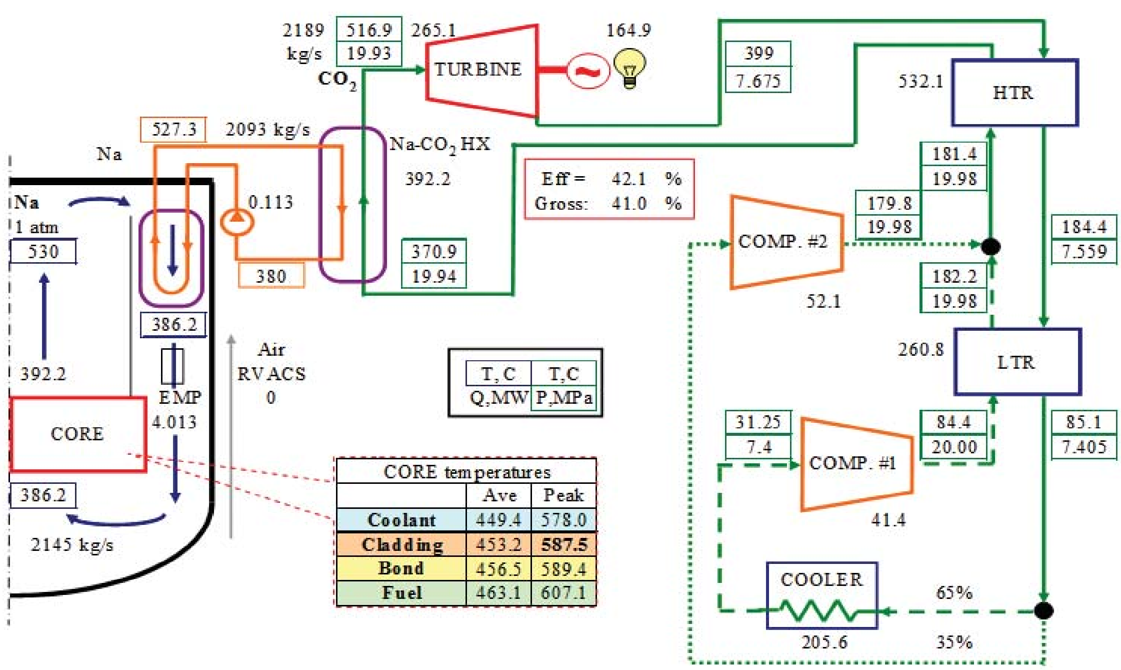


Figure D-29. Task 3 Super-critical CO_2 Power Cycle Analysis Results.

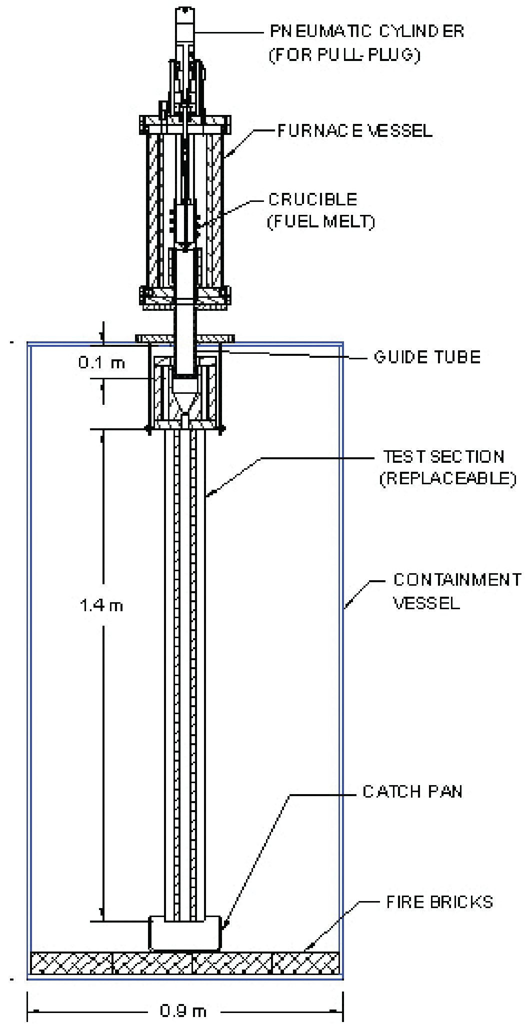


Figure D-30. Task 4 Test Apparatus.

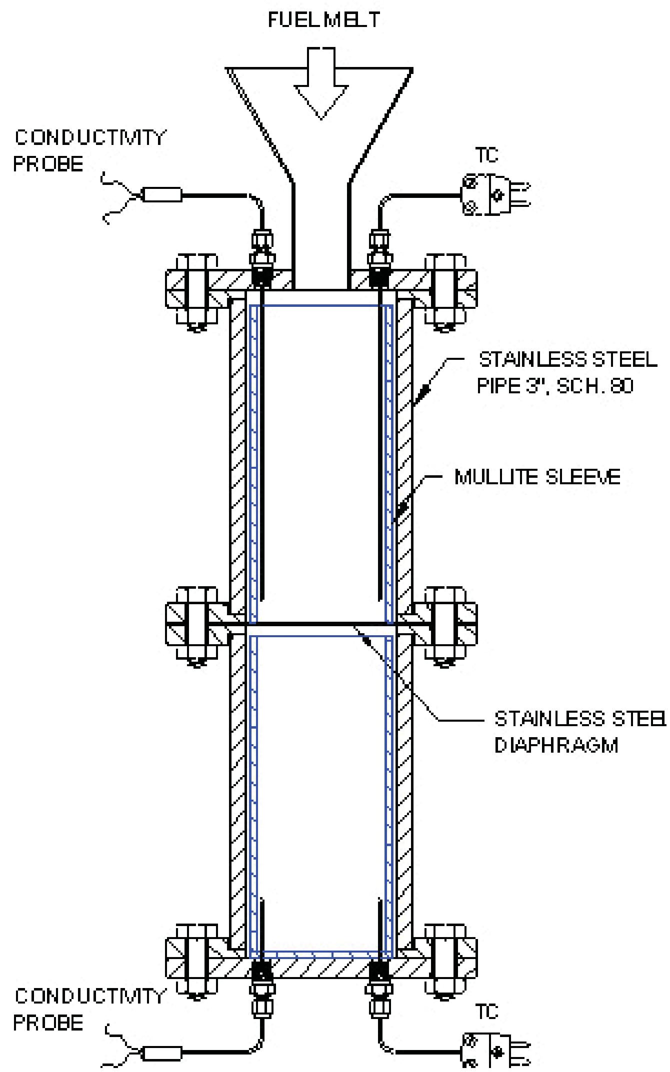


Figure D-31. Chemical Interaction Test Section.

Planned Activities

In 2005, work will progress as planned in all four tasks of the project. In Task 1, model testing will be completed, and model documentation, including an input description and user's guide, will be supplied. The analyses in Task 2 to characterize the performance potential of advanced passive safety design features will be completed, utilizing the Task 1 modeling, with comparison to the baseline results. In Task 3, investigations of transient safety performance of the coupled reactor/Brayton cycle plant will be evaluated and optimized control strategies will be developed. Finally, in Task 4, a test plan for the evaluation of the consequences of blowdown and intermixing of CO_2 in a sodium pool will be completed.

International Nuclear Energy Research Initiative

Developing and Evaluating Candidate Materials for Generation IV Super-Critical Water-Cooled Reactors

Principal Investigator (U.S.): J. Cole, Idaho National Laboratory (INL)

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

Collaborators: Korea Advanced Institute of Science and Technology (KAIST), University of Michigan, University of Wisconsin

Project Number: 2003-008-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

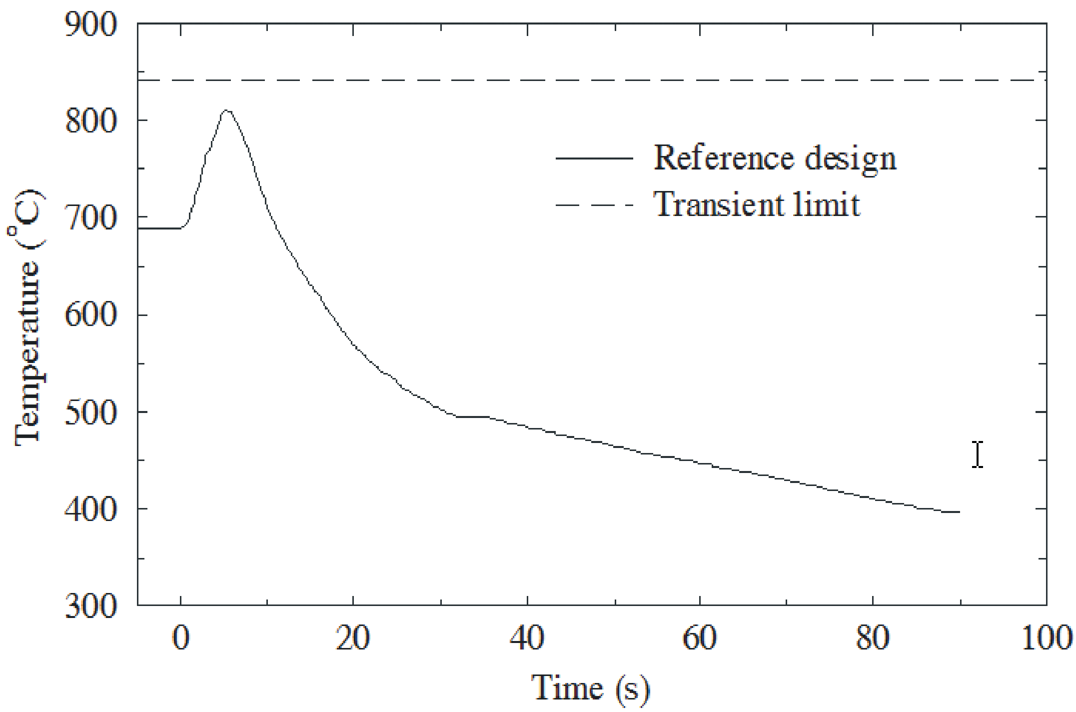
Research Objective

The Generation IV Super-Critical Water-Cooled Reactor (Generation IV SCWR) is being proposed as an advanced high efficiency thermal reactor for baseload electricity production. One of the major unknowns with this reactor concept is the behavior of fuel cladding and structural components under the extremely aggressive SCWR environment. The objective of this project is to evaluate candidate materials for SCWR application. The work includes efforts to evaluate candidate alloys in terms of high temperature mechanical properties, corrosion and stress corrosion cracking, radiation stability, and weldability. Two major outcomes of the project are the production of information that can be used ultimately by SCWR systems designers, and the preparation of recommendations for a further course of investigations involving in-reactor irradiation experiments.

Research Progress

The project has made progress in several key areas over the last year, including the completion of thermal transient studies on two ferritic-martensitic (F-M) alloys, evaluation of the microstructure and mechanical properties of two commercially available ODS alloys, corrosion and stress corrosion cracking tests of F-M candidate alloys at three separate institutions, and charged particle irradiations to investigate candidate alloy irradiation stability. The following summarizes some specific details of these results.

In a study conducted at INL, two F-M alloys were exposed to thermal transients to explore the impact potential reactor thermal transients might have on the structure and mechanical properties. Exposure of alloys T91 (a 9 Cr alloy) and HCM12A (a 12 Cr alloy) was performed using a Gleeble thermal simulator. Five and ten cycle tests were performed to 810°C (reference transient) and 845°C (limit transient). Because the maximum transient temperature lies near or above the equilibrium austenite transformation temperature for these steels, it has the potential to severely alter their microstructure and properties, either by formation of new untempered martensite or overtempering of the existing structure. Results revealed that thermal cycling under these conditions had negligible effect on microstructure and properties. Interestingly, a reduction in the creep rate of the 9 Cr alloy was observed following exposure to the thermal transient. A plot of the thermal transient employed along with a plot of the creep rupture behavior of the 9 Cr alloy is provided in Figure D-32. This initial study indicates properties would be adequate for the conditions examined. However, further investigations are necessary to establish the bounding transient conditions which don't result in deleterious changes.



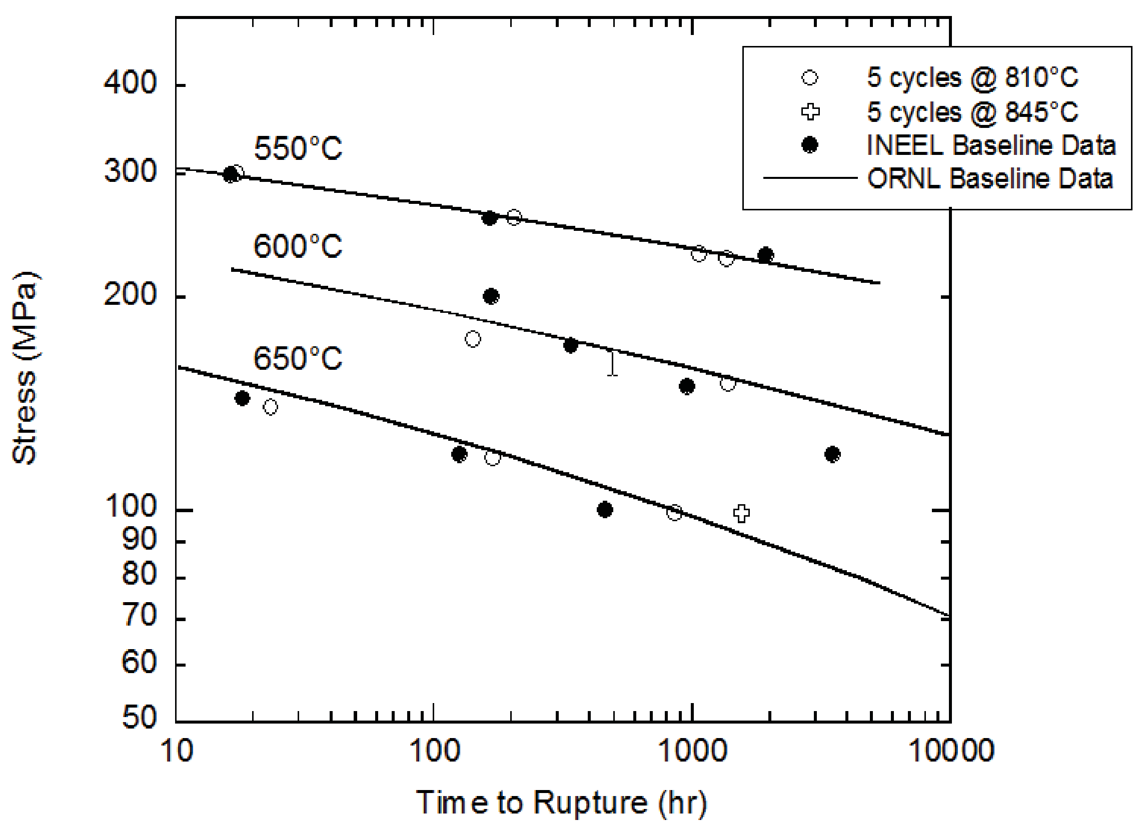


Figure D-32(b). Rupture Lives of Thermal Cycled and Baseline 9Cr-1MoVNb.

Commercially available oxide dispersion strengthened alloys MA956 and PM2000 were evaluated at KAIST in terms of microstructure and mechanical properties for possible use in Super-Critical Water-Cooled Reactor applications. These alloys have the potential to allow increased reactor coolant temperatures due to their superior creep strength compared to conventional F-M alloys. Results from an investigation of thermal aging of these alloys revealed that these higher chromium variants of the ODS alloys can be susceptible to embrittlement after exposure in the temperature range of 400 to 550°C for extended periods of time.

The project has obtained a substantial amount of corrosion and stress corrosion cracking data (SCC) over the past year. Three separate super-critical water corrosion and 2 separate SCC facilities are being utilized for evaluating and qualifying the candidate alloys. The focus to date has been on testing candidate F-M alloys to complement corrosion studies being conducted as part of the Generation IV SCWR initiative investigating a broader range of alloys, including Ni-based and austenitic stainless steel alloys.

Work at the University of Michigan has focused on corrosion and SCC tests of alloys HT9 (for baseline comparison), T91, and HCM12A. Tests have been conducted in 500°C super-critical water containing two different oxygen concentrations. The first test was conducted in deaerated super-critical water while a second was conducted in super-critical water containing approximately 100 ppb oxygen. All of the SCC samples tested failed in a ductile manner, as shown in Figure D-33, indicating the absence of enhanced cracking susceptibility. Overall, alloy HCM12A (12% Cr) oxidized less than T91 (9% Cr) in all environments. In addition, samples exposed to 100 ppb oxygen had a lower rate of oxidation than in the deaerated case. The reason for the lower oxidation rate in the higher oxygen content super-critical water is believed to be the formation of a hematite (Fe_2O_3) layer in addition to magnetite (Fe_3O_4) that promotes a dense, less permeable surface oxide, thereby reducing corrosion. The results of these experiments indicate that, as with currently operating Light-Water Reactors and fossil plants, controlling water chemistry can be critical to minimizing internals degradation.

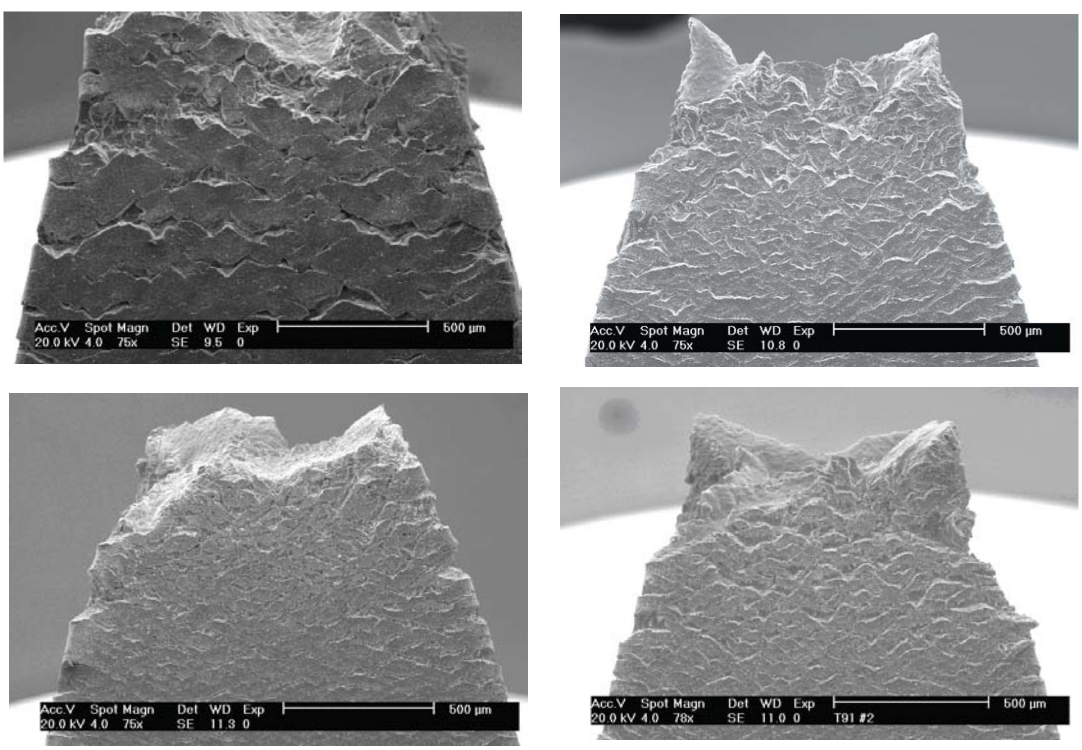


Figure D-33. SEM Images of the Gage Section Surfaces of
(a) HT-9, (b) HCM12A, (c) T91 #1, and (d) T91 #2.

In another series of studies, the same three alloys that underwent the SCC tests at the University of Michigan were tested in a flowing super-critical water loop at the University of Wisconsin. The samples were tested for one, two, and three weeks in order to evaluate corrosion kinetics. In addition, samples were surface modified with an oxygen plasma treatment and tested side-by-side with untreated samples. Results indicated that while the oxidation rate of the surface modified alloys was initially higher, the longer term oxidation rate was lower. This was true only for alloys HT9 and T91. The surface modified HCM12A sample did not exhibit a noticeable change in oxidation behavior compared to the unmodified sample. In addition to the surface modified corrosion coupons tested at the University of Wisconsin, SCC samples were fabricated and surface modified as illustrated in Figure D-34. These samples will be tested in the SCC loop at the University of Michigan.

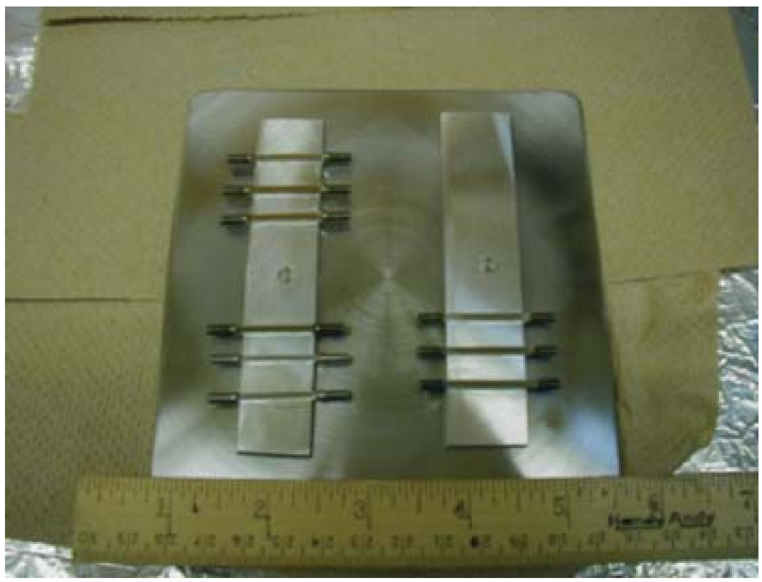


Figure D-34 (a). SCC samples of the three alloys mounted on aluminum stage to promote heat conductivity through the samples.

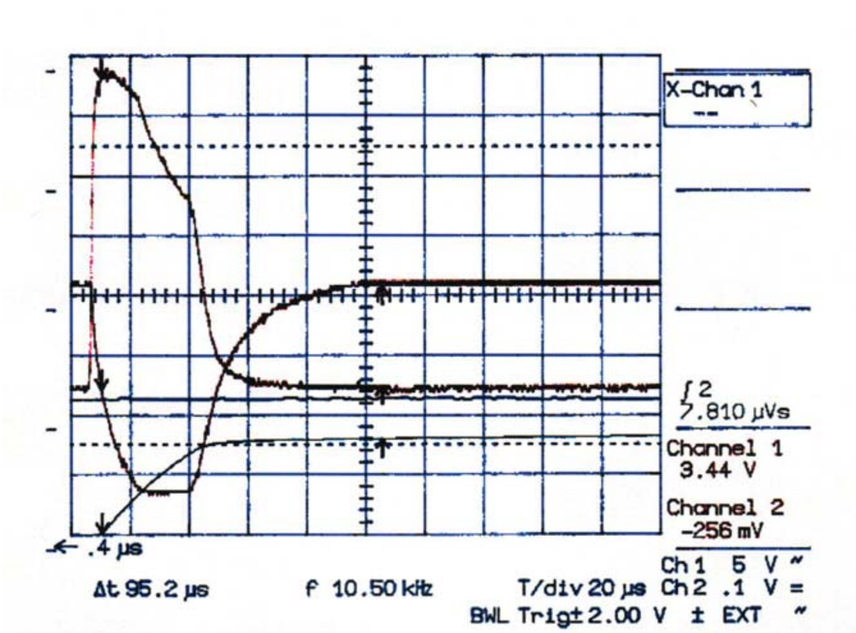


Figure D-34(b). Voltage and Current Variation During a Single Pulse During Oxygen Ion Implantation of the Samples.

The corrosion and SCC tests in the U.S. are being complemented by studies at KAERI in the Republic of Korea (ROK). In addition to the ferritic-martensitic alloys tested in the U.S., KAERI has conducted static corrosion tests on austenitic Fe and Ni-based alloys as well as commercially available oxide dispersion strengthened alloys. The tests were performed at temperatures ranging from 400–627°C for 100, 200, and 500 hours. The corrosion rate of the 9 Cr alloys increased noticeably above 550°C. Higher Cr alloys showed the most resistance to corrosion with a 20% Cr ODS alloy exhibiting the best behavior. The corrosion behavior of the high Cr austenitic alloys was significantly improved over the F-M alloys.

In a study conducted by ANL, the potential impact of in-core radiation damage on candidate alloys was evaluated. Alloys with a limited radiation damage database were selected for heavy-ion irradiation damage studies. The 12 Cr F-M alloy HCM12A and the austenitic alloy 800H were irradiated with heavy-ions (Ni) at 500°C to doses of 5 and 50 displacements per atom (dpa). The microstructures of the alloys were examined with TEM following irradiation. Neither alloy exhibited void formation up to a dose of 50 dpa. The radiation damage in the HCM12A alloy manifested itself as an increase in the density of network dislocations, while the radiation damage in the 800H alloy generated a high density of faulted dislocation loops. At the higher dose, a population of extremely small precipitates also formed in alloy 800H and the dislocation loop structure refined. The refinement of the dislocation structure is thought to be associated with the small precipitates acting as sinks for the point defects generated from the displacement damage.

Planned Activities

In the third year of the project, work will continue to further define the operational envelope of candidate core internal and cladding materials and begin to compile collected data into a concise format for distribution to SCWR systems designers. Corrosion and SCC tests will expand to further examine the effects of temperature and water chemistry on oxidation behavior. Tests will also be initiated to examine the impact of proton irradiation on corrosion and SCC behavior. In addition, the microstructures of the proton-irradiated samples will be characterized in an attempt to link the influence of microstructural changes to any changes observed in the SCC tests. The influence of the oxygen-ion surface implantation on SCC behavior will also be investigated to ensure SCC is not an issue with surface-modified samples. The final experimental task for the upcoming year will involve the fabrication of corrosion and SCC test coupons from welded samples and testing of these in super-critical water to evaluate the influence of microstructural changes induced by welding on the overall corrosion and SCC response. Based on the experimental investigations, recommendations for future in-reactor testing campaigns will be made.

International Nuclear Energy Research Initiative

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

Principal Investigator (U.S.): C. Oh, Idaho National Laboratory (INL)

Principal Investigator (Korea): H. Cheon No., Republic of Korea Advanced Institute of Science and Technology (KAIST)

Collaborators: Seoul National University, University of Michigan

Project Number: 2003-013-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

Research Objective

The Very-High-Temperature Gas-Cooled Reactors (VHTGRs) are those concepts that have average coolant temperatures above 900°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat application in addition to power generation and nuclear hydrogen generation. While all the HTGR concepts have sufficiently high temperatures to support process heat applications, such as desalination and cogeneration, the VHTGR's higher temperatures are suitable for particular applications such as thermochemical hydrogen production. However, the high temperature operation can be detrimental to safety following a LOCA initiated by pipe breaks caused by seismic or other events. Following the loss of coolant through the break and coolant depressurization, air from the containment will enter the core by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structures and fuel. The oxidation will release heat and accelerate the heatup of the reactor core.

Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. INL has investigated this event for the past three years for the HTGR. However, the computer codes used, and in fact none of the world's computer codes, have been sufficiently developed and validated to reliably predict this event. New code development, improvement of the existing codes and experimental validation are imperative to narrow the uncertainty in the predictions of this type of accident.

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

Research Progress

The collaborators for this research project are INL, KAIST, SNU, and the University of Michigan.

This project consists of six tasks for developing, improving, and validating computer codes for analysis of the VHGTR: (1) develop a computational fluid dynamics code for benchmarking, (2) perform an RCCS experiment, (3) perform an air ingress experiment, (4) improve the system analysis codes RELAP5/ATHENA and MELCOR, (5) develop an advanced neutronic model, and (6) verify and validate the computer codes. The primary activities and key accomplishments for each task are summarized below.

Task 1 – CFD thermal-hydraulic benchmark code development (KAIST). The researchers developed the multidimensional system analysis tool and validated it using various experiments: NACOK natural convection test, HTTR-simulated air ingress experiment, HTTR RCCS mockup test, and SANA afterheat removal test. First, they performed the simulation for the German NACOK test in order to check out the capability of the 1-D/3-D integrated GAMMA (Figure D-35) version, as well as to validate the natural convection behavior throughout the entire core during the second stage of an air ingress accident. Even with the assumption of uniform temperature condition, the predicted air or air-He mixture flows are consistent with the observed behavior at various temperature conditions. Second, the HTTR-simulated air ingress experiment which was conducted in Japan was analyzed by GAMMA using a combination of 1-D and 2-D components. The test cases are ten equal temperature conditions and five non-equal temperature conditions in graphite tubes. The predicted results are in agreement with the measured data, with about a 10% deviation for the onset time of natural convection. Third, the HTTR RCCS mockup tests, which were selected as benchmark problems in IAEA CRP for the analysis of afterheat removal under accident conditions, have been simulated by the GAMMA code. The temperatures of the pressure vessel (P.V.) are well predicted, demonstrating that GAMMA can be used to evaluate the hot spots on the P.V. and heat removal by thermal radiation and natural convection. Fourth, the SANA-1 self-acting afterheat removal tests, one of the IAEA benchmark problems, have been simulated to validate the porous media model that was incorporated into GAMMA. It was shown that the GAMMA code has comparable predictability with other codes (TINTE, THERMIX, TRIO-EF, and Flownex) for the SANA-1 simulations. In addition, general thermal radiation model and point reactor kinetics have been incorporated. In order to configure the power conversion system, the system component models are added: pump (or circulator), control systems, simple heat exchangers, turbine/compressor model, valve, and general tables, etc.

Task 2 – RCCS experiment (SNU). In Task 2 researchers performed three categories of experiments: the emissivity measurement test, the separate effect test for the SNU-RCCS (Figure D-36) water tank and the integral test for the SNU-RCCS. In the emissivity measurement test, the measurement methodology was established and verified by a series of experiments. The effects of

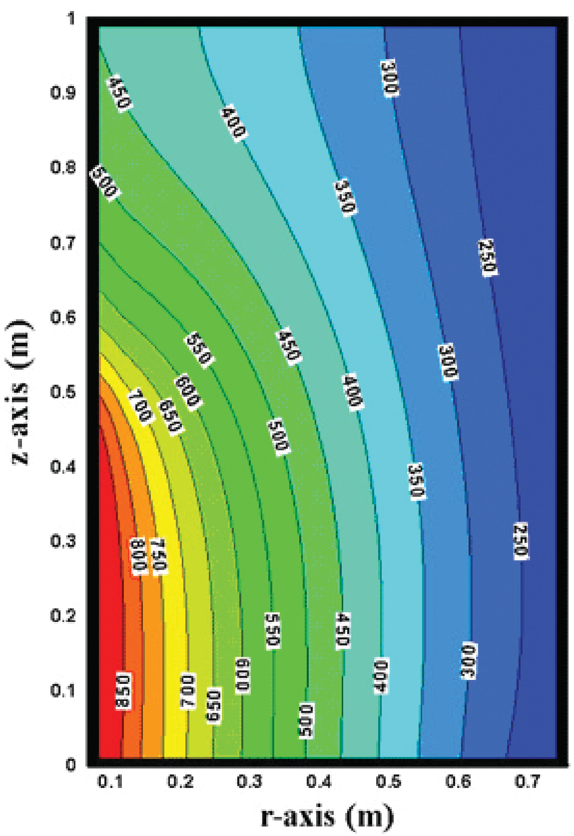


Figure D-35. Calculated pebble and gas temperatures using GAMMA.

filling gas, window, and reflection on the emissivity measurement were investigated. The separate effect test was performed to investigate the heat transfer phenomena in the water tank and cooling pipe. The temperature distributions of the water tank, cooling pipe surface and cooling pipe center were measured along the axis. Heat transfer coefficients of the forced convection inside the cooling pipe were derived from the experiments. The integral test facility was constructed to investigate the overall heat transfer process of the SNU-RCCS and a part of the experiments were performed to evaluate the cooling capability of the RCCS during normal operations. Prof. Park proposed the RETRAN-3D/INT and MARS as analytical tools of the SNU-RCCS. The RETRAN-3D/INT and the CFX calculations of the separate effect test were conducted for code-to-experiment validation and code-to-code benchmark respectively.

Task 3 – Air ingress experiment (KAIST). Task 3 measured chemical parameters at chemical reaction regime. In this experiment, the order of reaction (n) and activation energy (Ea) were estimated as 0.75±0.146 and 218±3.76 kJ/mol respectively, with a 95% confidence level. For analysis of the chemical reaction in an air-ingress accident, the initial reaction rates and CO/CO_2 were measured in a temperature range of 700 to 1500°C and an oxygen concentration of less than 20%. The empirical correlation was developed for CO/CO_2 ratio and it yields good predictions within 10% deviation of the experimental data. A CFD simulation was conducted and compared to the experimental data using the oxidation parameters and the CO/CO_2 ratio developed here. Researchers derived a graphite oxidation (Figure D-37) model to cover the chemical reaction and mass transfer over the whole temperature range and validated the model against the data. The separate experimental facility for the effects of geometry and size was designed and manufactured. As a result, it was found that the internal reaction in the graphite material is more effective than the surface reaction. For more quantitative analysis, this experiment will be performed. A high-temperature annular channel experiment was designed and conducted for confirmation of heat/mass transfer analogy. The analysis is in progress.

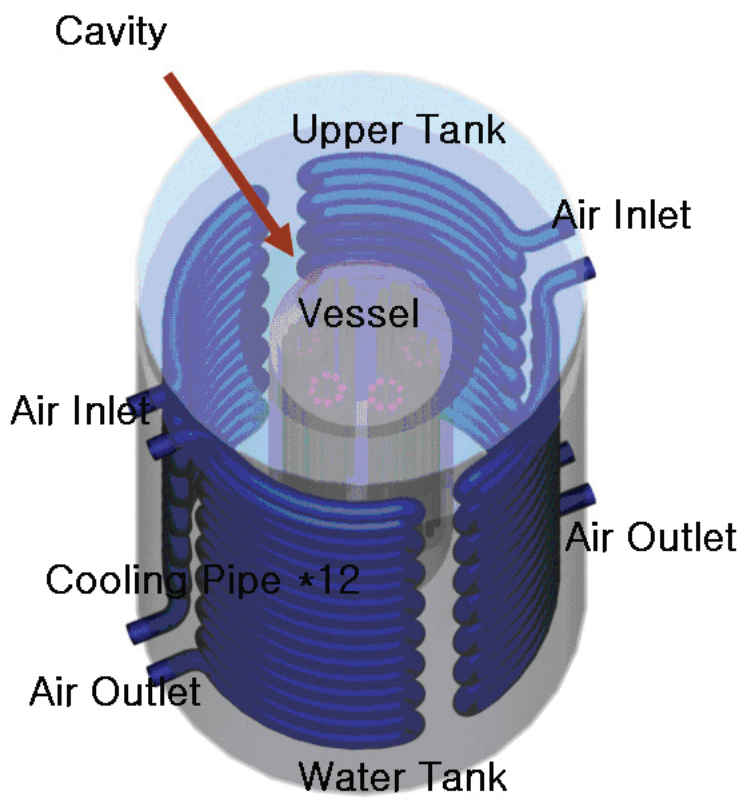


Figure D-36. RCCS-SNU Test Facility.

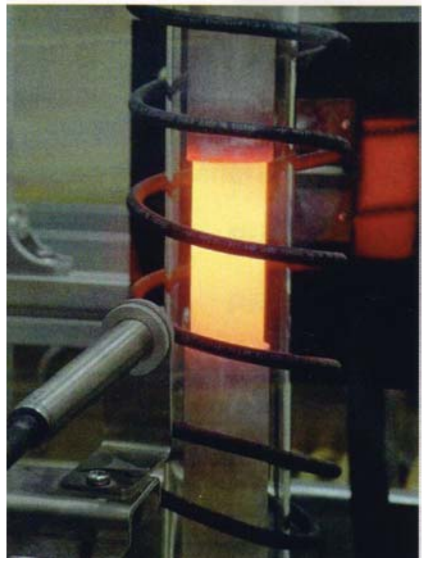


Figure D-37. Test Section of Graphite Oxidation.

Task 4 – Improvement of system codes (INL). The RELAP5/ATHENA and MELCOR computer codes were assessed using experiments that exhibited important phenomena relevant to a LOCA in the VHTGR. Such assessments are required to validate the codes for VHTGR applications. The assessments investigated code capabilities relative to the calculation of diffusion and natural circulation.

RELAP5/ATHENA was assessed using data from an inverted U-tube experiment. Isothermal and non-isothermal experiments were simulated. Diffusion was the most important phenomenon in the isothermal experiment while both diffusion and natural circulation were important in the non-isothermal experiment. The calculated results were in reasonable agreement with the measured values.

RELAP5/ATHENA code was also assessed using natural circulation data from the NACOK experimental facility. The facility simulated the natural circulation of air through a scaled model of a reactor containing a pebble bed core. The calculated mass flow rates were in reasonable agreement with the measured values.

The MELCOR code was assessed using data from a ternary two-bulb diffusion experiment. Previously, we reported some discrepancies between the MELCOR prediction and the measured data. We simulated the same experiment with the latest released version of the code, which included the entire set of gas reactor updates provided by INL. The results from this new calculation agree very well with the experimental results.

Task 5 – Neutronic modeling (UM). The University of Michigan (UM) team has made substantial progress in developing a full-core model of the VHTR, including double heterogeneities and thermal/hydraulic feedback. The particle fuel has been modeled at four levels of increasing complexity, from a single microsphere cell, to a fuel compact cell, to a hexagonal block (assembly) cell, to a full-core model with axial and radial reflectors. All models have included the particle fuel as a homogeneous mixture of fuel and graphite, as well as its true heterogeneous geometry. The modeling of the microsphere fuel has included (1) heterogeneous microspheres centered in a cubical cell of graphite (to preserve the packing fraction), (2) heterogeneous microspheres randomly located in the cubical cell of graphite, and (3) two-region microsphere versions of the heterogeneous microspheres. The results show the importance of modeling the double heterogeneity. The results also indicate that one can model the microspheres centered versus randomly located in the cubical cells and that two regions may be sufficient to accurately account for the multiple microsphere regions. Full-core MCNP5 calculations have been completed for homogeneous fuel and heterogeneous fuel with microspheres centered in the unit cells. Coupled neutronic thermal-hydraulic calculations have been performed with MCNP5 and RELAP5/ATHENA to determine the global flux/power distribution with temperature feedback. These calculations have been performed with homogeneous fuel blocks and are considered preliminary.

Task 6 — Verification and validation (INL and KAIST). Each party has completed initial assessments during the past year. Additional assessments will be performed in year 2005. Assessments of the data being generated at SNU will also be performed.

Planned Activities

The following work will be performed during Year 3.

- Test the recently developed graphite oxidation model, perform further validation of SNU RCCS experiments, and perform transient analysis of the depressurized LOFC accidents.
- Perform GAMMA analysis for verifying component models, point kinetics, and system component models.
- Investigate a geometry effect on graphite oxidation and the effect of moisture and burn-off.
- Complete neutronics models.
- Complete the coupled neutronic T-H calculations to determine the converged flux/power distribution in the VHTR.
- Perform coupled neutronic T-H calculations for heterogeneous fuel and compare with the homogeneous fuel calculations.
- Perform full-core MCNP5 calculations with heterogeneous fuel regions, but with microspheres randomly located within a fuel compact (preserving packing fraction) rather than arranged in a grid or confined to a cubical cell.
- Initiate analysis of depletion. This will involve choosing a depletion module to interface with MCNP5, as well as examining the effect of depletion on the analysis of the double heterogeneities.
- Obtain global flux/power distributions as a function of depletion, including temperature feedback.
- Determine decay heat distribution from the global flux/power distribution as a function of depletion.
- Validate and verify all the models developed from this study using experimental data. Test the recently developed graphite oxidation model, perform further validation of SNU RCCS experiments, and perform transient analysis of the depressurized LOFC accidents.

International Nuclear Energy Research Initiative

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

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

Principal Investigator (Korea): Y. Jeong, Republic of Korea Atomic Energy Research Institue (KAERI)

Collaborators: Westinghouse, University of Michigan, Hanyang University

Project Number: 2003-020-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

Research Objective

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

Research Progress

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

A set of 26 model alloys and standards described in the first year report were selected and prepared and are being corrosion tested in different environments: 360°C pure and lithiated water, 500°C steam and 500°C super-critical water. Corrosion testing is currently being performed in pure water at 360°C both at Westinghouse and at KAERI. An example of one such test (run by KAERI) is shown in Figure D-38 for Zr-Nb alloys. There is a significant decrease in the degree of protectiveness in the lithiated water environment.

The kinetics of the pre-transition regime (about first 50 days) were analyzed using the equation $w = At^n$, where *w* is the weight gain in mg/dm^2, *A* is a pre-exponential constant, *t* is the time of exposure, and *n* is the exponent. This analysis was performed for all the alloys studied and there were a few interesting observations. The first is that the alloy groups were clearly differentiated in their kinetics. The ZrNb alloys have *n* in the range 0.4-0.5, while ZrSnNb have *n*=0.34-0.38, and ZrSn has *n*=0.26-0.3. In contrast, the precipitate-forming alloys have lower *n*, with the Zr CrFe, as well as pure Zr and Zircaloy-4 alloys in the range 0.2-0.25, while the ZrCuMo was between 0.17-0.20. This is summarized in Figure D-39. Researchers plan to

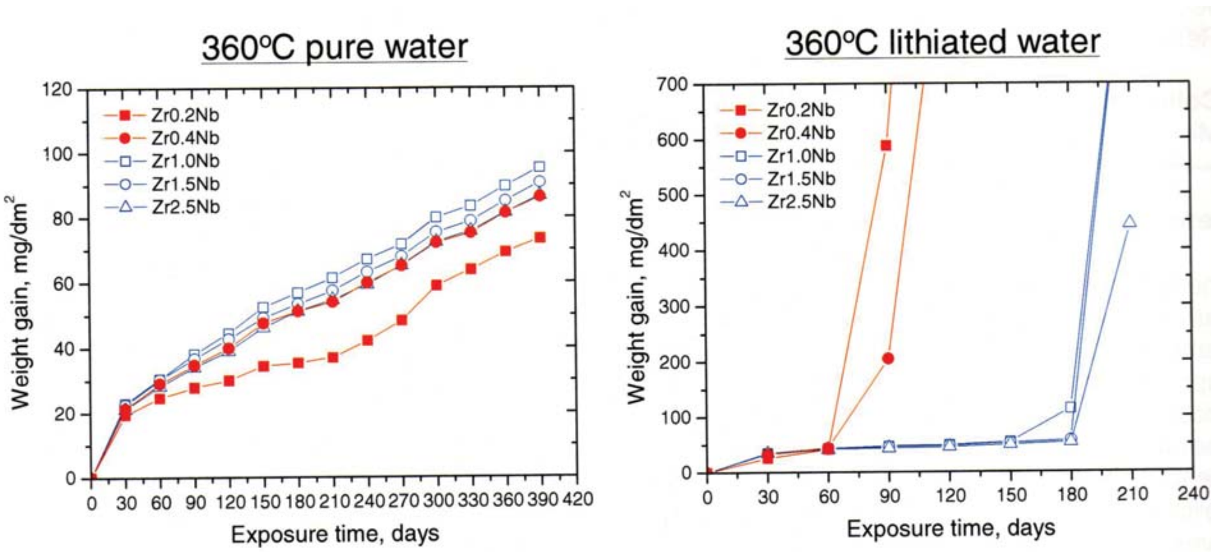


Figure D-38. Corrosion Behavior of Zr-Nb Alloys in 360°C Pure and Lithiated Water.

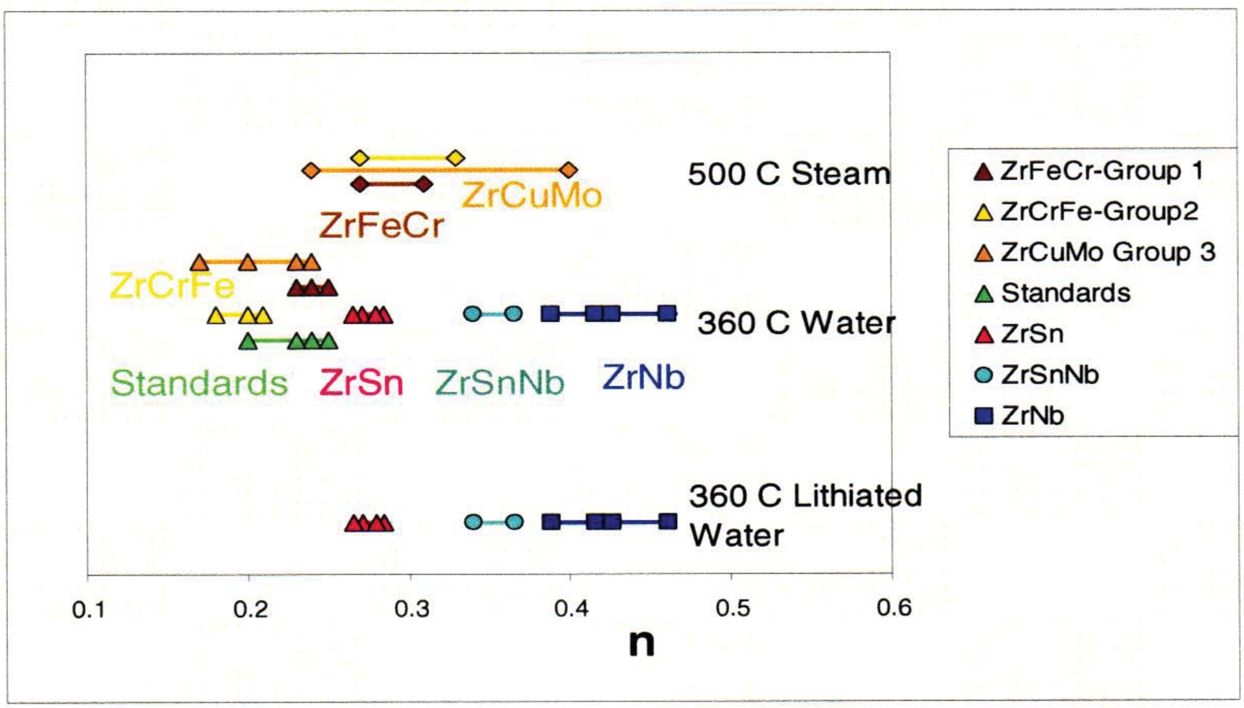


Figure D-39. Value of the Pre-transition Exponent "n"
For the Various Alloy Groups in Three Environments.

examine these oxides to correlate the values of n with features in the oxide layer.

During initial testing at 500°C in both steam and super-critical water, many of the alloys showed high corrosion rates. However, a subset of the alloys (based on Fe and Cr additions) behaved surprisingly well. The uniform corrosion rates of these more protective alloys were similar between 500°C steam and super-critical water tests and, as far as can be compared, similar for corrosion in static or dynamic SCW autoclaves.

It is of interest to compare these rates with those measured in other candidate SCW materials, including ferritic-martensitic alloys, austenitic steels and nickel-based alloys currently being tested in other DOE programs. This comparison is shown in Figure D-40, which includes 500°C corrosion data (both steam and SCW, as indicated) from other research programs at the University of Michigan and University of Wisconsin, as well as data from the present research program. The ferritic-martensitic alloys have a much higher (typically 10 times higher) corrosion rate. The best Zr model alloys have somewhat lower corrosion rates than the F-M alloys, which have high corrosion rates up to the maximum testing time of 20 days. The austenitics show the best uniform corrosion rate but have the tendency to crack in high temperature water. In the ZrFeCr alloys it can be seen that the protective behavior has lasted for the duration of the test, and that the corrosion rates are at or below those seen in the F-M alloys. For use in SCWR, clearly Zr alloys would still have to be tested in SCC conditions and the issue of mechanical strength has to be addressed. However the significant neutron economy they would provide gives incentive to further investigations.

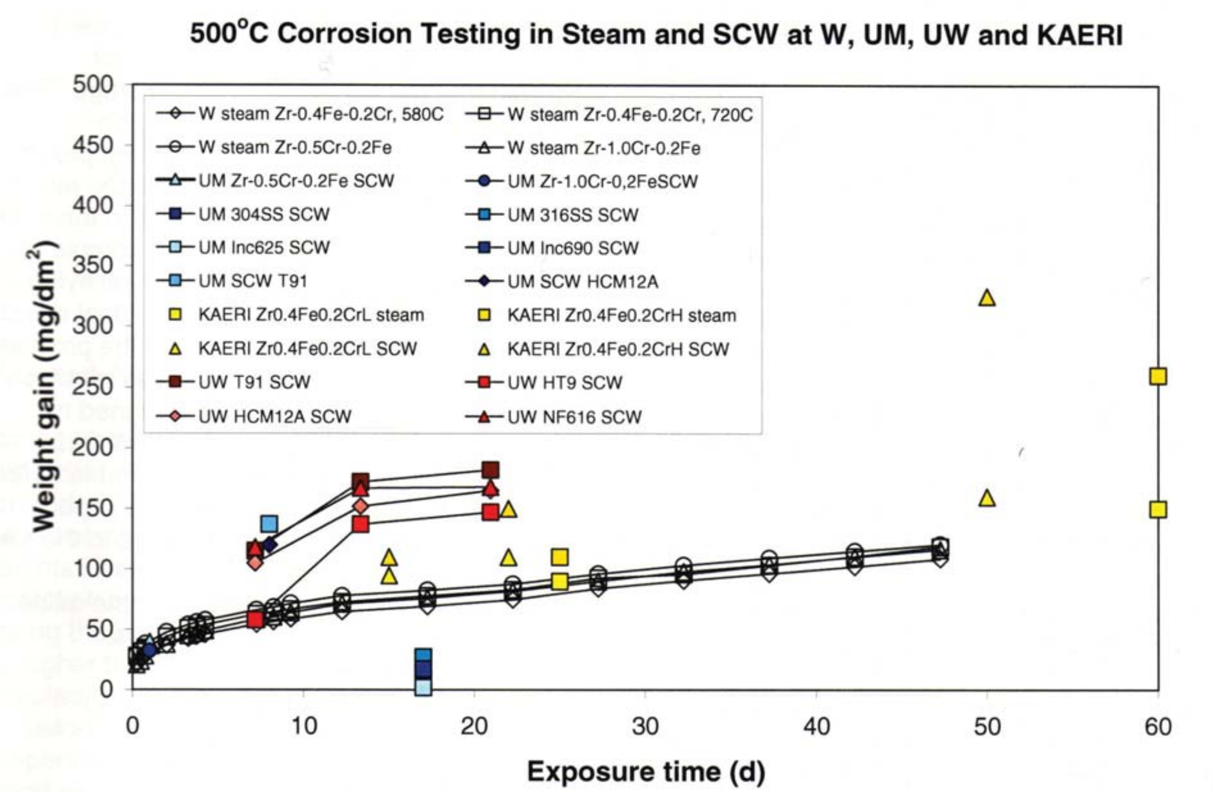


Figure D-40. Weight Gain Versus Exposure Time for 500ºC Corrosion Testing in Various Alloys.

The oxide layers formed in this research program are being examined to discern the clues that lead to different corrosion behavior with a suite of characterization techniques, including microbeam synchrotron radiation diffraction, transmission electron microscopy, conventional and glancing angle x-ray diffraction, electrical impedance spectroscopy, nano-indentation, and optical and scanning electron microscopy. We give here an example of the examination of an alloy oxide layer using microbeam synchrotron radiation at the Advanced Photon Source at Argonne. In the beam line used in this research program, a 0.25 micron x-ray beam is available that allows the examination of the oxide layers in diffraction at a very high spatial resolution.

Figure D-41 shows a series of diffraction patterns (diffracted intensity versus two theta diffraction angle) obtained in the examination of a 20-micron thick oxide layer as the microbeam is scanned from the oxide-metal to the oxide-water interface. There is a significant amount of detail on the oxide structure in Figure D-41 as explained in "Microstructure and Growth Mechanism of Oxide Layers Formed in Zr Alloys Studied with Micro Beam Synchrotron Radiation."[a] There is a marked periodicity of the diffracted intensities of the oxide phases observed in this sample. This periodicity corresponds with other periodic behavior of the oxide, including the thickness at which the first oxide transition occurs. The region near the oxide-metal interface exhibited a different oxide structure than that in the bulk of the oxide or the bulk of the metal. In the metal near the oxide, peaks consistent with the Zr_3O suboxide phase were observed, as well as Zr hydride peaks. In the oxide, the tetragonal phase fraction is highest near the oxide-metal interface. Also a peak corresponding to a highly-oriented tetragonal phase is seen near the oxide-metal interface that is a precursor of the highly textured monoclinic phase seen in the bulk of the oxide. Since this peak is only seen in the first 0.10.2 microns near the oxide-metal interface, it would not be possible to observe it with other techniques. Such detailed information, not available from other techniques, gives insight on the oxide growth mechanism.

The detailed information obtained from examinations such as shown in Figure D-41 allowed us to propose a model for oxide advancement. The model involves the continued nucleation of new highly-oriented tetragonal grains at the oxide-metal interface during the oxide growth process. These grains then transform to monoclinic after they reach a critical size, and then continue to grow into the metal until accumulated stresses force re-nucleation of a new grain. After a critical thickness, the oxide loses protectiveness as a whole and undergoes transition. After transition, the oxide reforms, and since all grains are re-nucleated, the fraction of tetragonal phase is highest at this stage. This model can serve as a template for the description of the oxide advancement process in the various alloys, but it is difficult to relate specific microstructural differences in the oxide to features of the alloy due to the complexity of the commercial alloys with many alloying elements, which prevents the individual effects of each microstructural feature on the process to be determined in isolation. In this research program the model alloys are designed to highlight and isolate the effect of individual alloying elements and microstructural features on the corrosion process. Thus, researchers expect to obtain significant insights as to the role of each alloying element in corrosion protection from the examination of the oxide layers formed on these model alloys.

Planned Activities

For the next year, the corrosion tests and the oxide characterizations in progress will be concluded. We expect that the conclusion of the research program will yield significant insights into the mechanisms of oxide protection by individual alloying elements, thus paving the way for the design of better alloys for higher burnup. Researchers also expect to have completed an initial assessment of the corrosion behavior of Zr-based alloys in high temperature water. It is expected that the final report will contain an assessment of the behavior of Zr alloy in uniform corrosion in super-critical water and an assessment of the reasons for the differentiated behavior of different model alloys in low temperature water.

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a. A.T. Motta, A. Yilmazbayhan, R.J. Comstock, J. Partezana, S. G.P., Z. Cai, and B. Lai, "Microstructure and Growth Mechanism of Oxide Layers Formed in Zr Alloys Studied with Micro Beam Synchrotron Radiation," to be published in *Journal of ASTM International*, 2004.

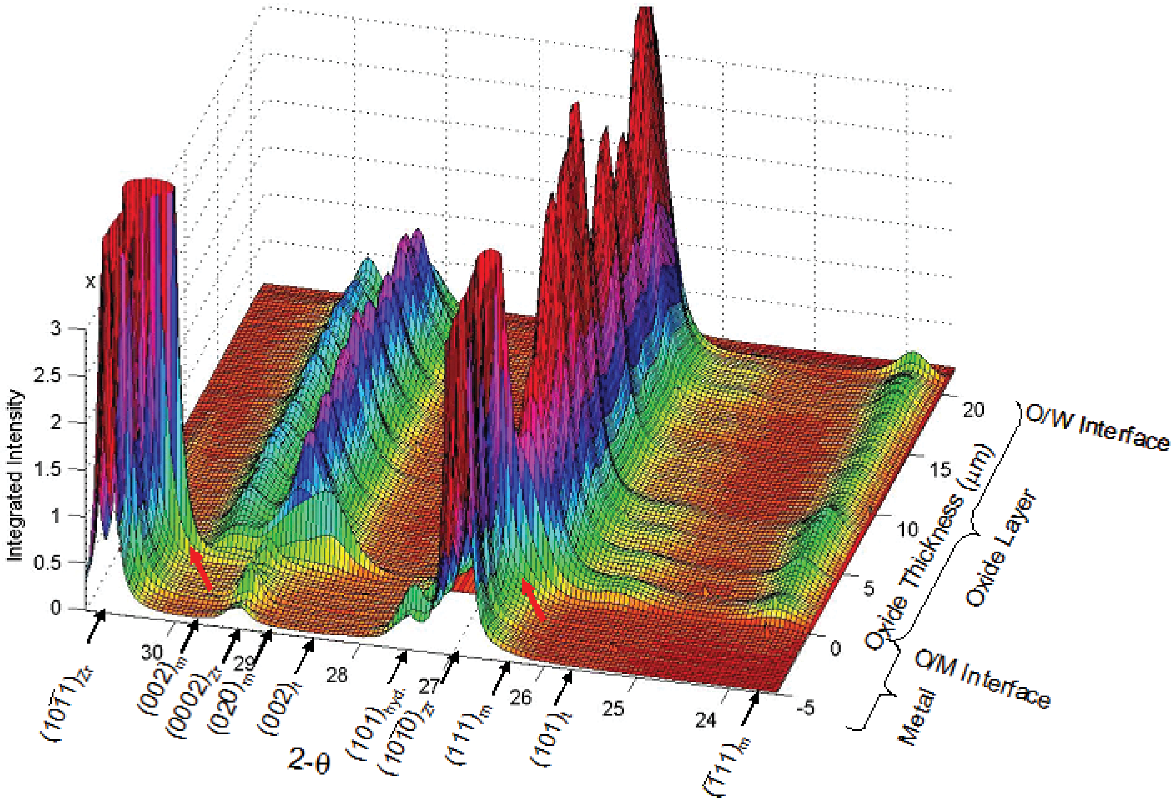

Figure D-41. Diffracted Intensity vs. 2θ Angle at Different Oxide Locations in ZIRLO, from Oxide-metal to Oxide-water Interface, Obtained with μbeam Synchrotron Radiation.

International Nuclear Energy Research Initiative

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

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

Principal Investigator (Korea): S. Park, Korea Atomic Energy Research Institute (KAERI)

Collaborators: University of Illinois, Chicago (UIC)

Project Number: 2003-024-K

Project Start Date: January 1, 2003

Project End Date: December 31, 2005

Research Objective

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

Research Progress

Research progress for 2004 began with test completion for the initial eight alloys (Haynes HR160, 556, and 230; Inconel 600, 601, and 690; IN MA754, and Nickel 200) at 725ºC for three days in 6% Li_2O in LiCl with Ar-10% O_2 gas flowing at 2 mL/min. The completion of this series of tests will allow us to compare the results of the 6% Li_2O and 3% Li_2O concentrations. In addition, we can compare our results with those of KAERI as this series of tests at 725ºC for three days in 3% Li_2O duplicates KAERI experiments for Inconel 600, 601, and Ni 200 at 725ºC for three days (Figures D-42, -43, and -44).

Three new alloys were received for evaluation: Haynes 188 (cobalt-based), Haynes 214 (Alrich), and Ni 201 (lower carbon content than Ni 200). The alloys were tested at 725ºC for three and nine days in 6% Li_2O in LiCl with Ar-10% O_2 gas flowing at 2 mL/min.

Results of heat treatment studies on as-received samples to assess the effects of heating alone show sensitization or carbide precipitation along the grain boundaries. The carbide formation in Nickel 200 may be iron carbide, while in Haynes 230 and 690, which contain chromium, the precipitates are most likely a mix of iron/chromium carbides. This precipitation depletes the metal surrounding the grain boundary of chromium and iron in solution, and diminishes the corrosion resistance. All three alloys were heat treated at 675ºC for ten hours.

Retesting began on three selected alloys: Haynes HR-160, IN MA754, and Haynes 214, on June 17, 2004. These alloys represent the most corrosion-resistant sample coupons in our previous tests. This series of immersion corrosion tests will evaluate the effects of particular parameters on the growth of the corrosion scale. The parameters are O_2 gas concentration in the argon gas (5% and 10% O_2), flowrate of the O_2/Ar gas mixture (2 and 33 mL/min), and duration of tests (three and nine days). The concentration of Li_2O will be kept at 3% and the temperature will be held constant at 675°C. Micrometer measurements and weights are taken pre- and post-testing for all sample coupons. Width and thickness are measured every ½ inch in length and weights are recorded. This will aid in the measurement of the scale thickness, possible passivating layer thickness, and will help determine the extent of corrosion. Composition and structure of the corrosion layers will be characterized, as will critical thickness. Careful measurements of intact and partially separated corrosion layers are also being performed. This testing will be completed by year's end.

Two cathodic protection experiments were performed between April 30 and May 18, 2004, to assess the feasibility of this technique. The testing periods were one and four days in duration; however, termination resulted from the loss of current due to equipment breakdown. The sample coupons are currently being characterized. Mounting and preparations for all metallography, scanning electron microscopy, and other post-test analyses are currently underway at University of Illinois at Chicago, under the direction of Professor J. E. Indacochea and Visiting Professor A. Polar.

Results from the latest immersion corrosion testing indicate that oxygen concentration is not a significant factor in shaping the corrosion scale structure. The comparison of Ar-5% O_2 and Ar-10% O_2 concentrations, respectively, for IN MA754 while all other conditions remained the same, shows the formation of a thin, white, adherent corrosion scale, approximately 12 µm thick, in the Ar-5% O_2 sample. The scale appears to remain unchanged in the coupon exposed to Ar-10% O_2. When the exposure time of the coupon was increased from three to nine days while still maintaining gas flow rate at 2 mL/min of Ar-10% O_2 at 675°C in 3% Li_2O in LiCl, the thickness of the scale shows noticeable growth. The scale remains intact, continuous, and compact; it is approximately 52 µm thick and exhibits a multilayer morphology. In both cases, the scale appears protective.

When the gas flow rate of Ar-10% O_2 was increased from 2 mL/min to 33 mL/min for the same exposure time of nine days, there was a significant impact on the corrosion scale. The scale appears thicker, but exhibits a tendency to break easily. The increased turbulence of the gas flow apparently causes the scale to spall, causing the underlying metal substrate to contact the salt and gas, resulting in more extensive corrosion. In those areas where the corrosion scale remained attached to the substrate, the thickness was measured up to 704 µm. The outer surface of the corrosion scale is dark and quite porous, while nearer to the interface with the base metal, the degree of porosity decreases and the scale appears to become denser.

Haynes HR160 corrosion behavior under the same conditions followed a trend similar to the IN MA754; however, Haynes 214 exhibited atypical invasive corrosion behavior. No corrosion scale remnants were found on the test coupon after the gas flow experiments at 33 mL/min. and there were large voids noted in the metal substrate.

Planned Activities

Research efforts in 2005 will focus on completing the third year test matrix, confirmation testing of corrosion results between the two facilities, completing all testing and characterization of samples, and finalizing the selection of recommended structural materials. The ANL revised test matrix will include immersion testing and testing of selected engineered strategies and barrier systems to arrive at the final recommendations for structural materials. The revised test matrix and chosen samples will be exchanged with KAERI for confirmation corrosion testing and reproducibility of results at each facility. All testing and characterization of samples in the revised test matrix will be completed. After a thorough review of the data and sample performance, a recommendation will be made for structural materials suitable for the electrochemical oxide reduction vessel.

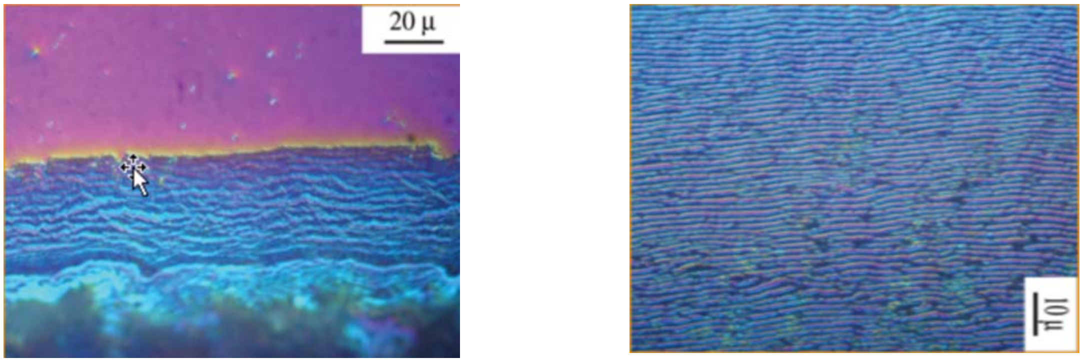

Figure D-42. Inconel MA754 Layered corrosion Scale Made Up of NiO and $LiNiO_2$ After 9 Days at 725°C in 6% Li_2O in LiCl, 10% O_2 in Ar at 2 mL/min (Nomarsky Contrast Illumination).

Figure D-43. Inconel MA754 Best Sample Coupons After 9 Days @ 625°C in 6% Li_2O in LiCl, 10% O_2 in Ar at 2 mL/min.

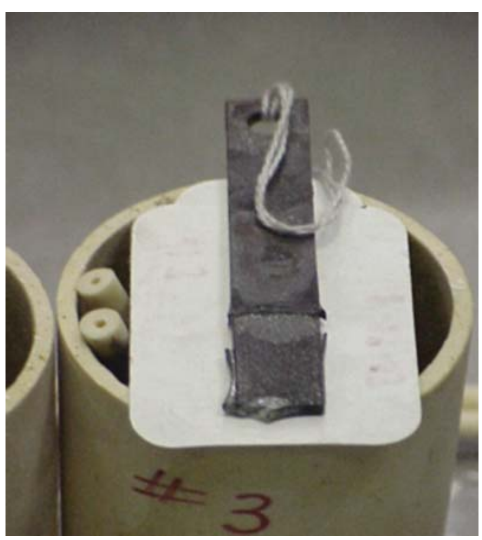

Figure D-44. Inconel 600 Worst Sample Coupons After 9 Days @ 625°C in 6% Li_2O in LiCl, 10% O_2 in Ar at 2 mL/min.

International Nuclear Energy Research Initiative

Screening of Gas-Cooled Reactor Thermal-Hydraulic and Safety Analysis Tools and Experiment Database

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

Principal Investigator (Korea): W. Lee, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2004-001-K

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

Advanced thermal-hydraulic and safety analysis software will play a pivotal role in the design and licensing of advanced reactors—beginning with the Next Generation Nuclear Plant (NGNP)—since it is the first Generation IV system that is likely to be constructed. Consequently, the selection of the tools to be used for advanced reactor thermal-hydraulic analysis and the validation of the selected tools is crucial.

Software with the required capability for analyzing advanced gas-cooled reactor behavior include the ATHENA/RELAP5-3D© systems analysis code and the commercial CFD codes. In addition, other software tools are available that have the capability to analyze the behavior of some gas-cooled system configurations for some transients. Hence, an effort to evaluate the software that may be used for the various portions of the relevant calculation envelope must be performed. Once the software tools are selected, the software must be validated for all the plant operational and postulated accident regions that will require analysis.

This I-NERI project is aimed at: (1) identifying the qualification methodology and the means for defining required thermal-hydraulic experiments for High Temperature Gas-Cooled Reactors, (2) defining a high-level scoping phenomena identification and ranking criteria to identify the dominant phenomena for the NGNP scenarios of interest, (3) performing initial scaling analyses, (4) screening the applicability of the existing experimental database, and (5) performing scoping calculations.

International Nuclear Energy Research Initiative

Investigation of Heat Transfer in Super-Critical Fluids for Application to the Generation IV Super-Critical Water-Cooled Reactor

Principal Investigator (U.S.): M. Modro, Idaho National Laboratory (INL)

Principal Investigator (Korea): Y. Bae, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2004-002-K

Project Start Date: October 1, 2004

Project End Date: September 31, 2007

Project Abstract

Due to the lack of phase change in the core, SCWRs, unlike LWRs, cannot use design criteria based on the critical heat flux concept. The commonly accepted practice is to specify cladding temperature limits that must be met during different events. This makes it very important to predict the heat transfer coefficient to the super-critical water coolant with great accuracy. However, while considerable information exists on heat transfer to super-critical water in round tubes for fossil boilers, little is known about the effect of the geometry and fluid conditions typical of the SCWR core.

This project addresses the critical issue of measuring heat transfer to super-critical water at prototypical SCWR conditions and developing the tools to predict the SCWR thermal behavior. In addition to actual super-critical water, surrogate fluids at super-critical conditions will be used because (1) valuable insight of the physical phenomena can be gained with these fluids, and (2) some existing facilities already use such fluids, which in general have lower critical pressure and temperature, thus affording significant cost and time savings in constructing and operating the experimental facilities.

The GIF SCWR Steering Committee believes that the above objectives can be achieved by means of an integrated international project involving multiple GIF countries. This U.S.-ROK I-NERI project is intended to be only a portion of that GIF project, which could eventually involve EURATOM, Japan, and Canada. This mini-proposal does not cover the full scope of the GIF project.

International Nuclear Energy Research Initiative

Development of Advanced Suite of Deterministic Codes for VHTR Physics Analysis

Principal Investigator (U.S.): T. Taiwo, Argonne National Laboratory (ANL)

Principal Investigator (Korea): H. Joo, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2004-003-K

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

Physics analysis of VHTR, which is a leading candidate for the NGNP, requires different modeling capabilities from those of current LWRs, including the double heterogeneity of coated fuel particles, neutron streaming in coolant channels, and core/reflector coupling. Monte Carlo techniques have these capabilities, but they are not sufficiently efficient for use in routine design calculations, in particular in calculating fuel pin level power distributions and small reactivity effects. Furthermore, several important phenomena such as reactivity feedbacks, flux uncertainty propagation into depletion calculations, and fission product buildup are not properly addressed yet. Therefore, in addition to Monte Carlo tools, it is essential to employ deterministic analysis tools for the scoping and design evaluation of VHTR, similar to the current approaches used for analyzing operating reactors.

However, current deterministic codes for High-Temperature Gas-Cooled Reactor analysis are generally based on old technology in pedigree and platform. These old approaches are cumbersome and provide only approximate treatments. For example, multi-group constants are generated based on pin cell calculations, and whole core calculations and depletion analyses are performed based on two-dimensional diffusion theory. This points to the need for adapting modern deterministic tools for VHTR analysis and for verifying and validating them for current and future applications. Improved computational methods and models will enhance the safety and economics of VHTR by reducing the computational uncertainties of performance parameters. Efforts are being made both at ANL and at KAERI (Pilot Study for Establishing VHTGR Design Concepts) to evaluate the currently available physics analysis tools and to identify and implement required enhancements. The proposed project is aimed at leveraging these U.S. and Korean efforts through international collaborations.

The objective of the project is to develop an advanced suite of deterministic codes for use in VHTR design and licensing. The capabilities of currently available physics analysis tools will be investigated, focusing on the modeling requirements for VHTR. Based on the assessment, required enhancements to the lattice physics, whole-core, and fuel cycle modeling capabilities will be identified. Focusing on application to block-type VHTR, an advanced suite of codes based on the conventional two-step lattice and whole-core calculation approach will be developed by implementing these identified enhancements. In parallel, adaptation for VHTR analysis of the 3-D whole-core transport code DeCART, which is being developed for LWR applications at KAERI under an ongoing U.S./ROK I-NERI project of ANL and KAERI, will be pursued. This code eliminates the approximations and laborious multi-group constant generation need of the two-step approach by representing local heterogeneity explicitly without homogenization, using a multi-group cross-section library directly without group condensation, and incorporating pin-wise thermal-hydraulic feedback. Verification and validation tests of the new suite of codes will be performed using appropriately defined benchmark problems and available experimental data.

International Nuclear Energy Research Initiative

Development of Voloxidation Process for Treatment of LWR Spent Fuel

Principal Investigator (U.S.): K. Howden, Idaho National Laboratory (INL)

Principal Investigator (Korea): J. Park, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2004-004-K

Project Start Date: June 1, 2004

Project End Date: April 30, 2007

Project Abstract

The United States Department of Energy is developing a head-end fuel treatment process, known as the voloxidation process, as part of the Advanced Fuel Cycle Initiative. Voloxidation, which involves oxidizing uranium oxide fuel using either air or oxygen, provides three important advantages when used as a head-end treatment process. First, it may be used to separate the fuel from the cladding. A decladding step is desirable in that it may simplify process flow sheets by excluding the cladding constituents from the fuel constituents. Segregation of cladding may also result in generation of less high level waste. In addition to decladding, voloxidation may be used to decrease the particle size of the fuel, which will in turn increase the kinetics of downstream treatment operations and the efficiency of dissolution. Finally, voloxidation treatment may remove problematic constituents from the fuel prior to downstream treatment operations. Fission products such as cesium-137, krypton-85, xenon-133, technetium-99, carbon-14, and tritium (H_3) may be removed prior to fuel dissolution, simplifying the process flow sheets and yielding more flexible waste treatment operations. Head-end removal of gaseous fission products is particularly important in aqueous reprocessing where the presence of NOx gases, generated during fuel dissolution operations, makes separation of these gases much more challenging. The head-end voloxidation process is equally applicable to aqueous or pyrochemical treatment processes.

KAERI is pursuing the development of a similar technology, known as the Oxidation and Reduction of Oxide fuel (OREOX) process. This process is being researched in association with development of the DUPIC (direct use of spent PWR fuel in CANDU reactor) process for recycle of PWR spent fuel to CANDU reactors. The objective of the OREOX process is to decrease the particle size of the fuel and to increase the sinterability of powder for being in the optimum range of surface area and crystallite size, etc. This will in turn give high density sintered pellets for CANDU fuel specification. During development of the DUPIC process, KAERI scientists have gained considerable experience with oxidation and reduction processes of spent LWR fuel. The oxidation process is the same as the voloxidation process. KAERI has been measuring and analyzing the release rates of Kr-85, cesium, Xe, etc. Also, in order to trap these fission products, various concepts have been developed and applied to each off-gas treatment system of the OREOX system and the sintering furnaces in the DFDF (DUPIC Fuel Development Facility).

In the project, the two parties will conduct joint experiments in a U.S. DOE laboratory with actual LWR spent fuel for the purpose of measuring the efficiency of recovery of volatile fission products (Xe, Kr, I, H-3, and oxides of carbon). DUPIC process experience will provide an extremely useful baseline process to support the successful development of the voloxidation process. The two parties will conduct independent experiments on the development of various trapping concepts for immobilizing the volatiles released in the course of the voloxidation treatment, with results to be shared in periodic information exchange meetings.

International Nuclear Energy Research Initiative

Development and Test of Cladding Materials for Lead-Alloy Cooled Transmutation Reactors

Principal Investigator (U.S.): N. Li, Los Alamos National Laboratory (LANL)

Principal Investigator (Korea): T. Song, Korea Atomic Energy Research Institute (KAERI)

Collaborators: Seoul National University

Project Number: 2004-005-K

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

Within the work scope of AFCI and Generation IV LFR coolant and materials projects, LANL will collaborate with the KAERI/SNU team on several important aspects of materials testing and coolant chemistry control. Lead-alloy corrosion tests of candidate materials of interest to advanced nuclear systems for transmutation will be performed in the forced circulation loops developed by LANL, KAERI, and SNU. The corrosion of materials in closed loops with coolant chemistry control (oxygen) is influenced not only by local conditions, but also by system configurations. LANL will apply the recently-developed kinetic models to analyze the test results, evaluate KAERI/SNU loop designs and choice of test conditions, and assess corrosion product transport and effects on loop performance. Corrosion product transport behavior will be monitored using liquid and particulate sampling techniques at LANL, KAERI, and SNU. Oxygen sensor designs will be improved to alleviate current limitations, including potential bias, sluggish kinetics, and long-term instability. Oxygen sensors will be tested at each institute to verify functions of sensors. Low temperature enmbrittlement of candidate steels will be tested and results will be analyzed to identify underlying mechanisms and develop mitigating means.

International Nuclear Energy Research Initiative

Alternative Methods for Treatment of TRISO Fuels

Principal Investigator (U.S.): C. Snyder, Argonne National Laboratory (ANL)

Principal Investigator (Korea): E. Kim, Korea Atomic Energy Research Institute (KAERI)

Project Number: 2004-006-K

Project Start Date: June 1, 2004

Project End Date: May 31, 2007

Project Abstract

The U.S. DOE AFCI and the Advanced Spent Fuel Conditioning Process of the KAERI are developing transmutation strategies for separating and transmuting the transuranic (Pu, Np, Am, and Cm) and long-lived fission products (^{99}Tc and ^{129}I) contained in spent nuclear fuel. Among the current inventory is coated particle TRISO fuel from HTGR. TRISO fuel particles have four layers surrounding the fissile material: porous carbon, inner pyrolytic carbon, silicon carbide barrier coating, and outer pyrolytic carbon. Current methods in the treatment of TRISO fuel call for dry mechanical crushing and/or grinding using a crusher assembly and an attritor grinding mill for breaching the outer pyrolytic carbon and silicon carbide layers of the fuel particle. The objective of this collaborative project is to develop alternative methods for breaching the protective outer layers of spent TRISO nuclear fuel and to explore new separation methods for the resulting fines and fuel.

Appendix E

U.S./OECD Collaboration
Project Summary

International Nuclear Energy Research Initiative

Project No.	Title
2002-001-N	Melt Coolability and Concrete Interaction (MCCI) Program

International Nuclear Energy Research Initiative

Melt Coolability and Concrete Interaction (MCCI) Program

Principal Investigator (U.S.): M. Farmer, Argonne National Laboratory

Foreign Institution: Organization for Economic Cooperation and Development (OECD)

Project Number: 2002-001-N

Project Start Date: March 1, 2002

Project End Date: December 31, 2005

Research Objectives

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

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

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

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

Research Progress

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

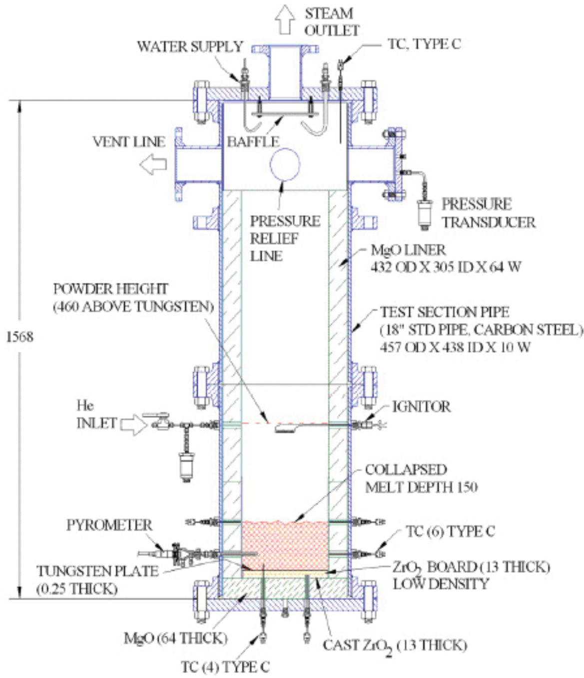

Figure E-1. SSWICS Test Section.

Figure E-2. SSWICS Test Facility.

Figure E-3. Radially Sectioned Corium Sample.

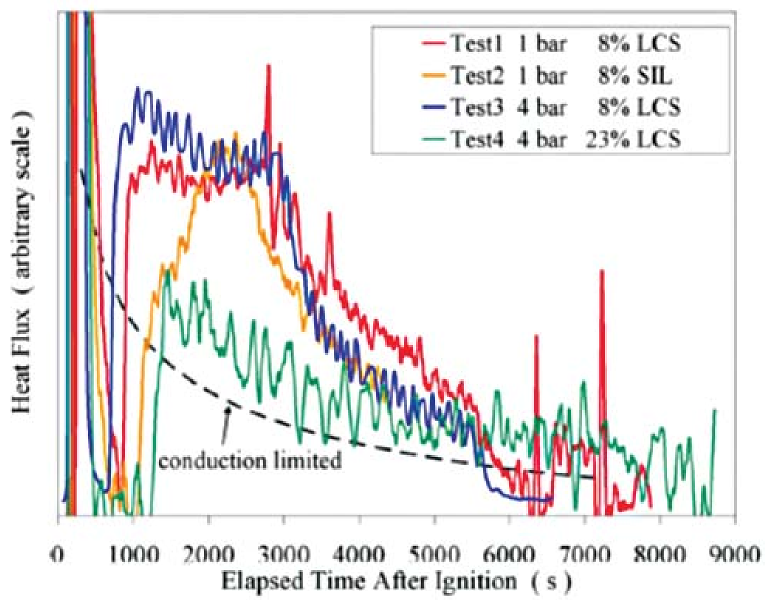

Figure E-4. Corium/Water Heat Flux.

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

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

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

Planned Activities

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

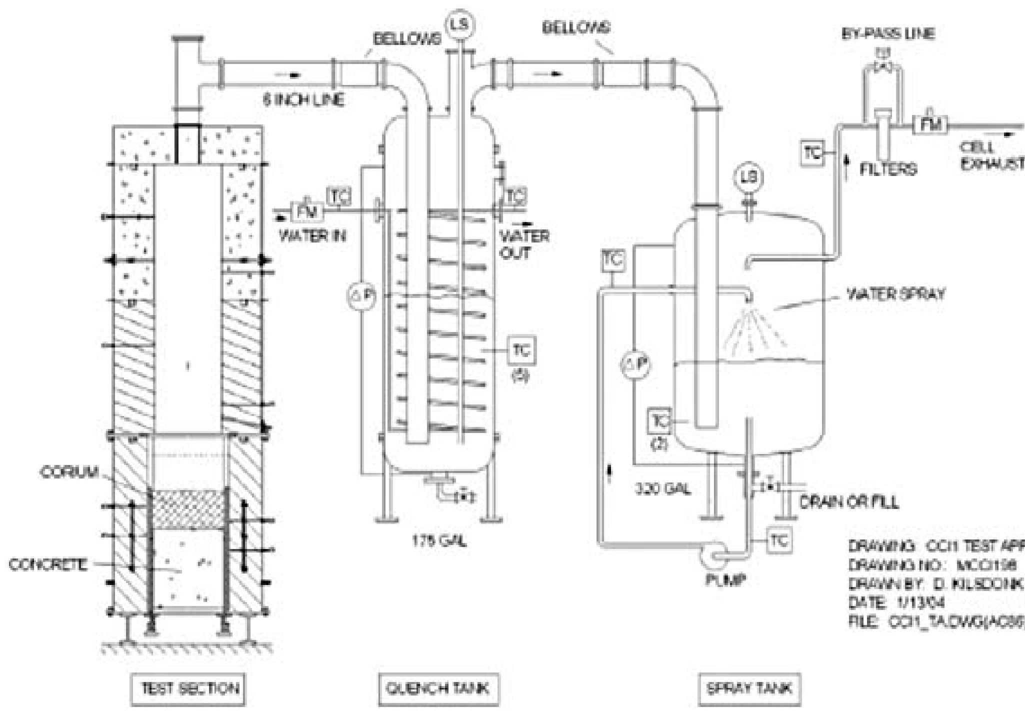

Figure E-5. CCI Test Facility.

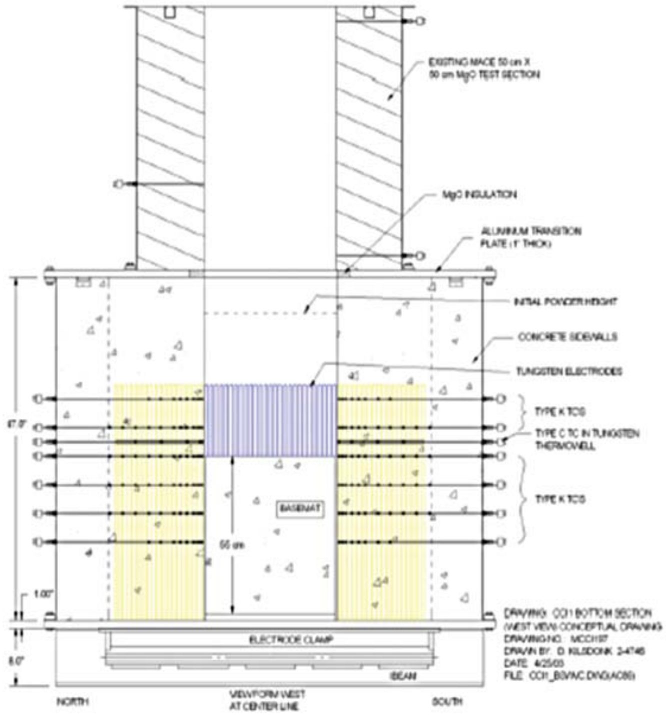

Figure E-6. CCI-1 Test Section.

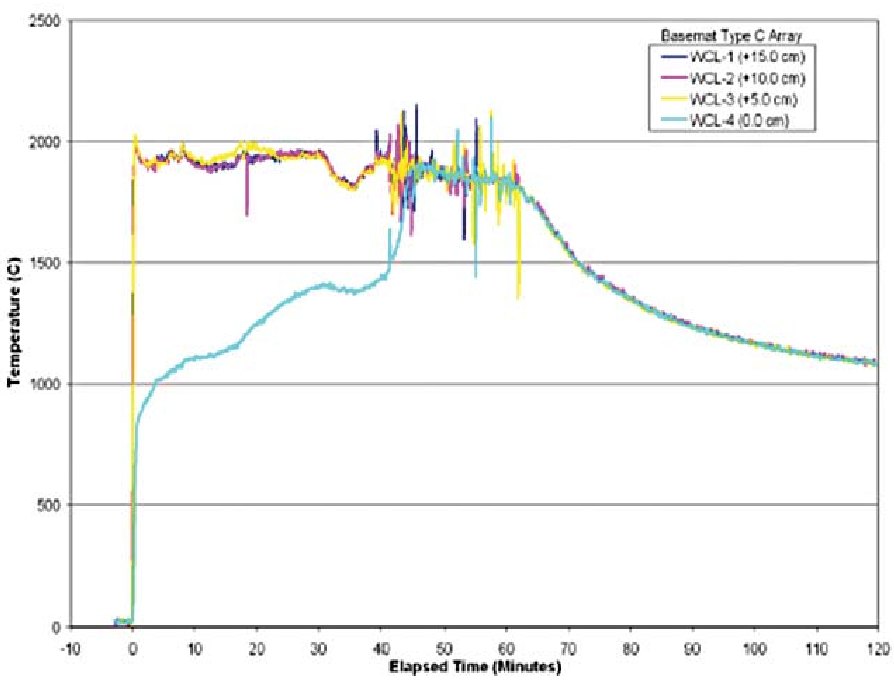

Figure E-7. CCI-1 Melt Temperature Data.

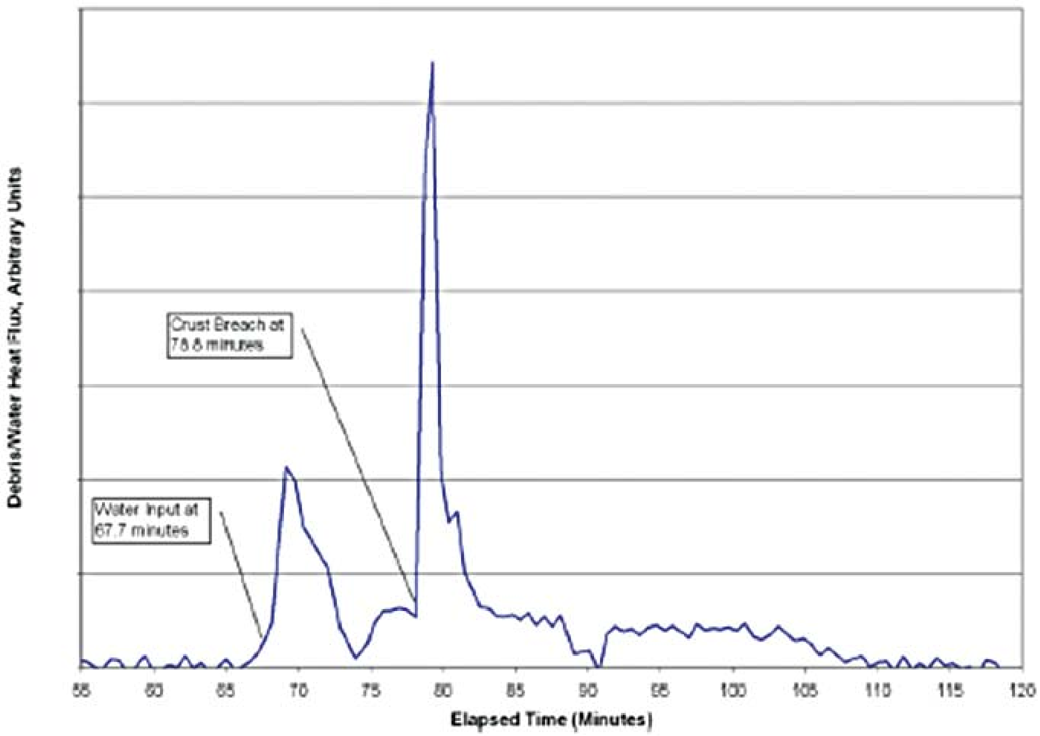

Figure E-8. CCI-1 Corium/Water Heat Flux.