“Nuclear Education and Research Initiative” Project
Global Center-Of-Excellence (G-COE) Grant
Japan Society for Promotion of Science (JSPS)

Proceedings
of
the Joint International Workshop

Nuclear Technology and Society
- Needs for Next Generation -

Faculty Club
University of California, Berkeley
January 6 – 9, 2008

Organized by
Department of Nuclear Engineering, University of California, Berkeley (UCBNE)
and
Department of Nuclear Engineering and Management, University of Tokyo (UTNEM)

Supported by
San Francisco Office, JSPS
FOREWORD

Nuclear energy and more general, nuclear technology fields are undergoing a renaissance in the U.S. and around the world, with strong emphasis on energy independence and security, the need to replace fossil fuels with non-green-house-emitting energy sources, the need to secure our borders from possible terrorist nuclear treats and the need to prevent proliferation of nuclear materials. While Japan and France continued to relay on nuclear power, U.S. has neglected this important source of energy for the last 30 years. After the U.S. Energy Policy Act was passed by the U.S. Congress in August 2005, which provided large incentives to encourage industry to start new construction of nuclear power plants (NPP) in the USA, several large U.S. and international industrial consortia announced that they are pursuing combined construction and operating licenses for construction of up to 36 NPPs.

A new strategy - Global Nuclear Energy Partnership (GNEP) - was recently developed in the US regarding the broader use of nuclear power in the world. The goal of this international partnership is to expand domestic use of nuclear power and reduce dependence on foreign sources of fossil fuel, recycle spent nuclear fuel using new proliferation-resistant technologies, develop fuel services program to enable developing nations to acquire nuclear energy economically while limiting proliferation risks, develop small scale reactors designed for the needs of developing countries, and improve nuclear safeguards to enhance the proliferation-resistance, monitoring and safety of expanded nuclear power.

However, nuclear science and engineering has been a sorely neglected field for many years in terms of support required to train the next generation of engineers and scientists. One of the major constraints in the planned expansion of nuclear energy in the U.S. is a growing shortage of qualified manpower at all levels. *Graduating Engineer & Computer Careers* (Winter/Spring 2007 issue) talks about “The Resurgence of Nuclear Energy” pointing out that “over the next 10 years studies indicated that as many as 90,000 nuclear professionals may be needed to meet the needs of all segments of the industry in the U.S. alone.”

The Nuclear Engineering Departments in U.S. which have a significant role in educating the next generation of experts and in contributing to the cutting-edge research in these strategic areas, cannot fulfill this role without global international collaboration. Thus, it is extremely important that the University of Tokyo and University of California, Berkeley, collaborated to win a major Global Center of Excellence award entitled “Nuclear Education and Research Initiative” and funded by Japan Ministry Monbu-kagaku-sho (MEXT) and Japan Society of Program of Science (JSPS).

This workshop represents the beginning of a long term international collaboration between the Department of Nuclear Engineering and Management at the University of Tokyo and the Department of Nuclear Engineering at University of California, Berkeley.

We would like to express our appreciation to MEXT and JSPS, as well as to San Francisco Consulate General of Japan and UC Berkeley College of Engineering for their support.

Jasmina Vujic, Professor and Chair
Department of Nuclear Engineering
University of California, Berkeley
Joint International Workshop

Nuclear Technology and Society – Needs for Next Generation –

Organized by

Department of Nuclear Engineering, University of California, Berkeley (UCBNE) and
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under

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Faculty Club, University of California, Berkeley, CA

By Invitation Only – Casual Dress Recommended

Sunday, January 6 – Wednesday, January 9, 2008

Sunday, January 6, 2008 (Faculty Club, University of California, Berkeley)

5:00 PM – 8:00 PM  Reception
Welcome:
Jasmina Vujic, Professor, Chair, UCBNE
Yoshiaki Oka, Professor, UTNEM, Principal Investigator of GCOE

Signaling Ceremony for Collaboration Agreement between UCBNE and UTNEM

Addresses:
S. Shankar Sastry, Dean, College of Engineering, UCB
Yasumasa Nagamine, Consul General, Consulate General of Japan in San Francisco
Seishi Takeda, Professor, Director, JSPS-SF

Monday, January 7, 2008 (Faculty Club, University of California, Berkeley)

7:30 AM – 8:00 AM  Continental Breakfast
8:00 AM – 8:10 AM  Opening Remarks
Jasmina Vujic, Professor and Chair, UCBNE

8:10 AM – 8:20 AM  Workshop Objectives and Organization
Joonhong Ahn, Professor, UCBNE
Plenary Session 1: Experiences and Future Directions of Nuclear Technologies
Chaired by Prof. J. Vujic, UCBNE

8:20 AM – 9:10 AM  Overview of Nuclear Technology
Donald R. Olander, Professor, UCBNE

9:10 AM – 10:00 AM  Experience and Views of Nuclear Reactor Studies
Yoshiaki Oka, Professor, UTNEM, Principal Investigator of GCOE

10:00 AM – 10:20 AM  Break

10:20 AM – 11:10 AM  Nuclear Technology and Society -- A Historian's View
Cathryn Carson, Associate Professor, History Department, UCB

11:10 AM – 12:00 PM  Combining Nuclear, Renewable, and Fossil Fuel Cycles For Sustainability
Charles Forsberg, Oak Ridge National Laboratory

12:00 PM – 1:00 PM  Luncheon

Plenary Session 2: Progress and Challenges in Nuclear Technologies – Safety and Regulations --
Chaired by Prof. Y. Oka, UTNEM

1:00 PM – 1:40 PM  Current and GNEP Regulatory Perspectives: Issues of High-Level Nuclear Waste Management
Tae Ahn, Nuclear Regulatory Commission

1:40 PM – 2:20 PM  US Radioactive Waste Management Policy
Robin Sampson, Office of Civilian Radioactive Waste Management, US DOE

2:20 PM – 3:00 PM  Technology Innovation in Instrumentation and Control System for Nuclear Power Plant
Takaharu Fukuzaki, Professor, UTNEM

3:00 PM – 3:20 PM  Break

Plenary Session 3: Progress and Challenges in Nuclear Technologies – Innovative Technologies for Future Industries --
Chaired by Prof. P. F. Peterson, UCBNE

3:20 PM – 4:00 PM  Nuclear Technologies for National Security
Kai Vetter, Associate Professor in Residence, UCBNE and LLNL

4:00 PM – 4:40 PM  Computer Simulation and New Materials Development
Taira Okita, Associate Professor, Department of Quantum Engineering and Systems Science, UT

4:40 PM – 5:20 PM  Seismic Safety, Risk Reduction, and Performance Based Design Aimed at Nuclear Facility Structures
Bozidar Stojadinovic, Associate Professor, Department of Civil and Environmental Engineering, UCB

6:00 PM – 7:30 PM  Dinner

After Dinner Lecture
Moderated by Dr. C. Forsberg, ORNL
Holistic Design: Safety, Reliability, Security and Sustainability
Insights for the PB-AHT
Per F. Peterson, Professor, UCBNE

Tuesday, January 8, 2008

7:30 AM – 7:50 AM Continental Breakfast
7:50 AM – 8:00 AM Objectives and Organization of the Second Day
Joonhong Ahn, Professor, UCBNE

Session 1: Student Paper Presentations– Risk Management and Radiolysis
Moderated by Prof. T. Okita, Department of Quantum Engineering and Systems Science, UT

8:00 AM – 8:15 AM Allocation of Resources Against Strategic and Non-Strategic Threats for Nuclear Energy Systems
Lance Kim, UCBNE

8:15 AM – 8:30 AM Application of Fluorescent Probe to Yield Measurement of OH in Water Radiolysis with Heavy Ions
Takuya Maeyama, UTNEM

8:30 AM – 8:45 AM Development of New Micro-Strip Gas Chambers for X-ray and Neutron Applications
Hisako Niko, UTNEM

8:45 AM – 9:00 AM Development of Geologic Repository Models for Site Selection and Design Optimization
Haruko Murakami, UCBNE

9:00 AM – 9:15 AM Study on a Sensor Network System with a Self-Maintenance Function for Plant Monitoring System
Takeshi Fujiwara, UTNEM

9:15 AM – 9:30 AM Recent Development of Compact Neutron Generators at LBNL
Ying Wu, UCBNE

9:30 AM – 9:45 AM A Steady State Model for the Heat Pipe-Encapsulated Nuclear Heat Source
Steven Mullet, UCBNE

9:45 AM – 10:00 AM Break

Session 2: Student Paper Presentations- Design and Analysis for Nuclear Reactor -
Moderated by Prof. Fukuzaki, UTNEM

10:00 AM – 10:15 AM Neutronic Design of the PB-AHT
Massimiliano Fratoni, UCBNE

10:15 AM – 10:30 AM Modeling and Transient Analysis for the Modular Pebble-Bed Advanced High Temperature
Aurelie Niquille, UCBNE

10:30 AM – 10:45 AM Intermediate Heat Exchanger Dynamic Thermal Response Model
Eugenio Urquiza-Fernández, Department of Mechanical Engineering, UCB

10:45 AM – 11:00 AM Partial Safety Analysis for a Reduced Uranium Enrichment Core for the High Flux Isotope Reactor
C. Galvez Velit, UCBNE
Joint International Workshop: Nuclear Technology and Society – Needs for Next Generation
Berkeley, California, January 6-8, 2008, Berkeley Faculty Club, UC Berkeley Campus

11:00 AM – 11:15 AM  PWR Control Rod Ejection Analysis with the MOC Code DECART
Mathieu Hursin, UCBNE

11:15 AM – 11:30 AM  Incompressible Free Surface Flow Analysis Using Moving Particle Semi-Implicit Method
Masahiro Kondo, Department of Quantum engineering and system science, UT

11:30 AM – 11:45 AM  Maximizing Power of Hydride Fuelled Pressurized Water Reactor Cores
Raluca Scarlat, UCBNE

11:45 PM – 1:00 PM  Lunch

Session 3: Student Paper Presentations – Advanced Materials
Moderated by Prof. B. Wirth, UCBNE

1:00 PM – 1:15 PM  Options for the Hydride Fuels Reprocessing
Vladimir Mozin, UCBNE

1:15 PM – 1:30 PM  Development of Portable X-Band LINAC X-Ray Source for Non-Destructive Testing
Tomohiko Yamamoto, UTNEM

1:30 PM – 1:45 PM  A Study of Reflection and Connection Materials Used for Transmitting and Condensing Scintillation Light by Means of Optical Fiber
Masato Yamawaki, Department of Quantum engineering and Systems Science, UT

1:45 PM – 2:00 PM  Transient Hydride Fuel Behavior in LWR
Kurt Terrani, UCBNE

2:00 PM – 2:15 PM  Micro-Structural Evolution in Cerium Dioxide Irradiated with Heavy Ions at High Temperature
Takeshi Mihara, UTNEM

2:15 PM – 2:30 PM  Screening of Fullerene C60 Crystallization using Micro-Fluidic Devices
Kyosuke Shinozaka, Graduate School of Frontier Sciences, UT

2:30 AM – 3:00 AM  Break

3:00 AM – 4:50 AM
Moderated by Tom Isaacs, LLNL

Panelists:  
Joonhong Ahn, Professor, UCBNE
Jor-Shan Choi, Professor, UTNEM, LLNL
Takaharu Fukuzaki, Professor, UTNEM
Yoshiaki Oka, Professor, UTNEM
Taira Okita, Associate Professor, Dept. Quantum Eng. Systems Science, UT
Jasmina Vujic, Professor, Chair, UCBNE

4:50 PM – 5:00 PM  Closing Remarks
Jasmina Vujic, Professor, Chair, UCBNE
Wednesday, January 9, 2008

Group Discussions by Students
Students will meet in groups to exchange informal discussions on interesting topics of researches and others.
Reception and Opening Ceremony

5:00 PM – 8:00 PM

January 6, 2008
Distinguished guests, ladies and gentlemen; Today we are present here in the beautiful California to have the joint international workshop of “Nuclear Technology and Society-Needs for Next Generation”.

I would like to extend my welcome to all guest, professors and students and to give my sincere thanks to University of California Barkley for hosting the workshop.

I also wish to express thanks to our ministry, Monbu-kagaku-sho (MEXT) and Japan Society for the Promotion of Science (JSPS) for supporting the workshop and our Global Center Of Excellence (COE) program “Nuclear Education and Research Initiative” called GoNERI program.

Nuclear power is a method for reducing greenhouse gas emission and satisfying our modern society’s high demand for energy. Its benefit of “clean air” becomes popular among general public in the world now.

The GoNERI program is a research and education program in response to a variety of worldwide nuclear utilization subjects such as: protection of the global environment, supplying safe and stable nuclear energy, and applying radiation for healthy, productive and prosperous lives.

We take up three major subjects in the GoNERI program; nuclear energy sociology, nuclear energy and radiation application.

Nuclear energy sociology is unique. It is a broad field pertaining to societal perception and interaction with nuclear energy. In the GoNERI program, we mainly deal with three subjects of nuclear energy sociology; nuclear energy law and legislation, nuclear nonproliferation and harmonization of technology and society. Through the studies, we prepare the next generation of researchers to grasp the perspectives of the complicated field of nuclear energy.

I wish to express my sincere thanks to Professor Sharkar Sastry, the Dean of College of Engineering of UC Berkeley, and Professor Jasmina Vujic and Professor Joonhong Ahn of the Department of Nuclear Engineering of UC Berkeley, and Mr. Yasumasa Nagamine, San Francisco Consulate General of Japan and Professor Seishi Takada, the Director of JSPS San Francisco Office.

Please enjoy the workshop.

Yoshiaki Oka, Professor
Principal Investigator of GCOE
Department of Nuclear Engineering and Management, the University of Tokyo
Plenary Session 1:
Experiences and Future Directions of Nuclear Technologies

Chaired by
Prof. J. Vujic
UCBNE

8:20 AM – 12:00 PM
January 7, 2008
Reflections on the early days of Nuclear (non-power reactor applications)

D. R. Olander

Department of Nuclear Engineering

Jan. 7, 2008

My (technical) Trips to Japan

May, 1983 -

Japan Society for the Promotion of Science

• Univ. of Osaka (Profs. Imoto & Miyake);
• Univ. of Kyoto (Profs. Oishi & Higashi)
• Univ. of Kyushu (Prof. Furuya)
• Univ. of Tokyo (Profs. Kanno & Ishino)
• JAERI (Dr. Iwamoto)
• PNC (Dr. Komatsu)
*The High-Temperature Chemistry of Light-Water Reactors and All-Metal Reactors*

<table>
<thead>
<tr>
<th>Japanese</th>
<th>U. S.</th>
</tr>
</thead>
<tbody>
<tr>
<td>T. Fujino (JAERI)</td>
<td>D. Cubicciotti (EPRI)</td>
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<tr>
<td>J. Komatsu (PNC)</td>
<td>M. Adamson (LLNL)</td>
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<tr>
<td>M. Miyake, S. Imoto (Osaka U.)</td>
<td>L. Leibowitz (ANL)</td>
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<tr>
<td>Y. Takahashi (Tokyo U.)</td>
<td>C. Johnson (ANL)</td>
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<tr>
<td>H. Furuya (Kyushu U.)</td>
<td>I. Johnson (ANL)</td>
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<tr>
<td>M. Yamawaki (Tokyo U.)</td>
<td>D. Olander (UCB)</td>
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<tr>
<td>J. Oishi (Kyoto U.)</td>
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<tr>
<td>K. Naito (Nagoya U.)</td>
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<tr>
<td>M. Kanno (Nagoya Tech. U.)</td>
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Jan. 1990: Panel on Nuclear Power
Japan Tech. Evaluation Center (JTEC)
*Nuclear Power in the U. S. and Japan*  
*a Comparison*  
(Funded by DOE and NSF)

Kent Hansen, MIT (Chair)
Ersel Evans, PNL
J. White, Mgr., Adv. Controls Progr., ORNL
V. Ransom, INL
D. Olander (UCB)
Sites Visited

CRIEPI (Kinoshita & Okamoto)
JAERI (Matsura & Konomi)
Nippon Nuclear Fuel Development (Nagai)
National Research Institute for Metals (Okada)
PNC (Konomi)
TEPCO (Omoto)
Kobe Steel (Ohizumi)
Mitsubishi Heavy Industries (Kondo)

Issues studied

- funding of nuclear R&D
- Reactor construction practices
- Advanced reactor programs
- Resolution of materials problems
- Reactor safety research
- Back end of the fuel cycle
Conclusions of JTEC Panel

“The Japanese nuclear power program is like a nimble pedestrian crossing a street against a flashing yellow light. The traffic represents potential LWR safety and reliability problems and the traffic signal is public opinion. In the U. S., the pedestrian is not as nimble, the traffic is heavier, and the light is flashing red.”

Early Materials Issues in the Storage of Nuclear Wastes

- Legacy of Purex separation at Hanford ‘50s – ’70s
- $^{137}$Cs and $^{90}$Sr separation from rest of fps
- Encapsulation as solid salts – Pool storage
- Practical uses as sources of radioactivity or heat
- Recall because of capsule failures
- Movement to above-ground storage ~ 1999
The Hanford site circa 1950

Rail transport of spent fuel to reprocessing facility
Reprocessing Area

HANFORD OPERATIONS 1944-1972

spent fuel → nitric acid → fuel dissolution → UO₄ → U⁶⁺ → Pu⁴⁺ → Pu²⁺ → solvent extraction → Stripping solution → Pu conversion → PuO₂

Cs⁺ → conversion to solid CsCl → pour melt into capsules

Sr²⁺ → conversion to solid SrF₂ → pack powder into capsules

ion exchange separation of ¹³⁷Cs and ⁹⁰Sr

Tank Farm
Hot cell for CsCl melting and pouring into capsules

- Outer capsule
- Gap #1
- Inner capsule

2.5 kg CsCl
160 W
60 kCl

20 cm
6 cm

UCBNE-5113/GoNERI-0003
Uses of Encapsulated Radioactivity (1970s)

- $^{137}$CsCl capsules rented to commercial medical sterilization firms – each capsule yields ~ 20 kCi of 661 keV gamma rays ($^{137}$Cs decays to $^{137}$Ba)

- At medical sterilizer facility, capsule is stored in pool, lifted out for use

- $^{90}$SrF$_2$ capsules loaned as “terrestrial heat sources” – each capsule produces ~ 1kW as heat, but no external radiation ($^{90}$Sr decays $^{90}$Zr by pure beta emission)
Failure of CsCl capsules (late 1970s)

- capsule removed from pool to expose items to be sterilized to gamma radiation
- When deprived of cooling water, the temperature exceeded the solid-solid phase transition temperature, causing an 18% volume expansion
- When returned to pool, salt shrinks
- Solid flaked off and fell to bottom of capsule
- Large strains at lower end of capsule caused rupture of both inner and outer capsules

Capsule failure mechanism

DOE response:
recall all capsules; prepare for long-term storage
dry storage - 1998

For 50 yrs then shipped to Yucca Mtn

Dry storage modules
Corrosion of stainless steel by salt

- Pure CsCl does not react with steel
- But, the salt contains up to 40 mole % impurities (as chlorides) - reactive impurity salts can corrode steel:
  
  \[-2\text{CuCl} + \text{Ni} \rightarrow \text{2Cu} + \text{NiCl}_2\]
  \[-\text{NiCl}_2 + \text{Fe} \rightarrow \text{FeCl}_2 + \text{Ni}\]
  \[-3\text{FeCl}_2 + 2\text{Cr} \rightarrow 2\text{CrCl}_3 + 3\text{Fe}\]
  etc.

- But, the barium decay product of $^{137}\text{Cs}$ is a corrosion inhibitor:
  \[^{137}\text{CsCl} \rightarrow ^{137}\text{BaCl}\]

- BaCl can chemically reduce any impurity salt MCl$_n$:
  \[n\text{BaCl} + \text{MCl}_n \rightarrow n\text{BaCl}_2 + \text{M} \quad (n = 1, 2, 3)\]
<table>
<thead>
<tr>
<th>Impurity in salt</th>
<th>$\Delta G_f^o$ (500 K)</th>
<th>1975</th>
<th>2006</th>
<th>2056</th>
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<tbody>
<tr>
<td>CuCl</td>
<td>26</td>
<td>0.4</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>NiCl₂</td>
<td>28</td>
<td>1.0</td>
<td>0</td>
<td>0</td>
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<tr>
<td>FeCl₂</td>
<td>33</td>
<td>1.2</td>
<td>0</td>
<td>0</td>
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<td>PbCl₂</td>
<td>34</td>
<td>0.9</td>
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<td>SiCl₄</td>
<td>35</td>
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<tr>
<td>CrCl₃</td>
<td>35</td>
<td>3.7</td>
<td>3.2</td>
<td>0.5</td>
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<tr>
<td>B₂O₃</td>
<td>-</td>
<td>5.1</td>
<td>5.1</td>
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<td>AlCl₃</td>
<td>46</td>
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<td>MnCl₂</td>
<td>50</td>
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<td>MgCl₂</td>
<td>67</td>
<td>1.4</td>
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<tr>
<td>CaCl₂</td>
<td>87</td>
<td>3.4</td>
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<tr>
<td>NaCl</td>
<td>87</td>
<td>18.2</td>
<td>18.2</td>
<td>18.2</td>
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<tr>
<td>SrCl₂</td>
<td>90</td>
<td>0.3</td>
<td>0.3</td>
<td>0.3</td>
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<tr>
<td>BaCl₂</td>
<td>93</td>
<td>0.0</td>
<td>11.9</td>
<td>19.9</td>
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<tr>
<td>KCl</td>
<td>93</td>
<td>3.1</td>
<td>3.1</td>
<td>3.1</td>
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<tr>
<td>CsCl</td>
<td>94</td>
<td>58.3</td>
<td>46.4</td>
<td>38.4</td>
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<tr>
<td>Total moles in salt</td>
<td></td>
<td>100.0</td>
<td>95.1</td>
<td>92.4</td>
</tr>
</tbody>
</table>

**End result**

- 50-year above-ground dry storage of highly radioactive encapsulated $^{137}$Cs and $^{90}$Sr can be accomplished safely.

- The most severe challenge to capsule integrity is repeated cycling through the solid-solid phase change temperature.

- Corrosion is limited by the availability of reactive chlorides (or fluorides) and by reduction by $^{137}$Ba.
Fast-Spectrum Reactors

- The *Integral Fast Reactor* Program aims at:
  - Closed fuel cycle
  - Proliferation resistance
Safety demonstrations at EBR-II

- Loss of flow without scram from full power
  - Simulated the Chernobyl accident

- Loss of heat sink without scram from full power
  - Simulated the TMI-2 accident
Loss of Flow without SCRAM Test Sequence

- Establish 100% power
- Bypass loss-of-flow reactor safety system
- Turn off the pumps

---

Loss of Flow without Scram from 100% Power with 100 sec Pump Coastdown Time.
Loss of Heat Sink Without SCRAM

Test Sequence

- Establish 100% power
- “Stop” flow in the intermediate sodium loop
- Monitor the passive power reduction and the leveling of tank temperature

Loss-of-Heat Sink Without Scram
Basic Safety Characteristics

• inherent feedbacks enabled return to a coolable state following abnormal events. Reasons:
  – Thermal expansion of the core increased neutron leakage
  – low fuel temperature minimized Doppler feedback
• No operator intervention or engineered components
• passive shutdown achieved by:
  - high thermal conductivity of metallic fuel
  - large thermal inertia of the secondary sodium pool

Flowsheet for EBR-II Fuel Reprocessing

EBR-II Spent Fuel
2.5 Tonnes Treated
23 Tonnes Remaining
Electrorefiner for Driver Fuel

Cathode Processor
Separates Salt-Uranium electrorefiner Products
Casting Furnace produces Uranium Ingots

Cladding Hulls are Cast into Stainless Steel – Zry Waste Form
Summary of EBR-II effort

- Sodium-cooled, metal-fuel fast reactor has inherent safety features absent in LWRs
- Pyrochemical reprocessing is an effective means for producing compact wasteforms and recycling uranium
  - >99.9% recovery of plutonium, uranium, and minor actinides
  - Immobilization of fission product wastes in form suitable for deep geologic disposal
  - Minimization of waste volume;
  - Elimination hazardous wastes by recycling (i.e., Cd)
Experience and Views of Nuclear Reactor Studies

Yoshiaki Oka
Professor, the University of Tokyo

Outline
1. Reactor noise / Research reactors
2. Radiation transport / Fast reactor shielding / Boron neutron capture therapy
3. Power reactor concept and analysis / SCWR (Super LWR and Super Fast Reactor)
4. MPS (moving particle semi-implicit) method
5. Education and research program including nuclear socio-science (GoNERI program)

Joint international workshop "Nuclear technology and Society
UC Berkeley, January 7, 2008

Research reactor / Reactor noise (graduate study)

Reactor Noise: Fluctuation of (neutron) detector signals;
measurement of neutron flux and vibration of control rods at JRR-4 and KUR for identifying the noise source:
Power spectral density (PSD): $\Phi_n$, Fourier transform of correlation function

$$\Phi_n(\omega) = \varepsilon F_0 + \varepsilon^2 F_0 \frac{\nu(\nu-1)}{\nu} |G_o(\omega)|^2 + \varepsilon^2 F_0 |G(\omega)|^2 \Phi_p(\omega)$$

$\varepsilon$: detector efficiency, $F_0$: total fission rate, $\nu$: number of neutrons per fission, $G_o(\omega)$: zero-power reactivity transfer function, $G(\omega)$: at power reactivity transfer function, $\Phi_p(\omega)$: PSD of external reactivity perturbation.
4. Reactor noise measurement at KUR (Kyoto Univ. Reactor)

![Diagram of reactor components and neutron flux]

Table 1: Noise sources of KUR

<table>
<thead>
<tr>
<th>Flow rate</th>
<th>Power level</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Low</td>
</tr>
<tr>
<td>Natural circulation</td>
<td>Branching process of fission neutrons or random bombardment of neutrons to a detector</td>
</tr>
<tr>
<td>Forced circulation</td>
<td>Flow induced vibration of shim control rods</td>
</tr>
</tbody>
</table>

![Continued text and diagrams]
Fast Neutron Source Reactor, YAYOI, Univ. of Tokyo
started to work at Nuclear Engineering Research Laboratory in Tokai-mura
Prompt-critical operation of YAYOI engaged as the chief lici

Peak pulsed power: 1GW (Steady state power: 2kW)

Experiences

1. Understand research reactors and the management.
   Familiar with professors / managers / technicians.
2. Useful for the management operation of the research reactor YAYOI.
3. Utilized the experience as an examiner of chief operator of nuclear power plants in late years. Learned guidelines and practices of reactor operation / accident management etc.
4. Understand the importance of human management issues in reactor operation and safety: Broaden the understanding of safety.

2. Radiation transport and shielding

Coupled neutron and gamma-ray transport calculation

\[ \Omega \cdot \text{grad} \quad \phi(r, E, \Omega) + \Sigma_t(r, E) \phi(r, E, \Omega) \]

\[ = \int_{\hat{s}} d\Omega' \int dE' \Sigma_s(r, E' \rightarrow E, \Omega' \rightarrow \Omega) \phi(r, E', \Omega') + s(r, E, \Omega) \]

where \( \phi(r, E, \Omega) \) is angular neutron flux in space \( r \), energy \( E \) and direction of motion \( \Omega \). It is the product of neutron density \( n(r, E, \Omega) \) and neutron speed, \( v \). \( \Sigma_s \) is the macroscopic scattering cross section and \( \Sigma_t \) is the macroscopic total cross section.
Discrete ordinates method

The angular flux \( \phi \) is represented by the set of the value at each of these mesh directions, \( \phi_n, n = 1, \ldots, N \). The integral over \( \Omega \) in the transport equation becomes a summation

\[
\int d\Omega \phi_n(\Omega') = \sum_{n=1}^{N} \omega_n \phi_n(\Omega')
\]

where the \( \omega_n \) are quadrature weights for particular numerical integration scheme to handle the integral. Then the transport equation reduces to a set of \( N \) equations

\[
\Omega_n \cdot \nabla \phi_n + \Sigma_n \phi_n(\Gamma, E) = \sum_{n=1}^{N} \omega_n \int dE \Sigma_n(\Gamma, E' \rightarrow E, \Omega_n' \rightarrow \Omega_n) \phi_{n'}(\Gamma, E') + S_n(\Gamma, E)
\]

This is called "discrete ordinates transport equation" or "\( S_n \) transport equation".

The energy variable, \( E \) is discretized into \( G \) intervals called energy groups. The energy dependence of cross sections are averaged in the group by assuming the spectrum, \( \phi(E) \)

\[
\sum_{g}^{E_g} \frac{\sum_{\Gamma \in \Gamma} \phi(E)}{\int_{E_{g-1}}^{E_g} \phi(E) dE}
\]

This is called group constants. We apply this scheme to \( S_n \) equations and obtain the coupled set of equations for the group angular fluxes \( \phi_n^g \):

\[
\Omega_n \cdot \nabla \phi_n^g + \sum_{n=1}^{N} \omega_n \sum_{g'=1}^{G} \phi_n^{g'}(\Gamma) = \sum_{n=1}^{N} \omega_n \sum_{g'=1}^{G} \phi_n^{g'}(\Gamma) + S_n^g(\Gamma)
\]

\( n = 1, \ldots, N, \quad g = 1, \ldots, G \).
Neutron and gamma-ray coupled cross sections
neutron: 42 group, gamma-ray: 21 group

Fast reactor shielding
grid-plate shield experiments for Monju

NUCLEAR TECHNOLOGY VOL. 31 PP. 287-296 (1976)

UCBNE-5113/GoNERI-0003
Shapes of shield plugs and shield blocks

Accuracy of discrete ordinates transport calculation

Sources of errors and uncertainties
(1) design uncertainty
   (a) configuration and volume ratio of materials
   (c) sizes and locations of ducts, voids and holes
   (d) source neutron distribution
(2) uncertainty in cross section data
(3) approximations assumed in the calculation
   (a) approximation in modeling the geometry and materials
   (b) approximation in the numerical calculation method
(4) errors associated with computational methods
   (a) incomplete convergence of iterative solution
   (b) discretization of variables
   (c) Legendre polynomial expansion of transfer cross sections
   (d) multigroup approximation
History of reactor shielding

Development of nuclear submarine

J. Simpson "Nuclear Power From Underseas to Outer Space" p.45, ANS (1994)

Nautilus prototype in the sea tank at INL

Radiation leakage of NS “MUTSU”

design fault of neutron shielding in 1974, Japan
Gamma-ray shielding: high Z material (Lead)
Fast neutron shielding: moderator + secondary gamma ray shielding

before

After repair

polyethylene

Heavy Concrete

Lead

U1BNE-5113/GoNERI-0003
Radiation transport code development

**ORNL: S\textsubscript{n} code:** ANISN, DOT, DORT, TORT
  - Monte Carlo Code: MORSE,
  - good experimental validation

**LASL: S\textsubscript{n} code:** TWOTRAN, ONEDANT, TWODANT
  - Monte Carlo Code, MCNP,
  - good accuracy of continuous energy calculation

**UT contribution:** SUSD code
  - cross section sensitivity and unartainty analysis including energy and angular distribution of secondary neutrons (for fusion neurotics)

ORNL / TR-88/18(1988)
Design of epithermal neutron column for boron neutron capture therapy (BNCT)

BNCT: (1) selective loading of tumor tissue with B-10 chemical compound
(2) destroy tumor cell selectively by alpha particle of B-10(n, alpha)Li-7
Epithermal neutron irradiation: treat deep tumors in a brain

Dose-depth distribution in the phantom head irradiate at the thermal neutron column

Dose-depth distribution in the phantom head irradiate at the epithermal neutron column

Epithermal neutron irradiation column (left)
Thermal neutron irradiation column (right)
Experience and Lessons

1. Joint study with medical doctors radiation biologist, chemist (boron compound), veterinarian (animal irradiation), engineers (research reactors) broadened the views and knowledge on biological effects of radiation etc.

2. Experience of radiation transport calculation was useful for the BNCT. It was an unexpected research subject when starting reactor shielding study.

3. Not continuing the study of BNCT made the results not well-known now.
3. Power Reactor Concept:
   Super LWR and Super FR
SCWR: supercritical-pressure water cooled reactor

Outline
(1) Introduction
(2) Fuel and core design
(3) Plant control, Start-up and Stability
(4) Safety
(5) Economic potential
Evolution of boilers

Supercritical Water in the Power Industry

Coal-fired SC plants in the world and their performance

<table>
<thead>
<tr>
<th>Country / Region</th>
<th>Number of SC Units</th>
<th>Installed MW</th>
<th>Year</th>
<th>Subcritical</th>
<th>Supercritical</th>
</tr>
</thead>
<tbody>
<tr>
<td>U.S.A.</td>
<td>149</td>
<td>106,454</td>
<td>1993</td>
<td>82.0</td>
<td>89.8</td>
</tr>
<tr>
<td>Japan</td>
<td>108</td>
<td>67,900</td>
<td>1994</td>
<td>83.8</td>
<td>83.0</td>
</tr>
<tr>
<td>Eastern Europe</td>
<td>123</td>
<td>51,810</td>
<td>1995</td>
<td>83.7</td>
<td>84.7</td>
</tr>
<tr>
<td>Western Europe</td>
<td>53</td>
<td>29,310</td>
<td>1996</td>
<td>86.6</td>
<td>79.5</td>
</tr>
<tr>
<td>Other Countries</td>
<td>29</td>
<td>13,520</td>
<td>1997</td>
<td>88.5</td>
<td>90.3</td>
</tr>
<tr>
<td>TOTAL</td>
<td>462</td>
<td>268,994</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Source: World Bank Organization

Most new coal-fired power plants are supercritical.
**SC turbines are proven technology**

*Major vendors of SCW components include GE, Toshiba, Hitachi, MHI, B&W, Siemens*

**Density and specific heat of supercritical water (25 MPa)**

**Toshiba: 700 MWe (24MPa, 593/593° C)**

**MHI: 1000 MWe (24.5MPa, 600/600° C)**
Question:

What are the guidelines of concept development of supercritical-pressure light water cooled reactor?

**Guidelines of the development**

1. Utilize supercritical fossil-fired power plant and LWR technology
2. Minimize large scale-developments of major components
   (Keep the temperatures below the experience)
3. Pursue simplicity in design

- Pressure: 25 MPa
- Inlet: 280°C
- Outlet (average): 500°C
- Flow rate: 1/8 of BWR
At supercritical-pressure:

No boiling phenomena
No boiling transition / dryout / burn out
No critical heat flux

Q1: What limits the design?

Large axial density change:

Q2: How to moderate?

How to estimate maximum cladding temperature?
Fuel load and reload pattern

- 120 FAs of 1st, 2nd and 3rd cycle fuels and one 4th cycle FA
- 3rd cycle FAs which have lowest reactivity are loaded at the peripheral region of the core to reduce the neutron leakage
  - This low leakage core is possible by downward flow cooling in peripheral FAs

Control rod patterns

- X: withdrawn rate (X/40)  Blank box: complete withdrawal (X = 40)
- At the EOC, some CRs are slightly inserted to prevent a high axial power peak near the top of the core
  - Prevent a high MCST
Does the cladding temperature of 3D core calculation show the maximum temperature among fuel rods?

No!

Q3: How to evaluate peak cladding temperature of a fuel rod in a fuel assembly?
Q4: What is the effect of design uncertainty and engineering uncertainty on the peak cladding temperature?

Q5. Plant control system?
   How to control reactor power, steam (outlet coolant) temperature, and reactor pressure?
Q6: Plant Start-up system?

Q7: How to start-up the reactor and turbine?

Q8: What are limiting parameters during start-up?

Q9: When starting from subcritical-pressure dryout occurs inevitably in the once-through cycle reactor, How to deal with it?
**Linear Stability Analysis Code**
*(Supercritical pressure)*

- Neutron kinetics model
- Fuel rod heat transfer model
- Water rod heat transfer model
- Fuel channel thermal-hydraulic model
- Water rod thermal-hydraulic model
- Ex-core circulation model

**Neutron kinetics model**

- **point kinetics model** with six delayed neutron precursor groups
- Doppler and density reactivity feedback

\[
\frac{\partial n(t)}{\partial t} = \frac{\Delta \rho}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t)
\]

\[
\frac{\partial C_i(t)}{\partial t} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t)
\]

\[
\delta \Delta \rho(t) = \left( \frac{\partial \Delta \rho}{\partial T_{ave}} \right) \delta T_{ave}(t) + \left( \frac{\partial \Delta \rho}{\partial \rho} \right) \delta \rho(t)
\]
Fuel Rod Heat Transfer Model

- Only radial heat transfer is considered.

\[
\frac{\partial}{\partial t} (\rho_f C_p T_f) = \frac{1}{r} \frac{\partial}{\partial r} \left( r k_f \frac{\partial T_f}{\partial r} \right) + q^* \\
T_{f,\text{ave}} - T_s = \left( \frac{r_f + t_c}{r_f} \right) \left[ \frac{r_f}{4k_f} + \frac{1}{h_g} + \frac{t_c}{k_c} \right] q^* \\
q^*(r_c, t) = h_c (T_c - T) 
\]

Water Rod Heat Transfer Model

\[
T - T_w = \frac{N_f}{N_w} \sum Q_w \left[ \frac{1}{\pi D_w h_{s1}} + \frac{1}{\pi (D_w - 2t_{ws}) h_{s2}} \right] 
\]

Thermal–Hydraulic Model

(Fuel Channel and Water Rod)

- Single-channel Single-phase One-dimensional Model
- Forward finite difference method for axial nodalization

Mass Conservation:

\[
\frac{\partial \rho}{\partial t} + \frac{\partial (\rho u)}{\partial z} = 0 
\]

Energy Conservation:

\[
\frac{\partial (\rho h)}{\partial t} + \frac{\partial (\rho u h)}{\partial z} = \frac{P}{A} + q^* \frac{N_f}{N_w A_w} Q_w 
\]

Momentum Conservation:

\[
- \frac{\partial p}{\partial z} = \frac{\partial (\rho u)}{\partial t} + \frac{\partial (\rho u^2)}{\partial z} + \rho g \cos \theta + \frac{2 f}{D_h} \rho u^2 \\
f = 0.0791 \times \text{Re}^{-0.25} \quad \text{(Blasius equation)}
\]

State Equation:

\[
\rho = \rho(P, h) 
\]
Ex-core Circulation Model

Orifice Model:
\[ \Delta P = \frac{\zeta}{2} \frac{\rho u^2}{2} \]

Feedwater pump model:
\[ \Delta P = C_{pump} \rho u \]

Feedwater pipe model:
\[ \frac{dP}{dz} = \frac{d}{dt} \rho u + \frac{d}{dz} \rho u^2 + \frac{2f}{D} \rho u^2 \]

Exit valve model:
\[ \Delta P = \frac{\zeta}{2} \frac{\rho u^2}{2} \]

Sliding Pressure Startup Procedure
Linear Stability Analysis (for Supercritical Pressure)

Stability Criteria

The same stability criteria for BWR are used for SCLWR-H.

<table>
<thead>
<tr>
<th></th>
<th>Normal operating conditions</th>
<th>All operating conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal-hydraulic stability</td>
<td>Decay ratio $\leq 0.5$</td>
<td>Decay ratio $&lt; 1.0$</td>
</tr>
<tr>
<td></td>
<td>(damping ratio $\geq 0.11$)</td>
<td>(damping ratio $&gt; 0$)</td>
</tr>
<tr>
<td>Coupled neutronic thermal-hydraulic stability</td>
<td>Decay ratio $\leq 0.25$</td>
<td>Decay ratio $&lt; 1.0$</td>
</tr>
<tr>
<td></td>
<td>(damping ratio $\geq 0.22$)</td>
<td>(damping ratio $&gt; 0$)</td>
</tr>
</tbody>
</table>
Q10: What is the fundamental safety requirement / monitoring parameter for safety of LWR
No water level at supercritical-pressure

Q11: What is the fundamental safety requirement / monitoring parameter of super LWR (SCWR)
Q12 : How to determine the LPCI capacity?

Initiating events for safety analyses

| Type of abnormality | Transients | | Accidents |
|---------------------|------------|-------------------------|
| Decrease in core coolant flow rate | 1. Partial loss of reactor coolant flow |  | 1. Total loss of reactor coolant flow |
|                     | 2. Loss of offsite power               |  | 2. Reactor coolant pump seizure |
| Abnormality in reactor pressure | 3. Loss of turbine load | 4. Isolation of main steam line | 5. Pressure control system failure |
| LOCA                             | 5. Large LOCA                                |  | 6. Small LOCA |

UCBNE-5113/GoNERI-0003
Analysis code for supercritical-pressure

Mass conservation
Energy conservation
Momentum conservation
- downcomer / water rod
- average / hot channels
Radial heat transfer
- Oka-Koshizuka correlation
Point kinetics

Summary of safety analysis results

<table>
<thead>
<tr>
<th>Transients</th>
<th>Accidents</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Partial loss of reactor coolant flow</td>
<td>1. Total loss of reactor coolant flow</td>
</tr>
<tr>
<td>2. Loss of offsite power</td>
<td>2. Reactor coolant pump seizure</td>
</tr>
<tr>
<td>3. Loss of turbine load</td>
<td>3. CR ejection at full power</td>
</tr>
<tr>
<td>4. Isolation of main steam line</td>
<td>4. CR ejection at hot standby</td>
</tr>
<tr>
<td>5. Pressure control system failure</td>
<td>5. Large LOCA</td>
</tr>
<tr>
<td>7. Inadvertent startup of AFS</td>
<td></td>
</tr>
<tr>
<td>8. Reactor coolant flow control system failure</td>
<td></td>
</tr>
<tr>
<td>9. Uncontrolled CR withdrawal at normal operation</td>
<td></td>
</tr>
<tr>
<td>10. Uncontrolled CR withdrawal at startup</td>
<td></td>
</tr>
</tbody>
</table>
Scope of studies and Computer codes

1. Fuel and core
   - Single channel thermal hydraulics (SPROD), 3D coupled core neutron/thermal-hydraulic (SRAC-SPROD),
   - Coupled sub-channel analysis, Statistical thermal design method, Fuel rod behavior (FEMAXI-6), Data base of heat transfer coefficients of supercritical water

2. Plant system; Plant heat balance and thermal efficiency

3. Plant control

4. Safety; Transient and accident analysis at supercritical- and subcritical pressure, ATWS analysis, LOCA analysis (SCRELA)

5. Start-up (sliding-pressure and constant-pressure)

6. Stability (TH and core stabilities at supercritical and subcritical-pressure)

7. Probabilistic safety assessment

Q13 : How to determine containment vessel (CV) volume?
Comparison of containments

 Improvement of 1700M We Super LWR from 1350M We ABWR

<table>
<thead>
<tr>
<th></th>
<th>SCLWR-H</th>
<th>ABWR</th>
<th>improvement in %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal efficiency, %</td>
<td>44.0</td>
<td>34.5</td>
<td>28%</td>
</tr>
<tr>
<td>RPV weight, t</td>
<td>750</td>
<td>910</td>
<td>18%</td>
</tr>
<tr>
<td>CV volume, m3</td>
<td>7900</td>
<td>17000</td>
<td>54%</td>
</tr>
<tr>
<td>Steam line number</td>
<td>2</td>
<td>4</td>
<td>50%</td>
</tr>
<tr>
<td>Turbine speed, rpm</td>
<td>3000*</td>
<td>1500*</td>
<td>50%</td>
</tr>
<tr>
<td>Condenser</td>
<td>2</td>
<td>3</td>
<td>33%</td>
</tr>
</tbody>
</table>

*3600rpm and 1800rpm in the western Japan
Advantages

1. Experience in LWR and fossil fuel power plant technologies.
2. Major components are within the temperature experience
3. Single phase flow; easy to analyze.
4. Compatible with tight lattice fast reactor core
5. Good subject for reactor knowledge transfer to young generation: LWR design, analysis and safety

Experience and its benefits

1. Chair of LWR design improvement committee of METI for 15 years (1984-1999) become familiar with elements of LWR design.
2. Invite LWR experts, mostly graduates for discussion, hear LWR practices and study/analyze super LWR by ourselves.
3. Collaboration with the excellent associate professor: S. Koshizuka
4. Power plant visit: 50 in Japan/10 USA some plants in Europe and Russia, Japanese LWR factories visit.
4. MPS method

The first particle (gridless) method for calculating fragmentation and large deformation of incompressible medium such as water/solids

\document{\textit{MPS} \textit{(Moving Particle Semi-implicit) method}}

\textbf{Weight function \(w\)}

The particle interacts with its neighbors with the weight function.

\[
w(r) = \begin{cases} 
  \frac{r_e}{r} - 1 & 0 \leq r < r_e \\
  0 & r_e \leq r 
\end{cases}
\]

- \(r\) : distance between two particle
- \(r_e\) : radius of interaction area

Particle interaction models (1)

A gradient vector between two particles is simply defined by their positions and coordinates. The gradient vectors are weighted with the kernel function and averaged to obtain a gradient vector.

\[
\langle \nabla \phi \rangle = \frac{d}{n} \sum_{j=1}^{n} \frac{\phi_j - \phi_i}{|r_j - r_i|} \| r_j - r_i \| w(|r_j - r_i|)
\]

A gradient vector is defined by particles’ positions and coordinates ← Idea of Finite volume method

Particle interaction models (2)

Laplacian operator’s physical meaning is diffusion → we treat it as distribution

\[
\langle \nabla^2 \phi \rangle = \frac{2d}{\lambda} \sum_{j=1}^{n} (\phi_j - \phi_i) w(|r_j - r_i|)
\]

where \( \lambda = \sum_{j=1}^{n} |r_j - r_i|^2 w(|r_j - r_i|) \)
Governing equations discretized by MPS (Moving Particle Semi-implicit) method

<table>
<thead>
<tr>
<th>Governing equation</th>
<th>Particle interaction model</th>
<th>Particle interaction</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass conservation</td>
<td>divergence</td>
<td></td>
</tr>
<tr>
<td>( \frac{D\rho}{Dt} + \rho \nabla \cdot \mathbf{u} = 0 )</td>
<td>gradient</td>
<td></td>
</tr>
<tr>
<td>Momentum conservation</td>
<td>Energy conservation</td>
<td>Laplacian</td>
</tr>
<tr>
<td>( \frac{D\mathbf{u}}{Dt} = \frac{1}{\rho} \mathbf{p} + \nu \nabla^2 \mathbf{u} + \frac{1}{\rho} \mathbf{f} )</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \frac{Dh}{Dt} = k \nabla^2 T + S )</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Green function relates governing equation to particle interaction

Collapse of water column

MPS calculation
Comparison with experiment(1)

Comparison with experiment(2)
Collapse of water column

Flexible right wall (strong wall)

Collapse of water column

Flexible right wall (weak wall)
Sloshing (fluid-structure interaction)


Simulation of pool boiling

- **Fluid:** Water
- **Pressure:** Atmospheric pressure
- **Wall temperature:** 110°C
- **Bulk temperature:** 96°C
- **Initial bubble radius:** 0.3 mm
- **Contact angle:** 45°

Numerical result

- Calculational condition

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial bubble radius</td>
<td>50 μm</td>
</tr>
<tr>
<td>Initial fluid temperature</td>
<td>27 °C</td>
</tr>
<tr>
<td>Heat flux</td>
<td>2 MW/m²</td>
</tr>
</tbody>
</table>

Computer graphics of jelly simulations with particle method
Tofu (bean-curd)

Chalk
Lessons

1. Do not force an excellent researcher to study other subjects. Let him study

5. Education and Research Program including social science

1. Foundation of Two Nuclear Departments
2. Department of Nuclear Engineering and Management
3. Nuclear Professional School
4. Global COE Program “Nuclear Education and Research Initiative” (GoNERI)
1. Foundation of Two Nuclear Departments

The University of Tokyo

Graduate School of Engineering

Department of Quantum Engineering and Systems Science

Department of Nuclear Engineering and Management

Nuclear Professional School

Started in April of 2005 as the 20th and 21st departments of the Graduate School of Engineering

Operated and managed together

Cooperated in some studies

2. Department of Nuclear Engineering and Management

- Regular graduate school with master course and doctoral course
- Unique features to cover the social science aspects as well as the nuclear science and engineering

Consisting of three main research areas

Nuclear-Socio Engineering

Nuclear Energy

Radiation Application
3. Nuclear Professional School

**Purpose**
- To meet the needs of highly educated professionals such as chief licensed reactor engineers of nuclear power plants
- To establish comprehensive schooling of nuclear engineering

**Administration**
- Located in Tokai-mura
- A type of professional school like a law school
- Jointly operated with JAEA (Japan Atomic Energy Agency) and the University of Tokyo
- Taking the responsibility and converting Nuclear Engineering Research Laboratory (NERL) into the Nuclear Professional School

**Textbooks**
- Early textbooks:
  - Out of print,
  - Not well-covered progress and practice of the nuclear power utilization in these 30 years
- Nineteen textbooks:
  - Under preparation,
  - Including nuclear socio-engineering,
  - For the human resource development in Japan
4. Global COE program of MEXT (J SPS)
Nuclear Education and Research Initiative
Systematic Education and Research including Nuclear Energy Sociology

**Nuclear Energy Sociology**
What is Technology for Society?
In collaboration with people outside Univ.

**Radiation Application**
Therapy, diagnosis, biology, etc.
Spread in interdisciplinary fields: medicine, agriculture and so on

“We prepare next generation researchers to grasp the perspectives of complicated and divergent fields of nuclear energy.” - Dr. Yoshiaki OKA, Prof. UT, Program Leader -

4.1. Nuclear Energy Sociology

**Nuclear Law/Legislation**
- To pursue the issue of efficient yet safe nuclear energy
- To deal with the relation between technology and law

**Nuclear Non-Proliferation**
- To coexist with the peaceful use of nuclear energy
- To identify the technological and systematic problems

**Public Communication**
- To inform the public along with ways to improve general science and technological understanding
- To facilitate the education through civil lectures

**Method: Practical Education**
- Law/Legislation: Cooperation with Prof. Shiroyama of the Graduate School of Law and Public Policy
- Non-Proliferation: Collaboration with visiting professors, Internship to IAEA
- Communication: Collaboration with Visiting Professor and Public Relation Specialist

Harmonization of Society and Nuclear Technology

To produce Ph.D students with the following capabilities:
- To study the regulation system, to identify problems, and to find solutions
- To serve in policy making in the world with the necessary expertise
- To communicate with the public on social aspects of nuclear energy
4.2. Nuclear Energy

To develop University-of-Tokyo Originated and Japan Leading technologies for worldwide contribution

Three Key Subjects

Future Nuclear Energy System
- Important issues of nuclear energy utilization in past, present and future
- GCOE fund for researcher education, while other external funds mainly for R&D

Disposal of Radioactive Waste

Safe and Stable Operation

Light Water Reactor

Generation III & IV Nuclear Reactor

Fast Breeder Reactor

Nuclear Fusion

Recycling of Spent Fuel Disposal of Rad. Waste

Roadmap of Nuclear Energy R&D

Now 2020 2040 2060

4.3. Radiation Application

To develop interdisciplinary research in connection with medicine, biology, agriculture, physics, chemistry and pharmacology

R&D Medical Physics

Application of the capability and techniques into space-time control of radiation therapy, diagnosis and inspection by developing table-top accelerator for cancer treatment and micro-PET

Radiation Chemistry

Initial reaction stage R&D by highest performance pulse radiolysis device, and its application to radiobiology

Radiation Safety/Biology/Environment

Radiation safety, radiobiology and earth environment analysis by accelerator mass spectrometry, etc.

To lead world by Advanced R&D on Radiation Application

Key issues to spread:
- accelerator minimization
- diagnosis modality development

Table-top accelerator

Positron Emission Tomography

Radiation cancer therapy realizes better Quality of Life
What is definition of risk?
(definition of safety)

1. Reactor Engineer / Scientist
2. Socio-psychologist (Public)
3. Manager, CEO, Investor

Risk / Uncertainty of US Nuclear Power Business

- Waste disposal (Yucca)
- Disaster (Price-Anderson)
- Regulatory (NRC / DOE)
- Technology / Design
- Development / Plant siting
- Transmission availability
- Construction
- Commissioning
- Operating
- Fuel price / supply
- Demand
- Dispatch
- Lawsuit
Type of public communications

1. Risk communication
   communicate with public on safety (risk) of nuclear utilization (Regulatory body).

2. Crisis communication
   communication with media/public when accident happens (companies).

3. Public communication
   build positive image of nuclear utilization among public.
   Such as clean-air benefit of nuclear power (Nuclear Energy Institute (NEI) in USA is trying hard and looks successful).

Difficulty in public communication in Japan

- Reactor engineer / specialists did not understand the difference of “risk” definition between public and them.
- The problem cannot be solved without clear understanding of the problem / definition.
- Lack of communication specialists who know fundamentals of socio-psychology, communication techniques / experiences AND key elements of nuclear issues.
Purpose of education and research of nuclear-socio science at UT

- Reduce the risk / uncertainties of nuclear utilization for achieving level playing field in the competitive / de-regulated economic environment.
- Make technical people to understand the difficulty / complexity of public communication.

USNRC Risk Communication guidelines

- Failure of public hearing of Yucca Mountain in 1999 made NRC to create communication manager and the guidelines;
- NUREG/BR-0318)“Guideline for Internal RiskCommunication”[link](http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0318/)
Thank you for your attention.
Nuclear technology and society –
*a historian’s view*

For the UCBNE-UTNEM workshop
*Nuclear technology and society –
Needs for next generation*
6-9 January 2008
Berkeley, CA, USA

Cathryn Carson
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“Technology and society” –
*when nuclear engineers talk about this,*
*what has it historically evoked?*

- Our efforts are in the service of society.
- What we provide is a social good.
- But society has trouble accepting it.

* A relationship problem – what can we do about it? (Or could we please find someone else to deal with it and get back to what we do well?)

What strikes a historian about this? (1)
Framing the “technology and society” problem

- What’s this thing called society?
- Where do engineers fit within it?
- Who determines social needs or social goods?
- Where and when did we get this schema anyways?

Does it still work better than, say,

sociotechnical system?

What strikes a historian about this? (2)
What does this way of framing the problem obscure?

- Why is this boundary-drawing so historically fraught? – What legacy dynamics do you have to deal with?
- How have “social” concerns gotten built into the technology already? – Going to scale, regulatory science.

Assuming the world has changed ... what are smart ways of acting within a sociotechnical system?
- Do sociotechnical systems have different dynamics?
- Is the name of the game knowledge? Or trust?
How do you tell your history?
Observations from a curious outsider (1)

Scope and content

- It tends to cover your own lifespans.
- It's a narrative of change, and mostly a narrative of technical advance.
- But somehow it’s not quite a narrative of learning (puzzles me).
  - Caveat: internal vs. public.
  - Caveat: locally motivated deviations.
- It has some trouble with unexpected developments.

How do you tell your history?
Observations from a curious outsider (2)

Narrative structure

- It’s organized by landmarks and major moving forces.
- They’re mostly punctual and decontextualized.
- It marks out the present as distinct from the past.
- It always points to the future.
Combining Nuclear, Renewable, and Fossil Fuel Cycles For Sustainability

Charles Forsberg

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Oak Ridge National Laboratory
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Nuclear Technology and Society: Needs for Next Generation
University of California at Berkeley
Berkeley, California
Monday, January 7, 2008

Outline

Global Sustainability Goals
Combined Fuel Cycles
Nuclear-Fossil Liquid Fuels
Nuclear-Biomass Liquid Fuels
Nuclear-Renewable Electricity
Nuclear Energy Implications
Two Goals are Likely to Determine What is Required for Sustainability

No Crude Oil

No Climate Change

Traditional Sustainability Strategies
Treat Each Fuel Cycle Separately

Separate Fuel Cycles will not Eliminate Oil or Stop Climate Change
Combined Fuel Cycles are Required for Sustainability That has Major Nuclear Energy Implications

Examples of Combined Fuel Cycles
Example: Combined Nuclear-Fossil Liquid-Fuels Fuel Cycle

Underground Refining

Liquid-Fuels Fuel Cycle for Crude Oil
Conversion of Fossil Fuels to Liquid Fuels Requires Energy

Greenhouse Gas Releases and Energy Use in Fuel Processing Increase as Use Lower-Quality Feedstocks

An Alternative: Underground Refining

Produces Light Crude Oil While Sequestering Carbon From the Production and Refining Processes as Carbon

In-Situ Refining May Require Nuclear Heat Source
Nuclear-Heated In-Situ Oil-Shale Conversion Process

Nuclear Heat Avoids Greenhouse-Gas Releases from Oil Production

Example: Combined Nuclear-Biomass Liquid-Fuels Fuel Cycle

Process Energy from a Nuclear Reactor
Fuel Cycle for Liquid Fuels from Biomass

Atmospheric Carbon Dioxide

No Net Greenhouse Gas Emissions

Liquid Fuels

Cars, Trucks, and Planes

Biomass Production, Transport, and Fuel Factories Use Energy

Atmospheric Carbon Dioxide

Liquid Fuels

Cars, Trucks, and Planes
1.3-Billion-Tons Biomass are Available per Year to Produce Liquid Fuels

Available Biomass in the United States without Significantly Impacting Food, Fiber, and Timber

Biomass Liquid-Fuel Yield Depends upon How the Biomass is Processed

Measured in Equivalent Barrels of Diesel Fuel/Day

Can Meet U.S. Liquid-Fuel Demand If an Outside Energy Source For Processing Biomass
The Nuclear-Hydrogen-Biomass Liquid-Fuel Cycle

Nuclear Energy With Biomass Liquid Fuels Could Replace Oil-Based Transport Fuels in the United States

Other Parts of the World Have Different Biomass Liquid-Fuel Options

Many Potential Feedstocks for Nuclear-Biomass Liquid Fuels Production
Example: Combined Nuclear-Renewable Electricity

Peak Electricity Production

Electricity Demand Varies with Time

Example: Daily Cycle
Large-Scale Renewable Electric Production may not be Viable without Electricity Storage

- Renewable electric output does not match electric demand
- Problems exist on windless days, cloudy days, and at night
- Low-cost backup power options are required

Fossil Fuels are Used Today to Match Electricity Demand with Production

- Fossil fuels are inexpensive to store (coal piles, oil tanks, etc.)
- Systems to convert fossil fuels to electricity have relatively low capital costs
- Carbon dioxide sequestration is likely to be very expensive for peak-load fossil-fueled plants
- If fossil fuel consumption is limited by greenhouse or cost constraints, what are the alternatives for peak power production?
Hydrogen Intermediate and Peak Electric System (HIPES)

<table>
<thead>
<tr>
<th>Energy Production Rate vs Time</th>
<th>Facility</th>
<th>Base Load</th>
<th>Heat and/or Electricity</th>
<th>2H₂O → 2H₂ + O₂</th>
<th>Underground Hydrogen/Oxygen (Optional) Storage</th>
<th>Fuel Cells, Steam Turbines, or Other Technology</th>
</tr>
</thead>
</table>

Relative Capital Cost/KW
- Nuclear Reactor: $$$$  
- H₂ Production: $  
- Underground Hydrogen/Oxygen (Optional) Storage: $  
- Fuel Cells, Steam Turbines, or Other Technology: $$

Nuclear Hydrogen Production Options

- Near term
  - Electrolysis
  - Electricity supply options
    - Base load
    - Night time and surplus renewables

- Longer term
  - High-temperature electrolysis
  - Hybrid
  - Thermochemical

Key Nuclear Hydrogen Characteristics (H₂, O₂, Heat, Centralized Delivery) are Independent of the Nuclear Hydrogen Technology
Bulk Hydrogen Storage is a Low-Cost Commercial Technology

- Chevron Phillips H₂ Clemens Terminal
- 160 x 1000 ft cylinder salt cavern
- Same technology used for natural gas
- In the United States, one-third of a year’s supply of natural gas is in 400 storage facilities in the fall

Use Same Technology for Oxygen Storage

Oxy-Hydrogen Turbine for Electricity

Low-Capital-Cost Efficient Conversion of H₂ and O₂ to Electricity for a Limited Number of Hours per Year

- High-temperature steam cycle
  - \(2\text{H}_2 + \text{O}_2 \rightarrow \text{Steam}\)
- Low cost
  - No boiler
  - High efficiency (70%)
- Unique feature: Direct production of high-pressure high-temperature steam
Oxy-Fuel Combustors Are Being Developed for Advanced Fossil Plants

- A hydrogen-oxygen combustor similar to natural gas-oxygen combustor
- CES test unit
  - 20 MW(t)
  - Pressures from 2.07 to 10.34 MPa
  - Combustion chamber temperature: 1760°C

HIPES may Enable Large-Scale Nuclear-Renewable Electricity

- HIPES strategy
  - Low-cost daily, weekly, and seasonal bulk H₂ and O₂ storage
  - Low-cost conversion to electricity
- Match production with demand
  - Renewables have highly variable power output
  - Can adjust to rapidly varying renewables output (full utilization)
Combined Fuel Cycles have Implications for Nuclear Energy

Requirements for Sustainable Nuclear Combined Cycles

Nuclear—Fossil—Biomass—Renewable

- Different nuclear inputs required for combined-cycle energy futures
  - Low-temperature steam
  - High-temperature heat
  - Hydrogen and oxygen

- Different options require different mixes of energy inputs

- Many combined fuel cycles require development of auxiliary technologies
Biomass to Ethanol and Diesel

Example Option Requiring Large Quantities of Low-Temperature Steam and Small Quantities of Hydrogen

Biomass (1.3 billion tons/year)

Cellulose (65-85% Biomass)

Lignin (15-35% Biomass)

Ethanol Plant

Steam Plant

Lignin Plant

Nuclear Reactor

Ethanol Plant

Ethanol

Gasoline/Diesel

Electricity

Biomass

50% Increase Liquid Fuel/Unit Biomass

Reactor Implications for Sustainable Nuclear Combined Cycles

Nuclear—Fossil—Biomass—Renewable

- Many applications may need smaller reactors
  - Underground refining heat demand per acre limits reactor size
  - Cost of biomass transport limits transport distances and thus the size of reactor
- Need for high-temperature reactors
  - Oil processing temperatures
  - Peak electricity production
- Need for reactors in different environments
  - Site security costs must be controlled
  - Safety systems must be simplified
Some Combined Cycles may Require Alternative Nuclear Reactor Designs

Requires Limits on the Size of Operating Crews and Security Forces

Traditional

“Fuel-Based” Advanced System

Emergency Systems

Containment

Security Fence

Fuel

Nuclear Systems Protects Fuel and Public

Fuel

Decay Heat Removed By Conduction

Doppler Shutdown

Fuel Protects Nuclear System and Public

Goal: “Indestructable” Fuel

Alternative Nuclear Reactor Designs may Require Alternative Fuel Cycles

Fuel Fabrication

Nuclear Reactor

Spent Fuel

Fuel Type

Characteristics

Rating

Magnox

Metal

Low

LWR

Oxide

Low Medium

LWR MOX

Refractory Oxide

High Medium

Coated Particle

Multi-layer

High

Fuel Characteristics

Reactor Safety

Non-Proliferation

Cost of Reprocessing

Waste Form Quality

Processing

Recycle

Direct Disposal

Repository

Waste

“Abuse-Resistant” Fuel Characteristics and Processing Cost are More Favorable for Direct Disposal
Conclusions

- Sustainability goals
  - No oil consumption
  - No climate change
- Sustainability will require integration of fossil, biomass, and nuclear fuel cycles with different nuclear products
  - Steam
  - High-temperature heat
  - Hydrogen
- Combined fossil, renewable, nuclear fuel cycles create requirements for nuclear reactors
- Some sustainability options may require reactors with “abuse-resistant” fuels

Questions

OAK RIDGE NATIONAL LABORATORY
U.S. DEPARTMENT OF ENERGY

UCBNE-5113/GoNERI-0003
—Abstract—

Combining Nuclear, Renewable, and Fossil Fuel Cycles For Sustainability

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The energy and chemical industries face two great sustainability challenges: the need to avoid climate change and the need to replace crude oil as the basis of our transport and chemical industries. These challenges can be met by changing and synergistically combining the fossil, biomass, renewable, and nuclear fuel cycles.

Fossil fuel cycles. Fossil fuel cycles must be changed to reduce greenhouse impacts and will require options beyond carbon-dioxide sequestration. In situ thermal cracking of heavy oils, oil shale, and coal may enable the production of high-quality transport fuels while sequestering the byproduct carbon from the production processes without moving it from the original underground deposits. Nuclear-fossil combined-cycle power plants may enable the large scale use of renewable electricity by matching electricity production to demand. However, these and other options require integration of high-temperature heat from nuclear reactors with fossil systems.

Biomass fuel cycles. The use of biomass for production of liquid fuels and chemicals avoids the release of greenhouse gases. However, biomass resources are insufficient to (1) meet liquid fuel demands and (2) provide the energy required to process biomass into liquid fuels and chemicals. For biomass to ultimately meet our needs for liquid fuels and chemicals, outside sources of heat and hydrogen are required for the production facilities with biomass limited to use as a feedstock to maximize liquid-fuels production per unit biomass.

Renewable electric fuel cycles. Nuclear energy can economically provide base-load but not peak-load electricity. Increased use of renewable electric systems implies variable electricity production that does not match electric demand. Today, peak electricity is produced using fossil fuels—an option that may not be viable if there are constraints on greenhouse gas emissions. Nuclear-produced hydrogen combined with underground hydrogen storage may create new methods to meet peak power production such as HIPES and NCCCs.

Nuclear fuel cycles. Nuclear energy can provide the stationary greenhouse-neutral steam, high-temperature heat and hydrogen for alternative biomass, fossil, and renewable fuel cycles. However, in many cases this will require high-temperature reactors, a potential change in reactor safety philosophy, and nuclear fuels that are nearly indestructible.
Biography: Charles Forsberg

Dr. Charles Forsberg is a Corporate Fellow at Oak Ridge National Laboratory, a Fellow of the American Nuclear Society, and recipient of the 2005 Robert E. Wilson Award from the American Institute of Chemical Engineers for outstanding chemical engineering contributions to nuclear energy, including his work in hydrogen production and nuclear-renewable energy futures. He received the American Nuclear Society special award for innovative nuclear reactor design and the Oak Ridge National Laboratory Engineer of the Year Award. Dr. Forsberg earned his bachelor's degree in chemical engineering from the University of Minnesota and his doctorate in Nuclear Engineering from MIT. He has been awarded 10 patents and has published over 200 papers.

Example: Combined Nuclear-Fossil-Renewable Electricity Fuel Cycle

Nuclear-Fossil Peak Electricity
Electricity Demand Varies with Time

- Variable electric demand met by fossil units (natural gas, etc.)
  - Low fuel-storage cost
  - Relatively low fossil-to-electricity capital costs
- What if greenhouse gas emission limits on fossil fuels?
- A capital-intensive nuclear-renewables electric system has no good method to match electricity production with demand

Nuclear-Combustion Combined Cycle (NCCC) System

High-Temperature Nuclear Heat with Natural Gas or Hydrogen

Gas Turbine Cycle

- Two Heat Sources
  - Heat from Reactor (Base-Load Electricity)
  - Fuel

Steam Turbine Cycle

- To Stack
- Feedwater Pump
- Condenser
- Steam Turbine
- Generator
Natural Gas or Hydrogen Can Be the Fuels for Peak Electricity Production

- **Natural gas**
  - Base-load electricity production uses nuclear heat
  - Natural gas used only for peak electricity production

- **Hydrogen**
  - Hydrogen produced during periods of low electricity demand (electrolysis or other technology)
  - Hydrogen stored in underground storage systems like natural gas
  - Hydrogen used only for peak power production

An NCCC Plant has Fast Response Times

- **Key characteristics**
  - Air is heated above the auto-ignition temperature so any air-fuel ratio is combustible (Dial-in power levels)
  - Compressor operates at constant speed and powered by nuclear heat—no additional compressor inertia or load with increased electricity production

- **Theoretical response speed limited by:**
  - Valve opening speed
  - Flight time from injector to the gas turbine
An NCCC may Enable Large-Scale Nuclear-Renewable Electricity

- Reduce fossil greenhouse gas releases
  - Only used for peak power production
- Match production with demand
  - Solar and some other renewables have highly variable power output
  - Can adjust to rapidly varying renewables output (full utilization)

Potential Requirements for Small Dispersed-Reactor Fuel

<table>
<thead>
<tr>
<th>Property</th>
<th>Basis</th>
<th>Technical Fuel Implication</th>
</tr>
</thead>
<tbody>
<tr>
<td>High-Temperature</td>
<td>Prevent accident releases</td>
<td>Refractory</td>
</tr>
<tr>
<td>Chemical Resistant</td>
<td>Prevent accident releases</td>
<td>Multilayer resistance to oxidation and reduction</td>
</tr>
<tr>
<td>Stress resistance</td>
<td>Avoid structural failure</td>
<td>Encapsulated fuel; matrix for stress</td>
</tr>
<tr>
<td>Shock resistant</td>
<td>Accident and assault</td>
<td>Encapsulated fuel with inert matrix to withstand shock damage to fuel</td>
</tr>
</tbody>
</table>
Comparison of Traditional Nuclear Fuels and “Abuse-Resistant” Fuels

<table>
<thead>
<tr>
<th>Property</th>
<th>Traditional Fuel (Magnox, LWR)</th>
<th>Abuse-Resistant Fuel (Coated Particle, etc.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Fraction</td>
<td>65–95%</td>
<td>&lt;10%</td>
</tr>
<tr>
<td>Reactor Safety</td>
<td>Reactor systems protect fuel</td>
<td>Fuel is the safety system</td>
</tr>
<tr>
<td>Reprocessing</td>
<td>Simple</td>
<td>Fuel designed to be indestructible</td>
</tr>
<tr>
<td>Security</td>
<td>Not a design consideration</td>
<td>Intrinsic resistance to abuse</td>
</tr>
</tbody>
</table>

Abuse-resistant fuel properties make such fuel: (1) expensive to recycle and (2) an excellent waste form.
Plenary Session 2:
Progress and Challenges in Nuclear Technologies - Safety and Regulations -

Chaired by
Prof. Y. Oka
UTNEM

1:00 PM – 3:00 PM
January 7, 2008
Current and GNEP Perspectives: Issues of High-Level Nuclear Waste Management

T. Ahn, S. Whaley and A. Murray
U.S. Nuclear Regulatory Commission
Washington, D.C., U.S.A.

International Workshop, Nuclear Technology and Society – Needs for Next Generation
University of California, Berkeley, CA
January 6 – 8, 2008

Purpose

• Present a summary of current status of Global Nuclear Energy Partnership (GNEP)

• Present a summary of current U.S. high-level waste (HLW) management: risk-informed performance-based regulation for the safety during operations and long-term waste isolation in HLW management

• Present issues associated with the engineered barrier system including waste package and waste form

• Present GNEP Perspectives in HLW management
Outline

• Purpose
• Overview and Summary of GNEP
• Current HLW Management: NRC Pre-closure Safety Analysis
• Current HLW Management: NRC Post-closure Total-system Performance Assessment and Engineered Barrier System
• GNEP Perspectives in HLW
• Summary
• Materials Engineer at NRC

Global Nuclear Energy Partnership (GNEP) Overview
Overview and Summary of GNEP

- Compare waste management aspects of
  - Open Fuel Cycle
  - Closed Fuel Cycle (including GNEP)
- Found
  - Large quantities of spent nuclear fuel (SNF) exist, with inventory growing
  - Closed Cycles reduce repository requirements
  - Large facilities needed (1,000 MTIHM/yr)
  - Long timeframes for working and cycling needed to eliminate SNF inventory (decades)
- Conclude
  - Closed Cycle provides waste management benefits, especially HLW
  - Potential GNEP benefits significant, but more program definition and regulatory changes needed

Inventory of Power Reactor SNF

- World-Wide
  - Generation: about 11,000 MTIHM/yr, 94% of SNF is uranium and not a waste
  - Inventory: about 200,000 MTIHM
  - About 50/50 Open/Closed Cycles
- United States (NRC, 2007)
  - Generation: about 2,000 – 2,500 MTIHM/yr
  - Inventory: 55,000 MTIHM (2005)
  - Disposition strategy: Open Cycle
U.S. SNF Inventory and Projected Repository Space

Spent Fuel Projections From Existing Reactors
(Assumes 100 LWR/100 GWe basis)

Statutory Capacity Essentially Filled Now

Impact of DOE SNF adds about 7,000 MTIHM
Assumes current DOE approach – one level, heat conduction only

U.S. Currently Uses an “Open” Fuel Cycle

No Recycle – SNF Directly To Disposal
Current SNF Discharges Range 2,000-2,500 MTHM Annually
Use 2,500 MTHM Per Year As Basis
Closed Cycle – Conventional Reprocessing

- Reprocessing
  - Proven Technology
  - Optimized Purex solvent extraction process
- Recycles as Mixed Oxide (MOX) fuel in Light-Water Reactor (LWR)
  - Proven technology
  - One cycle as MOX
- MOX SNF to repository
  - assumption – could be stored, disposed or recycled

Closed Nuclear Fuel Cycle

- Reprocess SNF
- Recycle – MOX to LWRs
- Repository Disposal – HLW and MOX SNF
- Current SNF Discharges Range 2,000-2,500 MTHM Annually
- Use 2,500 MTHM Per Year As Basis Reprocess SNF
Current U.S. HLW Management:
Risk-informed Performance-based Regulation

View Looking Down Exploratory Studies Facility
(DOE, 2002)
Pre-closure Safety Analysis:
selected DOE’s surface facilities

- Initial Operating Capability (DOE, 2007):
  aging pad, cask receipt security station, railcar staging area, truck staging area, wet handling facility, initial handling facility, canister receipt and closure facility 1, central control center facility

- Full Operating Capability (DOE, 2007):
  Phase 2 – receipt facility
  Phase 3 – aging pad, canister receipt and closure facility 2
  Phase 4 – canister receipt and closure facility 3

Pre-closure Safety Analysis: processes

- Areas of Consideration: TAD (transportation, aging and disposal) canister; wet handling facility; dry handling facility; ventilation; on-site transportation; aging facility; waste package emplacement (DOE, 2007)

- Initiating Event and Event Sequences

- Normal and Accident Conditions (internal and external Category 1 and Category 2)

- Design Mitigation

- Worker Dose

- Public Dose

- Pre-closure Safety Analysis (PCSA)
Pre-closure Safety Analysis:
canister drop

- About 90% of SNF in TAD canister and existing canisters will not be opened at the potential Yucca Mountain repository; about 10% of SNF will be packaged in SNF pool at wet handling facility (DOE, 2007).

- Canisters handled inside canister handling building may need robust integrity against drop or collision.

- Drop height, canister fabrication/welding defects, canister and internal materials may be important parameters in assessing potential radionuclide release.

- Failed canister may be partly credited in the PCSA compliance assessment.

Pre-closure Safety Analysis:
source term

- Release Fraction – impact energy (an example in Sanders, et al., 1992), oxidation from UO₂ to U₃O₈, high-burnup (radionuclide inventory, rim, and cladding)

- Leak Path Factor – HEPA efficiency, stack height/mixing volume, building discharge
Post-closure Total-system Performance Assessment: repository schematics

(DOE, 2002)

Post-closure Total-system Performance Assessment: engineered barrier system

(DOE, 2002)
Post-closure Total-system Performance Assessment: processes

- Normal Processes
  - Climate
  - Unsaturated Zone Flow
  - Effects of Decay Heat on Water Movement
  - In-Drift Physical and Chemical Environment
  - Water Diversion Performance of the Engineered Barriers
  - Waste Package and Drip Shield Degradation
  - Waste Form Degradation and Radionuclide Release
  - Engineered Barrier System Transport
  - Unsaturated Zone Transport
  - Saturated Zone Flow
  - Biosphere

- Feature, Event and Process (FEP) Analysis

- Normal and Disruptive Scenarios

- Design Mitigation of Engineered Barrier System

- Public Dose: individual, groundwater, human intrusion

- Total-system Performance Assessment (TPA)

Post-closure Total-system Performance Assessment: repository conditions

- Persistence of long-term passive film in general corrosion (I, II, and longer period)
- Dust deliquescence corrosion (I)
- Seepage water brines - crevice corrosion (II)
- Microbially influenced corrosion (II, and longer period)
- Hydrogen effects on Titanium (II, and longer period)
- Mechanical failure by drift degradation and seismicity (I, II, and longer period)

(After Pensado, 2006; 100 ºC = 212 ºF)
Post-closure Total-system Performance Assessment: mechanical failure

- Mechanical failure – Drip Shield/Waste Package interaction, rock rubble (Gute, et al., 2003)

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Post-closure Total-system Performance Assessment: persistence of passive film

Uncertainties of passive film stability affect long-term general corrosion rates

- Passive film stability is primarily affected by changes in
  - chemical composition
  - microstructure
  - thickness

A Cross-Sectional View of a Solution Annealed Alloy 22 Substrate (Orme, 2005)
Post-closure Total-system Performance Assessment: Radionuclide Release

- Matrix (i.e., irradiated UO₂) Dissolution – congruent release of Tc-99 and I-129 is principal contribution to dose for 10,000 year; Instant Release from grain boundary and gap inventory (e.g., C-14, Tc-99, I-129, Cs-137 and Sr-90)

- Colloid Release – Pu-(239+240) is a major contributor to the early peak dose

- Np-237 Release – an important contributor to the ultimate peak dose

Post-closure Total-system Performance Assessment: colloid release

Bates, et al. (1992)

Major carrier of Pu-(239 + 240)

- Waste form colloids: radioactive alteration product colloids, true colloids
- Pseudo-colloids: reversible and irreversible sorption on nonradioactive colloids
- Nonradioactive colloids: groundwater colloids, iron colloids from WP internal corrosion
GNEP Perspectives in HLW

- HLW Glass (an example)
  - TAD canister
  - pre-closure source term
  - waste package and drip shield performance
  - Tc-99 and I-129 release

Summary

- Summarized current status of GNEP, including overview, role and option of NRC, inventory of power reactor SNF, and closed nuclear fuel cycle
- Summarized PCSA and TPA processes for current U.S. HLW management: risk-informed performance-based regulation for the safety during operations and long-term waste isolation in HLW management
- Presented issues associated with the performance of the engineered barrier system, including canister drop, pre-closure source term, repository environment, corrosion and mechanical failure of waste package, and post-closure radionuclide release
- Presented GNEP Perspectives in HLW
Materials Engineer at NRC (T. Ahn)

- Understanding of current research issues in materials science: characteristics of spent nuclear fuel, cladding performance, aqueous corrosion of alloys, colloid chemistry, mechanical behavior of materials
- Process-level performance modeling of the above issues, and model abstraction for probabilistic system risk assessments (PCSA and TPA), sensitivity analyses of PCSA and TPA
- Issue resolution with the Department of Energy (DOE), preparation of Yucca Mountain Review Plan (YMRP), potential licensing review of DOE’s Yucca Mountain Project, potential preparation of Safety Evaluation Report and participation in hearing
- Participation in consensus standardization (e.g., ASTM) in nuclear material performance
- Participation in risk-informed performance-based rule making

Materials Engineer at NRC (Continued)

- Directing materials research group at the Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, Texas
- Participation in establishing the technical specs and the NRC’s endorsement for the U.S. vitrification project of liquid HLW
- Consultation of various material issues in transportation and interim storage of SNF, decommissioning, fuel cycle, and nuclear reactor
- Mentoring of coop students and interns from various universities
- Participation in international collaboration/exchange in nuclear materials research with Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA), International Atomic Energy Agency (IAEA), European Commission and Japanese laboratories
- Participation in various national and international conferences/journal publications in HLW as chair, organizer, and invited speaker/writer; and service for research proposal evaluation as a U.S. national committee member
Disclaimer

The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of a license application for a geological repository at Yucca Mountain.

References


C. A. Orme, The Passive Film on Alloy 22, UCRL-TR-215277, Lawrence Livermore National Laboratory, Livermore, CA, 2005


U.S. Radioactive Waste Management Policy and the Yucca Mountain Project

Presented to:
“Nuclear Technology and Society – Needs for the Next Generation” University of California at Berkeley and University of Tokyo Joint International Workshop

Presented by:
Robin L. Sampson
Disposal Operations Office
Office of Civilian Radioactive Waste Management
U.S. Department of Energy

January 7, 2008
Berkeley, CA

Outline

- History
- Yucca Mountain
- Current & Near Term Steps
- Why Yucca is Important
History of US Nuclear Waste Policy

In 1953, President Dwight D. Eisenhower addressed the United Nations, General Assembly. His speech, Atoms for Peace, called on all world leaders to move toward peaceful uses of nuclear technology.

In 2002, Congress passed a Joint Resolution and President George W. Bush signed into law, (PL107-200) which approved Yucca Mountain as the future site for the national repository for spent nuclear fuel & high-level radioactive waste.

U.S. Nuclear Waste Policy and Yucca Mountain

- 1957: National Academies of Science (NAS) supports deep geologic disposal
- 1970s: Nat’s screening & search for potential repositories
U.S. Nuclear Waste Policy and Yucca Mountain

DOE identifies 9 POTENTIAL SITES

1983

1984

1985

1986

1987

NWPA Amended: Act mandates ONE SITE for characterization

3 SITES Approved for Further Study

Secretary of Energy NOMINATES 5 SITES

1982

EPA to set radiation protection standard

1986

1988

1992

1998

DOE issues Viability Assessment of Yucca Mountain

2002

Site Characterization Studies Selection

Five Year Decision Process

9 Potential Sites

5 Nominated Sites

3 Presidentially Approved Sites

1 Site Identified by Statute

Yucca Mountain Site

- **Site Location**
  - Nye County, NV; average population of ~1 person/km²
  - 90 miles (150 km) northwest of Las Vegas, NV

- **Geology**
  - Ridge composed of layers of volcanic tuff deposited by neighboring volcanic eruptions between 11 and 14 million years ago.
  - All volcanic activity ceased millions of years ago.
  - Unsaturated Zone – air and water in rock pores control gas flows and infiltration of water

- **Climate**
  - Arid climate, mean temperature ~ 63°F (17°C)
  - Less than 7.5 in (19 cm) average yearly rain fall
Transportation, Aging, and Disposal

- Transport, Aging and Disposal (TAD) canistered system approach
- TAD system offers several technical benefits:
  - Support the standardization of SNF storage, transport, aging and disposal packaging, allowing integration of SNF handling operations
  - Utilize utility fuel handling experience in packaging SNF
  - Simplify DOE operations and minimize redundant handling of bare SNF assemblies at the repository
  - Reduce low-level waste production and worker radiation exposure
  - Reduce complexity and cost of DOE facilities

Repository Reference Design Concept*
Subsurface Reference Design*

* Picture is conceptual only and does not reflect current repository design

Emplacement Drift
Supplemental Environmental Impact Statements

- Final Environmental Impact Statement (FEIS) completed 2002
- Supplemental Environmental Impact Statement (SEIS)
  - Describes repository design developments since 2002
  - Draft SEIS for public review and comment issued in October 2007
- Rail Alignment Environmental Impact Statement (RA-EIS)
  - Analyzes the alignment, construction, and operation of a rail line to Yucca Mountain
  - Draft Rail Alignment EIS for public review and comment issued in October 2007
- Both environmental impact statements will be completed by June 2008
Licensing Support Network

- **NRC Licensing Support Network (LSN)**
  - Publicly accessible and searchable via Internet 24/7
  - All parties, including DOE, make documentary material available electronically
  - These documentary materials are available for use in the Yucca Mountain licensing proceeding
  - DOE Certified its document collection in October 2007
  - All parties must certify in accordance with NRC regulations

NRC REGULATION (10 CFR 63)

- **Is risk-informed and performance-based regulation**
  - Risk is the probability of exceeding the regulatory limit for worker and public dose
  - “Risk-informed” means that risk is one of the important factors that NRC will use to make a determination of acceptability for construction and operation
- **Requires DOE to determine acceptability of systems, structures and components based on a risk-informed evaluation for site-identified event sequences**
- **Calculation of probability/frequency of event sequences**
  - Uses probabilistic risk assessment methodology and techniques
  - Determines and justifies probability/frequency of initiating events
  - Determines and justifies probability of failure (reliability) of systems, structures, and components
Why Yucca Mountain is Important

- National Security
  - Support continued operations of the Navy's principal combat vessels.
- Nuclear Non-Proliferation
  - Dispose of surplus weapons-grade plutonium.
- Energy and Economic Security
  - Maintain nuclear energy option that supplies 20% of our electricity needs to sustain present and future economic security.
- Homeland Security
  - Consolidate nuclear materials presently stored at sites within 75 miles of 162 million Americans.
- Environmental Protection
  - Ensure environmentally sound disposition of our commercial and defense wastes.

Nuclear Power in the United States

America produces approximately 2000 MTU of SNF a year

There are currently 104 operating commercial nuclear reactors throughout the United States and over 36 research and training reactors that create nuclear waste.

There are currently 121 sites in 39 states which contains spent nuclear fuel and high-level waste destined for geologic disposal.

55,700 metric tons of spent fuel existed in December 2006; 119,000 metric tons of spent fuel projected by 2035.

At Least 21 Applications Expected to the Nuclear Regulatory Commission Between 2008-2010 for 32 New Reactor Units.
Conclusions

- **United States Law** requires the permanent disposition of spent nuclear fuel and highly radioactive wastes while protecting public health and safety and preserving the environment
  - We are responsible for disposing of our nuclear wastes while protecting human health and safety, and preserving the environment
  - Responsibility is shared by the Government, the Utilities, the Public

- Engineering, design, and system understanding of the proposed Yucca Mountain repository is sound and ready to support the License Application.

Conclusions

- Essential to 21st century energy security, National Security domestically and internationally, protection of the environment

- The US repository program balances complex technical, political, societal issues

- Ultimate repository success requires progress in many areas – in addition to the technical and operational challenges, the next generation will face many political and social obstacles that have no clear solutions
THE YUCCA MOUNTAIN PROJECT

For More Information:

Office of Civilian Radioactive Waste Management (OCRWM)
US Department of Energy, Room 5A-085
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Washington, D.C. 20585
(202) 586-5321

web:  http://www.ocrwm.doe.gov
Technology Innovation in Instrumentation and Control System for Nuclear Power Plant

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Contents

1. Introduction
2. Core Operation and Management System (COMS) for Nuclear Power Plant
3. Computer-Based Console for Nuclear Power Plant Trial (COCONUT)
4. Lessons Learned from R&D Experience
5. Conclusion
**Introduction**

### Achievement #1

**BWR Core Performance Calculation System**

*BWR: Boiling Water Reactor*

*World leading & EPRI collaboration*

On-line monitoring of power distribution in reactor core by physical-model-based simulation program

---

### Achievement #2

**ABWR Instrumentation and Control System**

*ABWR: Advanced BWR*

*One-man control prototype console*

Enlargement of automatic operation

Digitalization of control console
Core Performance Calculation Program

Constitution

- High accuracy
- Short computing time

Core performance calculation program

Flowchart and Models

Fig. 3 BWR core configuration

Fig. 4 Convergence calculation of the whole core three-dimensional nuclear-thermal-hydraulic mode.

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Core Performance Calculation Program

Adaptive capability

Parameter optimization

1. Adjustable parameters
   • related to reactivity transfer: 2 kernel mixing factors
   • related to boundary conditions: 4 albedos

2. Performance index

\[ J = \alpha \left( \text{Root mean square of TIP calculation error} \right) \]

\[ J = \frac{1}{L \cdot K} \sum_{i=1}^{L} \left( \text{TIP}_{i} - \text{TIP}_{i}^{m} \right)^{2} \]

\[ \text{TIP}_{i} : \text{calculated TIP at node } k \text{ of string } i \]

\[ \text{TIP}_{i}^{m} : \text{measured TIP at node } k \text{ of string } i \]

\[ L, K : \text{numbers of string and node} \]

3. Optimization algorithms
   Steepest gradient method
   Power correction factor calculation
   Adaptive capability

Core performance calculation program

Kernel mixing factors
- vertical: \( Q_v \)
- horizontal: \( Q_h \)

Albedo:
- vertical: \( a_v \)
- core top: \( a_t \)
- core bottom: \( a_b \)
- one surface: \( a_1 \)
- horizontal: \( a_g \)

Optimally adjusted parameter values

Parameter optimization by steepest gradient method
Core Performance Calculation Program
Power correction factor calculation

Parameter optimization

1. Power correction factors
- related to model deficiencies: control rod dependent
- related to input constants: axial position dependent
- related to measurement errors: string dependent

2. Calculation steps
- Axial position dependent factor
- Control rod dependent factor
- String dependent factor

Global component
Local component

Adaptive capability — Power correction factor calculation

Core Performance Calculation Program
Control rod dependent power correction

Control rod patterns
1 : Full in
2 : Full out (blank)

Control rod dependent power correction factor
Core Performance Calculation Program
Predicting mode calculation

1. Off-line simulation
   • using actual operating data
   • at 784 MWe (2281 MWt) BWR
   • during both steady states and transient states

2. Evaluation of calculation error
   • TIP calculation errors in monitoring and predicting models
   • Thermal power level calculation error in predicting mode

Evaluation of core performance calculation program

Core Performance Calculation Program
Evaluation of thermal power prediction

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Core Performance Calculation Program
Evaluation of power distribution

CONCLUSION

1. A core performance calculation program has been developed.
   - Two nodal coupling type models are included:
     - 3-D model with adaptive capability
     - 1-D model collapsed from 3-D model
   - Adaptive capability utilizes in-core neutron detector readings and consists of two steps:
     - Optimization of six adjustable parameters
     - Calculation of three power correction factors

2. Through offline simulations, this program has been verified to have enough accuracy to be installed in COMC.

Transition of Man-Machine Interfaces
in Main Control Room

UCBNE-5113/GoNERI-0003
NUCAMM-90  
(NUCAMM: Nuclear Power Plant Complex with Advanced Man-Machine Interface)

Selection of Important Alarms during Plant Transient

(a) Load rejection at 50% generating power

(b) Turbine trip at 50% generating power

(c) Load rejection at 100% generating power

(d) MSIV closure at 100% generating power
Systematic Problem Solving
Infinite Final Solution

Which approach is preferable?

You might be embarrassed by moving targets.

You might be required a little bit more effort and imagination.

Systematic Problem Solving
Technology Evolution S-curve

Every technology evolves in S-curve. Technology transfer is inevitable to reach higher degree.

Focus of invention is dependent on S-curve position.

To minimize cost
To increase reliability
To increase efficiency
To increase performance
To operate stably
To be operable
Two achievements were introduced:
- BWR Core Performance Calculation System for on-line monitoring of power distribution in reactor core by physical-model-based simulation program
- ABWR Instrumentation and Control System for enlargement of automatic operation, and digitalization of control console

Two approaches of Systematic Problem Solving in Technology Innovation were proposed:
- to search from GOAL to START
- to recognize Technology Evolution S-curve
Plenary Session 3:
Progress and Challenges in Nuclear Technologies - Innovative Technologies for Future Industries -

Chaired by

Prof. Per F. Peterson

UCBNE

3:20 PM – 5:20 PM
January 7, 2008
Nuclear Technologies for National Security

Kai Vetter
Nuclear Engineering, UC Berkeley
Lawrence Livermore National Laboratory

Nuclear Technology and Society – Needs for Next Generation –
UC Berkeley, January 6-8 2007

This work performed under the auspices of the U.S. Department of Energy by
Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344

Nuclear Technologies

- Nuclear technologies find a wide range of applications
  - Energy & power
  - Health
  - Oil and gas exploitation/ well logging
  - Forensics
  - Trace/ elemental analysis
  - Age dating
  - Food
  - Materials analysis (materials research, biology, chemistry,...)
  - Smoke detectors, pipe gauges, moisture/ density gauges, ...
  - Basic & applied research
  - ...
  - Security (nuclear detection methods)
    - Domestic and international
    - Nuclear and non-nuclear
Nuclear Detection Technologies for Non-Nuclear Materials

- Contraband detection
- Weapons
- Conventional (chemical) explosives
- Narcotics
- Forensics (e.g. mass spectrometry)
- Landmine detection

Active methods are required for this type of elemental analysis:
Detect nuclear fingerprint of materials by observing specific decays after external excitation e.g. by neutrons or photons

Focus:
Nuclear Detection Technologies for Nuclear Materials

- Already before, but dramatically more so after the events of 9/11/2001 the threat due to nuclear materials became of particular concern.
- Although a nuclear terrorist attack might be difficult and associated with a small probability, the impact will be disastrous, not necessarily in terms of casualties but economically and politically.

- To counter the threat of nuclear terrorism, a multi-layered approach is being pursued...
  - But first, what is the threat? What and where are the sources?
  - What is multi-layered approach? What are the concepts to counter nuclear terrorism?
  - What are current technologies?
  - How can new technologies help? What new technologies are being developed?
What is the nuclear threat?

- Terrorist acquiring nuclear materials and smuggling it to a point of interest for detonation (or other means of dispersion…)
- Problem: Nuclear/ radioactive materials are being used and are potentially available world wide
- We distinguish between types of nuclear threats:
  - Radiological Dispersive Devices (RDDs, dirty bombs)
    - Combine conventional explosives with nuclear materials (local dispersal of radioactivity with mainly socio-economical impact)
  - Nuclear Devices
    - Improvised Nuclear Devices (INDs) or Nuclear Weapon Devices consisting of Special Nuclear Materials (SNMs):
      - $^{235}\text{U}$ (Highly Enriched Uranium, HEU)
      - $^{239}\text{Pu}$ (Weapons Grade Plutonium, WGPu)

The international context: A layered, global defense architecture is required to prevent the misuse of nuclear and radiological materials

International efforts to secure nuclear materials are considered by most experts to be the most effective in preventing nuclear and radiological terrorism.
Safeguards is the front line for keeping SNM from getting to the wrong people

Interior-modal detection
CONUS entry
Air, Land, Sea

Domestic Nuclear Detection
Requires customs, border patrol, law enforcement, and other agents, at remote locations around the globe, to make snap decisions about whether an object is or contains a threat.

Multi-layered approach:
1st line of defense: Securing nuclear materials
2nd line of defense: Interdict materials transport

International Safeguards
Dominated by the need to verify the declarations made by States about their nuclear material activities.
Measurements are made by experts and other well-trained personnel, governed by protocols and various types of agreements.

The Challenge in Homeland Security
- Protect against Weapon of Mass Destruction (WMD)
- Identify and interdict potential threats before attack
- The Mandate: 100% screening at US Ports of Entry

Additional Challenge:
Multitude of legitimate and natural radioactive sources!
Countering the radiological and nuclear threats requires a broad-based approach

The core of the U.S. strategy continues to be the securing of nuclear materials, but technological advances can also have a significant impact.

**Intel Interdiction**
- Prevention/Preparation
- Crisis Management
- Recovery/Restoration
- Attribution/Retaliation

**Intel Detection**
- Signatures
- Data collection/management

**Detection System Architectures**
- Detector technology development
- Operational testbeds and pilot deployments

**Crisis Response Tools**
- Real-time, mobile diagnostics
- Modeling tools

**Forensics**
- Isotopics
- Data archiving/informatics
- Physical/chemical characterization

**Decontamination and Recovery**
- Contamination assessment
- Decontamination

**Medical Diagnostics and Treatment**
- Biodosimetry
- Effective treatments

**System Analysis:** Threats, tradeoff studies

**Current Implementations and detectors**

- Primary screening + secondary inspection

**Radiography**

- "Conventional" X-ray
- Gamma-Ray

**Portal Monitors**

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The current implementation of this strategy has operational limitations

The limited capacity to deal with nuisance (false) alarms reduces the probability of detecting a threat

The current implementation of this strategy also has serious technology limitations

- Primary portal monitors have no selectivity (isotopic discrimination)
- Secondary screening has poor selectivity and sensitivity
- Commercially available isotope identifiers are too small, too expensive, or too hard to use; most have poor selectivity and low reliability
- Shielding is hard to detect
- Shielded SNM, especially HEU, is hard to detect

Improvements:
- Sensitivity (improved signal-to-background)
- Selectivity (isotope identification, localization, spatial extension of source)
Domestic Nuclear Detection Office (DNDO) was created to address these problems.

Operations
- Screening at US Ports of Entry
- Other detection architectures

Testbeds & Pilot Programs
- Testing against standards
- T&E facilities

Technology Development
- Radiation detection
- Analytical techniques
- Responder tools

Analysis and assessment
- Threat assessment
- Systems analysis

Technology Commercialization

To address architecture gaps, DNDO’s R&D program is pursuing a range of technologies.

Applications
- Ubiquitous detection -- enable use of high-performance detectors anywhere, anytime we need them
- Cargo security -- develop comprehensive toolkit for screening cargo and negligibly impacting commerce
- Remote and mobile detection -- enable search, surveillance, and stand-off detection applications

Enabling Technologies
- Innovative detector materials -- transform capabilities of all applications through low-cost materials or improved fabrication
- Algorithms development -- develop advanced algorithms to enhance maximize capabilities to detect, identify, and locate threats
Intelligent Personnel Radiation Locators (IPRL) such as LLNL’s UltraPeRL will provide unprecedented ubiquitous capabilities

- Low-power, low-cost CdZnTe modules to be integrated with cell phones providing:
  - Isotope identification, directionality, neutron detection & GPS
  - Robust, low-power, aggressive form factor, cell phone communications

Portable, mechanically-cooled HPGe detection technology

- HPGe provides the “gold standard” in isotopic identification, but requires cooling to ~90K
- Low-power electromechanical coolers are now available and are further refined– liquid nitrogen is no longer required!
Active interrogation approaches for detection SNM in cargo containers

**Fast fission neutron detection with low-dose, 60-keV neutron interrogation**
- Hand-truck portable
- Low radiation dose due to low-energy neutron beam
- Neutrons create 2.2 MeV capture gammas in hydrogenous shielding
- Neutrons, 2.2 MeV gammas penetrate high-Z shielding

**Delayed gamma-ray detection with fast neutron interrogation**
- Truck relocatable
- Works in active mode, potentially high radiation dose
- Incoming 7 MeV neutrons have better penetration than 60 keV, 3+ MeV gammas penetrate high-Z shielding

**Nuclear fluorescence detection with a tunable x-ray beam**
- Fixed installation, potentially relocatable
- Works in active mode, potentially high radiation dose
- Fluorescence fingerprints may enable “simultaneous” detection of SNM and explosives and narcotics

7-MeV neutron interrogation of cargo containers uses delayed, high-energy $\gamma$-rays as SNM signature

**Interrogation system is packaged into 3 cargo containers for portability**
- Top container houses the neutron source directed down onto cargo
- Cargo is irradiated as it is towed through a gauntlet of detectors
- $\beta$-delayed fission product $\gamma$- and neutron radiation identifies the SNM
Gamma-ray imaging will provide important new capabilities for stand-off detection

Large, scintillator-based, 1-D, coded aperture imager

Compton imager, based on strip-detector technology

Gamma-ray imaging provides a way to distinguish a distant point source from slowly varying background

With large-area imager (LAI), we have demonstrated detection ranges well beyond non-imaging systems

- 1D coded aperture along with vehicle motion provide 2D imaging capability
- Coded apertures on either side of detector array enable imaging on both sides of vehicle

6 sigma detection of 1 mCi Cs-137 at 50 m and 20 mph
Gamma-ray imagers are detectors that separate radioactive objects from the local background.

Conventional detectors accept gamma-rays from all directions and can be overwhelmed by local backgrounds. Gamma-ray imagers isolate sources from each other and from the local background.

Observables:
- Count rate
- Energy spectrum

Power of imaging:
- Simulations of Uranium through 7 cm Fe
- 5 meters away
- Black = source + background
- Red = background

Today’s commercial detectors have limited ability to separate the source from background. An imager with 1 degree angular resolution provides significant separation from background.
Gamma-ray imaging will increase sensitivity by improving signal-to-background

Imaging can “count” on a pixel-by-pixel basis and can therefore distinguish between a local signal and global background to improve the signal-to-background ratio.

**Assumptions:**
- 500 source counts per minute in detector
- Detector:
  - 1 m² detector
  - 100% efficient
  - Use all $E \geq 90$ keV
- “Natural” background:
  - Ground emits 2 counts/cm²/s
  - In vacuum this is 1,200,000 counts/min into 1 m²
    → 3000 per bin for 400 bins in image
  - In air at 100 m background rate is about the same as in vacuum due to in-scatter

Detection probability at 100m with air attenuation for different numbers of spatial bins

Compton imaging as most promising approach for gamma-ray detection by passive means

- **Concept:**
  - Measure energies and positions of individual gamma-ray interactions
  - “Track” the gamma-ray, e.g. find the scattering sequence
  - Back-project cone of possible incident directions on 2D surface or 3D cube.
- **Advantages:**
  - Gamma-ray imaging without using a collimator
    - Large field-of-view
    - Compact and less weight
    - Signal-to-background improvement on event-by-event level

 USAGE NOTE: Overall, one step closer to ultimate gamma-ray detector (no gamma-ray left behind...)
Technical Approach: Compton Imaging with High-Resolution Si and Ge Detectors

- Large-volume, high-resolution Si+Ge detectors:
  - Excellent energy resolution for isotope identification and Compton imaging
  - High intrinsic and imaging efficiency
  - High Compton imaging sensitivity above 150 keV
- Double-sided strip detectors + digital signal processing:
  - Excellent position resolution in three-dimensions
  - Reduced number of readout channels (e.g. 76 channels for 86640 1mm³ voxels)
- Goal: Compact Compton Imager (CCI)

"Gold" standard for Compton imaging

We have built a second generation CCI system which provides high-sensitivity “real-time” imaging

- CCI-2 detection system
  - Two large HPGe doubles-sided strip detectors (DSSD) in one cryostat
  - Two large Si(Li) DSSD in one cryostat
  - Compact, high-bandwidth and resolution preamplifiers
  - Fully digital data acquisition system
  - State-of-the art graphical user interface to setup, monitor, display, and analyze data.
  - Realtime imaging and gating capabilities
- Full CCI-2 10x more efficient than our first generation system, CCI-1

Assembled CCI-2 instrument
Differentiation between DU and HEU employing 2nd generation Compact Compton Imager (CCI-2)

**Example of Recent Results:**
186 keV-gated images

| 400g DU @ 4ft | 2.5g HEU @ 9ft |

**DU+HEU+BGD Energy Spectrum**

- **DU**
- **HEU**
- **Background**

**CCI-2**

---

**DU-HEU Measurements: Gating on 186 keV & 1000 keV**
DU-HEU Measurements: Gating on 1000 keV

DU-HEU Measurements: Gating on 186 keV
3D Image Mapping of Radioisotopes

- We have combined a visual 3D mapping system with our Compact Compton Imager (here, 2 HPGe detectors)

![Image](image1.png)

A 3D Design Information Verification (DIV) laser range scanner and a Compact Compton Imager (CCI) gamma-ray camera on a mobile cart

3D Gamma-Ray and “Visual” Image Merging

- We have developed and demonstrated three-dimensional gamma-ray image reconstruction and the merging with three-dimensional visual object information

  - Identification of objects in three dimensions
    - Radio-isotope identification
    - Distribution of activity in 3D even behind walls, inside enclosures, etc.

Three-dimensional image reconstruction and data merging

![Image](image2.png)

3D imaging of extended $^{152}$Eu line source

Three-dimensional gamma-ray and “visual” image merging as new capability to identify and localize nuclear materials in search scenarios
“Real-time imaging”

- We have implemented and demonstrated real-time data processing and gamma-ray imaging with CCI
  - Real-time data processing consisting of event filtering, gamma-ray tracking, and gamma-ray image processing and display
  - Energy-image correlation and gating
  - Merging of gamma-ray and “visual” images in 4π
  - Compton image reconstruction

Real-time processing represents an important step towards a fully operational and deployable Compton imager.

“Static” Tomographic Compton Imaging

- Compton imaging enables 3D or tomographic imaging for objects in near field (Distance < Detector dimension/2)
- Experimental demonstration using CCI-1 and iterative Listmode-Maximum Likelihood image reconstruction:

2 spherical $^{113}$Sn sources (391 keV):
  - Diameter: Each 4 mm
  - Distance: 10 mm

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Next step… Electron tracking Compton imaging…

- To reduce the Compton “cone” to an “arc” the initial vertex of the Compton-scattered electron needs to be measured
- This would allow significantly increased contrast and sensitivity, particularly in complex radiation fields (assuming the efficiency and overall resolution can be maintained)
- Challenge: ~10 μm spatial and high energy resolution at least for first interaction

Many more technologies for 1st and 2nd line of defense are being pursued…

- Detection in and around nuclear facilities
  - For example, anti-neutrino detectors
  - Fuel rod counter in storage facilities
  - Ultra-high resolution gamma-ray spectroscopy
  - ...
- Detection during transport or storage
  - Specific signature evaluation, e.g. multiplicity measurements
- Detector materials research:
  - Cheap, rugged, high efficiency and high energy resolution at room-temperature
  - The impact of all individual technologies need to be maximized by combining them with complementary technologies and modalities and by using intelligence information
  - Foremost, however, are policies that needs to be agreed upon and put in place
Conclusions

• Many new nuclear technologies and concepts are being developed, demonstrated, and ultimately deployed increasing the global and national security...

• More work needs to be done to explore fundamental physics limits to increase sensitivity in the detection of nuclear materials

• Many other applications will benefit from these efforts

• While historically, technologies developed for basic and applied research were leveraged in nuclear security and non-proliferation, now it works also the other way around ...

• However, the technological advances need to be embedded in appropriate architectures to be established through international policies and agreements

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  ➢ Davor Protic, Thomas Krings

• UC Berkeley:
  ➢ Dan Chivers, Sara Matafirri

The Operational Strategy: From screening to reachback

100% Vehicle, cargo, people, etc.

- Primary Screening
  - No alarm
  - Natural or Legitimate
    - Release vehicle
    - SNM or on-site support req'd
  - Alarm
    - Inconclusive or confirmation Desired
      - Nuclear Incident Response Team

- Secondary Screening
  - Natural or Legitimate
    - Inconclusive or confirmation desired
      - Secondary Reachback

- Primary Reachback
  - Natural or Legitimate
    - SNM or on-site support req'd
  - Inconclusive or confirmation desired
    - Nuclear Assessment Program
Ageing management and maintenance of Japanese nuclear power plants and its academic roles

Taira Okita

Department of Quantum Engineering and Systems Science, the University of Tokyo

---

Brief introduction of Taira Okita

**Education Background**

1997.4 - 1999.3 : Master course, graduate school of Engineering, the University of Tokyo
Experimental studies for microstructural evolution under irradiation of nuclear materials.

TEM observations for the neutron-irradiated alloys.

2002.3 : Doctor of engineering
"Effects of Dose Rate on Irradiation Behavior in Structural Materials"

**Work Experience**

2002.8 - 2004.10 : Post doctoral researcher in LLNL, DOE NERI project
Simulations for the material strength and deformation mechanism

2005.3 - 2007.3 : Research assistant, Department of Quantum Engineering and Systems Science, the University of Tokyo
Experimental observations of nano-structure, nuclear fuel systems, multi-scale modeling

2007.4 - : Associate professor, Department of Quantum Engineering and Systems Science, the University of Tokyo
Maintenance of nuclear power plants, multi-scale modeling, construction of the data model
Outlines of this presentation

1. Ageing management and current issues of Japanese nuclear power plants.
   - Ageing management
   - R&D roadmap and industry-government-academia partnership

2. Academic roles for the ageing management.
   - The dose rate effect is the key parameter for the ageing management.
   - Experimental and modeling approach to dose rate effects

3. Summary

Nuclear Power Plants in Japan

![Graph showing electrical output of nuclear power plants in Japan over time](Sekimura, 2006)
Number of plants for ageing management technical evaluation

Increase in number of plants
From initial plants to standardized plants

R&D Roadmaps for Ageing Management and Safe Long Term Operation
developed by Atomic Energy Society of Japan in 2004-2005

To Keep Safety and Reliability of Nuclear Power Plants for Long Term Operation

1. Establishment of Information Basis
   - Database for Degradation of Materials
   - Systematic Ageing Management Program

2. Technical Development
   - Evaluation Technology for Degradation of Components
   - IASCC

3. Codes and Standards
   - Standardization of Ageing Management Procedures
   - RPV Radiation Embrittlement
   - Schemes To Apply New Techniques
   - Performance Index
   - Risk-based Maintenance

4. Systematic Maintenance
   - Optimization of Maintenance
   - Human Resources

Note: This figure is drawn assuming that all of the plants currently in service will continue their operation.
A Coordinating Committee on Ageing Management was established with members from industries, utilities, research organization and regulatory authorities.

NISA project for ageing management infrastructure improvement
FY2006 - FY 2010

Tasks for the University of Tokyo

- Comprehensive research on construction of a knowledge base of stratified ageing data and a standardization strategy for more advanced regulation.
- Research on earthquake-resistance safety evaluation for piping wastage.
- Research on the applicability of new technologies for monitoring and preserving ageing equipment.
- Study on irradiation embrittlement of Heat Affected Zone of reactor pressure vessels.
- Systematic understanding of IASCC mechanisms.
the dose rate effect is the key parameter for ageing management

6 major ageing degradation of the nuclear power plants

- Embrittlement of RPV
  - Dose rate effects
- Stress Corrosion Cracking
  - IASCC
  - Dose rate effects
- Fatigue
- Thinning of piping
- Insulation degradation of electrical cables
- Strength and shielding capacity degradation of concrete

• RPV steels are exposed to the long term irradiation with a very low dose rate. The total dose is not so high that we can reach the equivalent cumulative dose within a few days or a few minutes by other irradiation facilities.
• We have to clarify whether such a low dose irradiation with the low dose rate causes a serious embrittlement.

Physical aspects of the dose rate effects

Interaction between high energetic radiation particles and materials

- Macroscopic changes of the materials arises from the energy deposition from the radiation into the materials.
- The radiation effects of materials, or macroscopic changes have been considered to be a function of the deposited energy into the irradiated materials; exposure dose, or cumulative dose.

\[(\text{cumulative dose}) = (\text{irradiation time}) \times (\text{dose rate})\]

The microstructural evolution is dominated by diffusion and reaction rate, which are strong function of irradiation time, and dose rate. The resultant macroscopic change are strongly affected by the difference in dose rate.
the dose rate effect is the key parameter for ageing management

- We will obtain the database for the irradiation embrittlement and IASCC susceptibility by using the higher dose rate irradiation facility such as JMTR or ion accelerator.
- To validate the database obtained by the higher dose rates, we need to clarify the essential mechanisms of the dose rate effects, and to build up the model which incorporates such effects.
- With these models, we will be able to obtain the knowledge base of the dose rate effects from the database.
Neutron irradiation experiments

Previous studies for the effect of dose rates

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<td></td>
<td>Kileas</td>
<td>JMTR</td>
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<tr>
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<td>Yanagida</td>
<td>KUR</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td>Austenitic alloys</td>
<td>Garner et al.</td>
<td>EBR-II</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Neustroev et al.</td>
<td>BOR-60</td>
<td>&gt; 1 order</td>
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<tr>
<td></td>
<td>Leuthwaite et al.</td>
<td>DFR</td>
<td>&gt; 1 order</td>
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<tr>
<td></td>
<td>Garner et al.</td>
<td>BOR-60</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Garner et al.</td>
<td>BN-350</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Porter et al.</td>
<td>EBR-II</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Kruglov et al.</td>
<td>BR-10</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Koslov et al.</td>
<td>BN-600</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Seran et al.</td>
<td>Rapsodie</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Seran et al.</td>
<td>Phoenix</td>
<td>&gt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Grossbeck et al.</td>
<td>BR-2</td>
<td>&lt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Cole et al.</td>
<td>EBR-II</td>
<td>&lt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Walters et al.</td>
<td>EBR-II</td>
<td>&lt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Schneider et al.</td>
<td>Rapsodie</td>
<td>&lt; 1 order</td>
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<tr>
<td></td>
<td>Allen et al.</td>
<td>EBR-II</td>
<td>&lt; 1 order</td>
</tr>
<tr>
<td></td>
<td>Garner et al.</td>
<td>EBR-II</td>
<td>&lt; 2 orders</td>
</tr>
</tbody>
</table>

Irradiation at seven positions in, below and above core. Either 1 or 2 cycles of irradiation to achieve two dose levels.

Simplified theory to analyze dose rate effects by the rate-equations

Assumption

Only a mono-interstitial and vacancy can diffuse individually.

\[
\frac{\partial C_i}{\partial t} = P - R \cdot C_i C_v - Z_i S \cdot D_i C_i \\
\frac{\partial C_v}{\partial t} = P - R \cdot C_i C_v - Z_v S \cdot D_v C_v
\]

Production rate \(\propto (\text{Dose rate})\)

Recombination \(\uparrow\)

Sink \(\uparrow\)

High dose rate

Recombination dominant

\[C_i \propto (\text{dose rate})^{1/2}\]

\[C_v \propto (\text{dose rate})^{1/2}\]

Low dose rate

Sink dominant

\[C_i \propto (\text{dose rate})^{1} \cdot \text{S}^{-1}\]

\[C_v \propto (\text{dose rate})^{1} \cdot \text{S}^{-1}\]

There are no ways to measure the point defect concentrations.
Simplified theory for the dose rate dependence of the net vacancy flow

**Assumption**: Only a mono-interstitial and vacancy can diffuse individually.

### Net Vacancy Flow
The difference between vacancy and interstitial flux
\[ Z_v D_v C_v - Z_i D_i C_i \]

#### High dose rate
Recombination dominant
\[ C_{v,i} \propto (\text{dose rate})^{1/2} \]
\[ Z_v D_v C_v - Z_i D_i C_i \propto (\text{dose rate})^{1/2} \]

#### Low dose rate
Sink dominant
\[ C_{v,i} \propto (\text{dose rate})^1 \]
\[ Z_v D_v C_v - Z_i D_i C_i \propto (\text{dose rate})^1 \]

We have to verify and modify this simplified theory by the systematic experiments.

---

Experimental evaluation for the dose rate dependence of the net vacancy flow

**Assumption**: Only a mono-interstitial and vacancy can diffuse individually.

### Net Vacancy Flow
The difference between vacancy and interstitial flux
\[ Z_v D_v C_v - Z_i D_i C_i \]

Experimentally measured net vacancy flow by the void growth rate is proportional to (dose rate)^1/2.

Annihilation of point-defects by recombination not by diffusion of interstitials and vacancies.

**There are other diffusive species under irradiation.**

T. Okita, et.al., 2002
Basic researches to evaluate the dose rate effects

- Molecular Dynamics Simulations to evaluate the displacement damage process in the initial stage of irradiation with high energetic neutrons
- Physically based analysis at nano-micro scales
  - Collision cascades directly create small clusters of self-interstitial atoms (SIA clusters).
  - Some of SIA clusters are highly mobile along certain crystallographic direction (Rapid diffusions of SIA clusters).
  - The mobility of these clusters are much higher than mono-interstitials or vacancies.
  - These are also experimentally observed with in-situ irradiation experiments.

A. Kubota, T. Okita, et. al., 2003

Experimental evaluation for the dose rate dependence of the net vacancy flow

- Recombination by SIA-clusters $\propto (\text{dose rate})^{1/2}$
- Recombination by interstitial and vacancy $\propto (\text{dose rate})^{1/2}$
- Sink dominant $\propto (\text{dose rate})^1$


Hierarchical Structure of Degradation of Materials under Radiation Environments

Effects of Radiation in Nuclear Materials and Ageing Management of Nuclear Systems

Radiation Damage Processes

- Nuclear Reaction Phase
- Collisional Phase
- Cooling Phase
- Thermal Phase
- Diffusional Phase
- Microstructure
- Property Changes
- Lifetime Optimization

Evaluation Methodologies

- Monte Carlo Simulation
- Molecular Dynamics
- Analytical Diffusion & Reaction Rate Equations
- Finite Element Method
- Dislocation Dynamics
- Post Irradiation Tests
- System Maintenance, Safety, Reliability, Economy

From Engineering Issues to Micro-World

Enhanced swelling at lower dose rate

Fe-15Cr-16Ni, SA

Temperature: ~ 427 °C

- Lower dose rate enhances swelling by shortening the incubation dose.
- The steady state swelling rate is not affected by the difference in dose rate.

UCBNE-5113/GoNERI-0003
The steady state swelling rate is also observed at dose rates as low as $< 10^{-7}$ dpa/sec and irradiated less than 1 dpa.

The incubation dose of swelling can therefore vary from $< 1$ dpa to $> 45$ dpa when the dose rate varies over more than two orders of magnitude.
Strong effect of dose rate on incubation dose

Fe-15Cr-16Ni, SA

Temperature: ~ 427 °C

• The incubation dose of swelling is almost linearly proportional to dose rate.

The strategy to apply the physical model to the commercial alloys

- Effects of minor elements on void structure -

Temperature: ~ 427 °C

Fe-15Cr-16Ni

1.7 x 10⁻⁶ dpa/sec

43.8 dpa

67.8 dpa

100 nm

Fe-15Cr-16Ni-0.25Ti

Fe-15Cr-16Ni-0.25Ti-0.05C
Effects of C addition on swelling

Swelling is strongly suppressed. The dose rate effect disappears.

C atoms appear to disturb the rapid diffusions of SIA clusters, resulting in erasing the recombination at low dose rate.

Dose rate effects of microstructural evolution and irradiation hardening

Interstitial cluster formation and their effects on hardening based on the model

(1) Doses before loop saturation
Loop density at a fixed dose is proportional to (dose rate)^{-1/2}.

\[ \Delta H_v \propto \sqrt{N_l \cdot d} \propto (\text{dose rate})^{-1/4} \]

Non-conservative

(2) Transient dose region
Loop density shows complex dependence on dose rate.

(3) Doses after loop saturation
Loop density at a fixed dose is proportional to (dose rate)^{1/2}.

\[ \Delta H_v \propto \sqrt{N_l \cdot d} \propto (\text{dose rate})^{1/4} \]

Conservative

* When dose rate effects on loop size is negligible.
Additional recombination mechanism modifies dose rate effects of microstructural evolution

**Additional recombination retards loop saturation.**

Loop density becomes proportional to \( (\text{dose rate})^{-1/2} \) even at higher dose, which has not been expected by the conventional model.

\[
\Delta H_v \propto \sqrt{N_L \cdot d} \propto (\text{dose rate})^{-1/4}
\]

Non-conservative effects, which would have been recognized to be conservative without our model.

---

**Dose rate dependence of irradiation hardening**

- Irradiation hardening has a good relationship with 0.25 power of the dose rate, which is well agreed with our model.

- The physical model enables to evaluate the experimental evidence of the dose rate effects, which would have been recognized as experimental errors without the model.

- With the physical model, the data points obtained by a higher dose rate irradiation are very valuable to predict the irradiation hardening.

*Torimaru et al., 2005*
Summary

• We have constructed the R&D roadmap for the ageing management of the nuclear power plants, and the new project has been started by the industry-government-academia partnership to improve the ageing management infrastructure.

• The essential mechanisms for the dose rate effect are described.
  – Rapid diffusions of SIA clusters by collision cascades can accelerate recombination of point-defects, and strongly modifies the microstructural evolution at low dose rate.
  – The physical model enables to predict the material behavior with low dose rate irradiation by utilizing the database with the higher dose rates.
  – The model gives the insight whether the evaluation based on the higher dose rate irradiation is conservative or non-conservative.

• Based on these academic knowledge, the project is ongoing, which includes the dose rate effects on irradiation embrittlement and IASCC susceptibility.
Seismic Safety, Risk Reduction and Performance-Based Design
Aimed at Nuclear Facility Structures

Bozidar Stojadinovic, Associate Professor
Department of Civil and Environmental Engineering
University of California, Berkeley

Outline

◆ What is performance-based design?
◆ How to design structures to reduce risk?
◆ What are the safety-increasing innovations in structural engineering?
◆ Why should we do this for the new nuclear cycle in the US?
Performance-Based Design

◆ Design to achieve specified results rather than to adhere to particular technologies or prescribed means (Moehle, EERI Distinguished Lecture, 2005)

◆ Directly address the needs of the owner or user of the system or structure in their risk environment

Prescription vs. Performance

◆ A code provision (ASCE 43-05: 6.2.2(a)):
  “Minimum joint reinforcement shall consist of X-pairs of #4 diagonal cross-ties spaced 12 in. on center.”
Prescription vs. Performance

What is the performance?
- Is such joint safe?
- If so, what is the level of safety?
- If so, how much does it cost to be so safe?
- Would #3 cross-ties spaced 6 in. on center be better or worse? Safer? Less expensive? Easier to build?

Performance-Based Design: Earthquake Engineering View

Prof. Mahin, CEE 227 Lectures
Performance-Based Design: Deterministic Quantification

Performance-Based Design: Probabilistic Quantification

Prof. Mahin, CEE 227 Lectures
How to Design for Performance?

Prof. Mahin, CEE 227 Lectures

Probabilistic Framework

\[ v(DV) = \int \int \int G(DV|DM) \cdot dG(DM|EDP) \cdot dG(EDP|IM) \cdot d\lambda(IM) \]

- **Impact**
- **Performance (Loss) Models and Simulation**
- **Hazard**

- IM – Intensity Measure
- EDP – Engineering Demand Parameter
- DM – Damage Measure
- DV – Decision Variable

\[ v(DV) \] – Probabilistic Description of Decision Variable
(e.g., Mean Annual Probability $\$ Loss > 50\% Replacement Cost)
Performance-based Evaluation Example:
How Safe are our Bridges?

Type 1
Type 11

Framework for Bridge Evaluation

Hazard Model
Demand Model
Damage Model
Decision Model

Select and scale ground motions
Framework for Bridge Evaluation

Do non-linear time-history analyses

Decision Variable (DV)

Damage Measure (DM)
discrete, continuous

Decision Model

Damage Model

Demand Model

Hazard Model

Framework for Bridge Evaluation

Performance (damage) states

Decision Variable (DV)

Damage Measure (DM)
discrete, continuous

Decision Model

Damage Model

Demand Model

Hazard Model
Framework for Bridge Evaluation

**Hazard Model**

Deaths? Dollars? Down-time?

**Demand Model**

**Damage Model**

Discrete
Continuous

**Decision Model**

Decision Variable (DV)

**Outcome: Repair**

cost ratio
fragility
curves

$S_d(T_s)=1g$
Common Probabilistic Basis for Civil and Nuclear Structures

Given a seismic hazard environment and a structure, the probability that a performance objective is achieved is:

\[ P_{PO} = \int P(PO \mid \text{hazard}) \, d(\text{hazard}) \]

Consider probability distributions of seismic hazard, of demand and of capacity due to:

- Lack of knowledge (epistemic uncertainty)
- Record-to-record ground motion randomness (aleatory uncertainty)

Seismic Hazard and Probability of Failure

- Hazard: probability of exceeding a value of ground motion intensity (hazard curve)

\[ P_H = H(s_a^{P_H}) = k_0 (s_a^{P_H})^{-k} \]

- Failure: a comparison demand and capacity

\[ P_F = P(C \leq D) = \int P(F \mid s_a) \, dH(s_a) \]
DOE-1020 and ASCE 43-05: (Nuclear) Acceptance Criteria

- Probability of failure is smaller than probability of hazard
- Risk reduction ratio at the structure level

\[ R_R = \frac{P_H}{P_F} \]

<table>
<thead>
<tr>
<th>Performance Category</th>
<th>Risk Reduction Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>PC-1 (conventional)</td>
<td>( R_R = 1.0 )</td>
</tr>
<tr>
<td>PC-2 (internal exposure risk)</td>
<td>( R_R = 1.0 )</td>
</tr>
<tr>
<td>PC-3 (labs, fuel cycle facilities)</td>
<td>( R_R = 10.0 )</td>
</tr>
<tr>
<td>PC-4 (experimental reactors)</td>
<td>( R_R = 20.0 )</td>
</tr>
</tbody>
</table>

Conventional Design: Acceptance Criteria

- Probability of failure is, implicitly, assumed equal to the probability of hazard
- Design equation:
  \[ \phi C > \sum \gamma D \]

\( R_F = P_H \)
Common Risk-Informed Design Framework

New nuclear power plants can be designed using a risk-informed performance-based framework.

Models for most elements of the structure exist, including aleatory and epistemic uncertainties.

Modeling can be extended to:
- Other extreme hazards (natural and man-made)
- Ageing effects (construction and maintenance)
- Accidents (effects on the environment and society)

Risk-based evaluation is used for some aspects of the nuclear fuel cycle design today.
Innovations in Civil Engineering (DOE NP2010 Initiative)

- Over the past 30 years civil engineering did not stand still:
  - Technologies ready for deployment
  - New and promising technologies worthy of additional exploration and development

- Note: this is just the CE side!
  - No NE-CE-ME synergies were explored

Ready-to-Use CE Technologies

- Response modification devices
- Steel-plate sandwich structures
- Advanced concrete admixtures
- Composite plastics for reinforcement
- Pipe bends vs. welded elbows
- Precision blasting for rock removal
- High-deposition rate and robotic welding
- Cable splicing
- 4-D modeling and BIM
- GPS use in construction
- Open-top installation
Upcoming CE Technologies

- Prefabrication, preassembly and modularization
- Advanced information management and control during design and construction

Earthquake Engineering of Heavy Structures

- Large weight, often positioned high above the foundation
- Combat inertia forces through:
  - Strength
  - Flexibility
  - Damping
Steel-plate Sandwich Walls

Steel plate used as:
- Form
- Reinforcement

Composite action with concrete enabled using studs
Steel-plate Sandwich Walls

Steel plate used as:
- Form
- Reinforcement

Composite action with concrete enabled using studs

Limited damage
Steel-plate Sandwich Walls

- Steel plate used as:
  - Form
  - Reinforcement
- Composite action with concrete enabled using studs
- Very strong
- Very ductile, too!

Concrete Wall at 65y

Steel-plate Sandwich Walls

- Steel plate used as:
  - Form
  - Reinforcement
- Modular, prefabricated components
- Rapid construction
Response Modification Devices

- Devices designed to alter dynamic response of structures:
  - Base isolation, to reduce input motion/energy
  - Added damping, to dissipate energy that enters the structure

Base Isolation Concept

- Provide a soft, deformable layer between the structure and the ground
- Not new!
  - Sanjusangendo Temple in Kyoto, built in 1164
Base Isolation Concept

Base Isolation Benefits

- Reduced motion of the structure
- Reduced acceleration of the content
Base Isolation Benefits

- Reduced motion of the structure
- Reduced acceleration of the content

Problems:
- Vertical acceleration
- Seismic gap
- Crossing the gap
Base Isolation Devices: Laminated Rubber Bearings

- Technology developed in 1980’s
- Used in non-nuclear but safety-critical structures:
  - LNG tanks
  - Hospitals
  - Emergency command centers

Base Isolation Devices: Friction-Pendulum Bearings

- Technology developed in 1990’s
- Used in conventional building structures
- Used in critical infrastructure:
  - Bay Area long-span bridge crossings
  - Off-shore platforms
Response Modification Devices: Seismic Dampers

- Steel damper
- Lead damper
- Oil damper
- Friction damper

Why Design Based on Performance?

Integrate the entire nuclear fuel cycle design to enable transparent risk-informed decisions on:

- Safety
- Security
- Economy
- Effects on the environment (sustainability)
Safety, Security, Economy and Sustainability

- Use simulation to evaluate effects of hazards:
  - Anticipate before we build them
- Balance safety and economy:
  - Do what is necessary, no more, no less
  - Find the sweet spots where small investments result in significant benefits
- Integrate security and sustainability:
  - Design right from the get-go
  - Reduce carbon emissions during construction, too!
  - Be modular, reuse and recycle

How Do We Get There?

- A unique opportunity is here:
  - A new building cycle is starting
  - There is little institutional memory left:
    - Bad: there is no experience
    - Good: there is no experience!
- Form cross-disciplinary engineering teams as early as possible:
  - State performance objectives, not prescriptions
  - Work together to formulate the design process and execute it right!
Role of Civil/Structural Engineering

◆ Performance-based design:
  ■ Utilize advances in conventional design to energize new nuclear construction
  ■ Bridge the engineering skill gap in structural and earthquake engineering

◆ New and emerging technologies:
  ■ Response modification devices
  ■ New composite structural systems
  ■ Modular construction and maintenance
  ■ Modern construction and life cycle management

Thank you!

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After Dinner Lecture

Moderated by
Dr. C. Forsberg
ORNL

January 7, 2008
Holistic Design: Safety, Reliability, Security and Sustainability for the PB-AHTR

Per F. Peterson
Department of Nuclear Engineering
University of California, Berkeley

January 7, 2007

Overview

• The goals for Generation IV aim toward holistic design of nuclear energy systems
  – Safety and reliability
  – Security
  – Sustainability
  – Economics

• UC Berkeley’s Pebble Bed Advanced High Temperature Reactor (PB-AHTR) project has provided a test bed
  – Supported under a 3-year NERI grant
  – Focused on an innovative liquid-cooled high temperature reactor
Some high level observations

- The capital cost of reactors dominates the economics of nuclear energy
  - Economics are affected strongly by (1) power density, (2) efficiency, (3) availability (reliability), and (4) value of energy products
- Reliability (important for economics) is closely related to safety, physical protection, and effective international safeguards
  - In many areas goals are complementary
    » Reduced maintenance/accessibility needs for passive safety systems
    » Human performance measures to prevent inadvertent and deliberate mistakes
    » Importance of knowing where materials are
  - Some may conflict
    » Access control versus emergency response

A few more observations

- To analyze reliability, safety, physical protection and proliferation resistance we should use a common approach:
  - Identify potential challenges/threats
  - Assess the system response
  - Evaluate outcomes against acceptance criteria
- The first element of design is to identify and categorize (by probability) challenges and threats
  - Safety - internal, external events
  - Reliability - materials/fuels degradation
  - Physical protection - theft, radiological sabotage
  - Proliferation - concealed acquisition, abrogation, clandestine facilities
- The second element is to analyze system response
  - Code scaling, applicability and uncertainty analysis
  - Use of Phenomena Identification and Ranking
    » identify complex phenomena and eliminate from design
    » emphasize scalability from low to high power levels
UCB PB-AHTR R&D has identified several goals important for economics and safety

- High efficiency, high power density energy conversion
  - Deliver heat at same average temperature as a modular helium reactor
  - Compact closed gas Brayton cycle
- Low-pressure containment/confinelement structure
  - Low volatility, chemically inert coolant
  - No in-containment stored energy sources
- Long thermal time constant for reactor core heat up
  - Large thermal inertia from fuel and coolant
  - Large temperature margins to fuel damage
- Flexibility to evolve rapidly
  - Unique simulant fluids for IET and SET experiments for licensing
  - Use well understood materials and fuel
- Flexibility to up-rate to high power outputs
  - High modularity in core and decay heat removal systems
- Flexibility for advanced fuel cycles
  - Improved uranium utilization, negative void reactivity for deep burn TRU fuel

UCB AHTR design work has been guided by these goals since 2002

The Advanced High-Temperature Reactor (AHTR)
combines two older technologies

Coated particle fuel

Liquid fluoride salt coolants
Boiling point ~1400°C
Reacts very slowly in air
Excellent heat transfer
Transparent, clean fluoride salt

UC Berkele
Liquid fluoride salts have fundamentally different properties than other reactor coolants

Thermophysical Properties of S-PRISM, GT-MHR, and AHTTR Reactor Coolants and Materials

<table>
<thead>
<tr>
<th>Material</th>
<th>( T_{\text{melt}} ) (°C)</th>
<th>( T_{\text{boil}} ) (°C)</th>
<th>( \rho ) (kg/m(^3))</th>
<th>( C_p ) (kJ/kg°C)</th>
<th>( \rho C_p ) (kJ/m(^3)°C)</th>
<th>( k ) (W/m°C)</th>
<th>( \nu \cdot 10^6 ) (m(^2)/s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^7)Li(_2)Be(_4) (Filibe)</td>
<td>459</td>
<td>1430</td>
<td>1940</td>
<td>2.34</td>
<td>4540</td>
<td>1.0</td>
<td>2.9</td>
</tr>
<tr>
<td>0.5NaF-0.42ZrF(_4)</td>
<td>500</td>
<td>1290</td>
<td>3140</td>
<td>1.17</td>
<td>360</td>
<td>0.01</td>
<td>0.53</td>
</tr>
<tr>
<td>Sodium</td>
<td>97.8</td>
<td>883</td>
<td>790</td>
<td>1.27</td>
<td>1000</td>
<td>0.01</td>
<td>0.25</td>
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<tr>
<td>Lead</td>
<td>328</td>
<td>1750</td>
<td>10540</td>
<td>0.16</td>
<td>1700</td>
<td>0.01</td>
<td>0.13</td>
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<tr>
<td>Helium (7.5 MPa)</td>
<td>0</td>
<td>100</td>
<td>732</td>
<td>3.8</td>
<td>20</td>
<td>0.01</td>
<td>0.29</td>
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<tr>
<td>Hastelloy C-276</td>
<td>~1350</td>
<td>8890</td>
<td>0.43</td>
<td>3820</td>
<td>9.8</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Graphite</td>
<td>1700</td>
<td>1.90</td>
<td>2320</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*Approximate physical properties 700°C except the pressurized water data shown at 290°C for comparison; \( \rho \) = density, \( C_p \) = specific heat, \( k \) = thermal conductivity, \( \nu \) = viscosity.

- High volumetric heat capacity provides high thermal inertia
  - High power density, low pressure operation possible compared to helium cooled reactors
  - High efficiency, compact primary loop equipment compared to water cooled reactors
  - Transparent coolant, low thermal shock, low chemical reactivity compared to sodium cooled reactors

The Modular PB-AHTR is a compact pool-type reactor with passive decay heat removal

UC Berkeley
UC Berkeley’s work with liquid fluoride salts began in the early 1990’s for fusion energy applications

- Flibe (Li$_2$BeF$_4$) is an attractive fusion coolant
- Extensive experience gained in modeling and experiments

Scaled jet experiments simulating liquid salts

UCB adopted pebble fuel as its baseline in April, 2006 and demonstrated viability using the Pebble Recirculation Experiment (PREX) in October, 2006
Now UCB is working on a Modular PB-AHTR with a nominal power output of ~900 MWth

- Core power density between 20 to 30 MW/m³ (versus 4.8 MW/m³ for the PBMR)
- Increase length of defueling chutes
- Eliminate PHX, move DHX into reflector volume above core
- Use replaceable “pebble channel assemblies” to guide pebble flow, provide added moderation and insertion locations for control assemblies

AHTR Economics

Top-down analysis
The new Modular PB-AHTR is designed to maintain superior economics with a modular HTR design

- **Comparison of PB-AHTR with the PBMR:**
  - 2 x power output per reactor
  - ~30 MWth/m³ core power density versus 4.8 MWth/m³
  - large reduction in vessel size
  - atmospheric pressure operation
  - 4 x reduction in spent fuel volume per unit of electricity/process steam produced
  - maximum fuel temperature during transients/accidents reduced from 1600°C to 1000°C

The smaller size and low mass of major components (reactor vessel weight < 150 tons) has implications for the construction schedule.

PB-AHTR fuel costs will be lower than for LWRs

- **Compare the fuel costs of a PB-AHTR (10% enrichment, 129 MWd/kg, 46% power conversion efficiency) versus an LWR (4.5% enrichment, 50 MWd/kg, 33% efficiency):**
  - Natural uranium consumption: 64.2% (extends uranium resource)
  - Enrichment: 86.2%
  - Fuel fabrication: 150%
  - Total fuel costs PB-AHTR vs. LWR’s: 80.7%

Assumptions: Tails assay 0.3%; LWR fuel cost breakdown 60% uranium mining/conversion, 28% enrichment, 12% fabrication.
The PB-AHTR can also provide transmutation services

- Neutronics simulations (2400 MWth design) have confirmed negative void reactivity for deep-burn TRU transmutation fuel
- Destruction of heat-generating actinides Am-241/Pu-241 50% greater than for VHTR, discharge burn up and spent fuel volume also improved

<table>
<thead>
<tr>
<th>System</th>
<th>PB-AHTR</th>
<th>VHTR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total core power [MWth]</td>
<td>2,400</td>
<td>600</td>
</tr>
<tr>
<td>Power density [MW/m³]</td>
<td>10.20</td>
<td>4.68</td>
</tr>
<tr>
<td>Initial HM mass [t]</td>
<td>2.33</td>
<td>1.02</td>
</tr>
<tr>
<td>C/HM (spent fuel volume)</td>
<td>1,993</td>
<td>2,955</td>
</tr>
<tr>
<td>Leakage probability [%]</td>
<td>3</td>
<td>5</td>
</tr>
<tr>
<td>Burnup [GWd/THM]</td>
<td>685.2</td>
<td>621.3</td>
</tr>
<tr>
<td>Residence time [EFPD]</td>
<td>663</td>
<td>1,052</td>
</tr>
<tr>
<td>HM inventory reduction [%]</td>
<td>69.46</td>
<td>63.24</td>
</tr>
<tr>
<td>Pu inventory reduction [%]</td>
<td>77.06</td>
<td>69.49</td>
</tr>
<tr>
<td>Fissile Pu reduction [%]</td>
<td>95.60</td>
<td>91.16</td>
</tr>
<tr>
<td>²³⁷Np and precursors inventory reduction [%]</td>
<td>88.80</td>
<td>58.44</td>
</tr>
</tbody>
</table>

Construction time

- ESBWR reactor building is
  - 70 m tall from bottom of base mat to top of reactor building
  - 40 m from bottom of reactor cavity to refueling deck floor
  - 20 m from grade to bottom of base mat
  - Reactor vessel ~ 1050 metric tons
- Modular PB-AHTR reactor building is
  - 35 to 40 m tall from bottom of base mat to top of reactor building
  - 15 m from bottom of reactor cavity to refueling deck floor
  - Need to determine excavation depth (20 m depth would fully bury the reactor to the refueling deck floor)
  - Reactor vessel ~ 150 metric tons is much lighter/more transportable than a ~ 1000 metric ton LWR or MHR vessel
Modular PB-AHTR Design Description

The Modular PB-AHTR will have a nominal power output of ~900 MWth
Modularity enables simple scaling from Pilot to Modular to Central-Station power levels

The PB-AHTR uses well understood materials and fuel

- TRISO based fuel is well understood
  - Peak temperature during normal operation and accidents < 1000°C
  - Capability to manufacture being reestablished
  - Uses special pebble design (see later slide), requires confirmatory testing
- Metallic components are Alloy 800H clad with Hastelloy N for corrosion resistance
  - The baseline design has a conservatively low 704°C core outlet temperature to assure high corrosion resistance (extensive test data available)
  - Alloy 800H provides structural strength and is ASME Section III code qualified for use up to 760°C; ORNL now extending code case to 900°C
  - Hastelloy N has well understood corrosion resistance with fluoride salts
- Reflectors are graphite
  - Capability to manufacture nuclear-grade graphite has been reestablished

The baseline PB-AHTR fuel and materials have well understood reliability
PB-AHTR power conversion provides high efficiency

- The baseline PB-AHTR design has a core inlet/outlet temperature of 600°C/704°C, delivering heat at an average temperature of 652°C.
- The GT-MHR has a core inlet temperature of 450°C and outlet temperature of 850°C, delivering heat at an average temperature of 650°C (same as the baseline PB-AHTR).
- The PB-AHTR and the GT-MHR can drive closed gas Brayton cycles with the same power conversion efficiency, ~46%.
  - The PB-AHTR can use a multiple reheat gas Brayton cycle.

Evolution of advanced Brayton cycles – multiple reheat allows higher power density

To Scale

1125 MW(e) MCGC-LT

1125 MW(e) MCGC-LT

165 MW(e) PBMR

286 MW(e) GT-MHR

high temperature Brayton cycle without reheat

high temperature Brayton cycle with reheat
Pebbles and Liquid Salts

- Pebbles float in liquid salts
  - Flibe, $^7\text{Li}_2\text{BeF}_4$ (1.938 g/cm$^3$ at 700°C) has higher density than pebbles (~1.8 g/cm$^3$), so pebbles have modest buoyancy and float.
  - Pebble density can be reduced further, if desired (particularly for the annular pebble configuration).
  - Pebble buoyancy provides low upward terminal velocities (~0.5 m/s) reducing pebble bed impact forces.

The UCB Pebble Recirculation Experiment (PREX) has studied pebble dynamics for the AHTR

- PREX reproduces the major phenomena required for pebble recirculation
  - injection
  - pebble terminal rise velocity
  - pebble bed dynamics
- PREX uses 2.54-cm diameter polypropylene spheres with water
  - 1/2 length scale, matches:
    - Reynolds number
    - Froude number
    - pebble/salt density ratio
Extensive experiments in PREX-1 have demonstrated viability of pebble recirculation

PREX-1 with 8300 pebbles

a) Scaled defueling chute
b) Pebbles outside of liquid bed

- Pebbles float and can be recovered below the liquid surface and re-injected.
- Measured packing density: 60% ±1% (independent of injection method)
- Measured pressure loss coefficients match existing correlations within 2 to 20%. (Additional experiments planned)

Next steps will include construction of PREX-2 to verify pebble recirculation in a Pebble Channel Assembly

Baseline design for lower half of PCA showing configuration of pebble channels

UC Berkeley

UCBNE-5113/GoNERI-0003
RELAP5-3D Modeling of 900 MWth PB-AHTR transient response to LOFC and ATWS transients

RELAP5-3D model for 900 MWth Modular PB-AHTR

Heat Sink
Upper reflector
Active core
Source at 600 C
Reflector gap

DHX
Fluidic diode
(200/1 loss coefficients)

UCBNE-5113/GoNERI-0003
Steady-state solution for 900-MWth core centerline temperatures

Results for 3-cm diameter annular pebbles

An annular pebble configuration is used to provide fuel temperature

- The annular configuration moves fuel to the outer 50% of the pebble volume, reduces maximum fuel temperature by ~140°C
- The reduction in fuel temperature is substantial, and improves ATWS response significantly
Transient response of 900 MWth PB-AHTR to LOFC

- Response is gentle even with 30 MW/m³ power density

Results for Anticipated Transient Without Scram for 900 MWth Design

- Under ATWS, coolant outlet temperature increases until reactor shuts down on negative temperature feedback
- ATWS results are sensitive to the fuel and coolant temperature reactivity feedback
PB-AHTR Experimental Program

The Modular PB-AHTR Experimental Program (Viability Phase)

- **Integral Effects Tests**
  - Compact Integral Effects Test (CIET) facility
    » Scaled simulant fluid IET to study system response to LOFC, ATWS, and other transients
  - Pebble Recirculation Experiment
    » Scaled simulant fluid IET to study pebble recirculation hydrodynamics

- **Separate Effects Tests**
  - Scaled High Temperature Heat Transfer (S-HT$^2$) facility
    » Heat transfer coefficient measurements using simulant fluids
  - Other SET experiments
    » Pebble confirmatory irradiation experiments, etc.
The Modular PB-AHTR Experimental Program
(Performance and Demonstration Phases)

• Component Tests
  – Various scaled component tests with simulant fluids (water)
  – Component Test Facility (CTF)
    » Major non-nuclear facility to test primary, intermediate and
      DRACS loop components under prototypical liquid salt
      conditions

• Pilot Plant Tests
  – nuclear fuel loading and pre-critical (zero power) testing
  – low-power (<5%) testing and operation
  – power ascension testing and operation not in excess of 100%
  – interim operation
  – maintenance and in-service inspection procedures

Dowtherm heat transfer oil will be used as the principal
simulant fluid for PB-AHTR IET/SET experiments

Scaling parameters to match Pr, Re, Gr, and Fr for flibe and Dowtherm A

<table>
<thead>
<tr>
<th>Flibe Temperature [°C]</th>
<th>600</th>
<th>650</th>
<th>700</th>
<th>750</th>
<th>800</th>
<th>850</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dowtherm A Temperature [°C]</td>
<td>63</td>
<td>82</td>
<td>104</td>
<td>129</td>
<td>157</td>
<td>191</td>
</tr>
<tr>
<td>Length scale</td>
<td>( \frac{l_m}{l} )</td>
<td>0.52</td>
<td>0.51</td>
<td>0.49</td>
<td>0.46</td>
<td>0.44</td>
</tr>
<tr>
<td>Velocity scale</td>
<td>( \frac{u_m}{u} )</td>
<td>0.72</td>
<td>0.72</td>
<td>0.70</td>
<td>0.68</td>
<td>0.66</td>
</tr>
<tr>
<td>( \Delta T ) scale</td>
<td>( \Delta T_m / \Delta T )</td>
<td>0.30</td>
<td>0.30</td>
<td>0.30</td>
<td>0.29</td>
<td>0.29</td>
</tr>
<tr>
<td>Heat conductivity</td>
<td>( \frac{\lambda_m}{\lambda} )</td>
<td>0.14</td>
<td>0.13</td>
<td>0.13</td>
<td>0.12</td>
<td>0.12</td>
</tr>
<tr>
<td>Ther. diffusivity</td>
<td>( \frac{\alpha_m}{\alpha} )</td>
<td>0.37</td>
<td>0.35</td>
<td>0.33</td>
<td>0.31</td>
<td>0.28</td>
</tr>
<tr>
<td>( \beta \Delta T )</td>
<td>( \frac{\beta \Delta T_m}{\beta \Delta T} )</td>
<td>1.00</td>
<td>1.00</td>
<td>1.00</td>
<td>1.00</td>
<td>1.00</td>
</tr>
<tr>
<td>( \gamma \Delta T )</td>
<td>( \frac{\gamma \Delta T_m}{\gamma \Delta T} )</td>
<td>0.81</td>
<td>0.94</td>
<td>1.06</td>
<td>1.13</td>
<td>1.13</td>
</tr>
<tr>
<td>( \kappa \Delta T )</td>
<td>( \frac{\kappa \Delta T_m}{\kappa \Delta T} )</td>
<td>-0.84</td>
<td>-0.86</td>
<td>-0.89</td>
<td>-0.92</td>
<td>-0.95</td>
</tr>
<tr>
<td>Pumping power</td>
<td>( P_{p,m}/P_p )</td>
<td>5.2%</td>
<td>5.0%</td>
<td>4.2%</td>
<td>3.4%</td>
<td>2.8%</td>
</tr>
<tr>
<td>Heating power</td>
<td>( P_{q,m}/P_q )</td>
<td>2.1%</td>
<td>2.1%</td>
<td>1.9%</td>
<td>1.7%</td>
<td>1.5%</td>
</tr>
</tbody>
</table>

• Note that Pr, Re, Gr and Fr can be matched at < 2% of
  prototypical heater power
• Water will be used for hydrodynamics experiments
The PB-AHTR Compact IET performs the same role as the INL Semiscale facility

- **Semiscale simulation of PWR LOCA**
  - 1:1 height
  - 1:1705 flow area
  - 1:1705 power (2 MW)
  - 1:1 time
  - prototype temperature / pressure

- **CIET simulation of the PB-AHTR LOFC/ATWS**
  - 1:1 effective height (1:2 actual)
  - 1:190 effective flow area (1:756 actual)
  - 1:190 effective power (1:9000 actual, 100 kW)
  - 1:(2)\(^{1/2}\) time
  - reduced temperature / pressure
  - reduced heat loss
  - small distortion from thermal radiation

See http://users.owt.com/smsrpm/nksafe/testfac.html for a list of other LWR IET's

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The smaller physical size of liquid salt equipment reduces the size and cost of a CTF

- The PBMR HTF is a 40-m prototypical height experimental facility, that tests very large and bulky equipment
- Liquid salt uses small, thin-walled, low pressure components
- Creates implications for schedule and cost
Conclusions

- Holistic design considers safety, reliability, security, sustainability and economics simultaneously from the beginning of conceptual design
- UC Berkeley has been integrating the holistic design approach in its DOE-NERI project to study the AHTR
- Work at UC Berkeley and elsewhere has identified attractive features of liquid-salt cooled high temperature reactors
  - Potential for high power density (20-30 MWt/m³)
  - Low pressure operation, chemically inert coolant
  - Capability to operate on both LEU and TRU fuels
  - Reduced spent fuel volume
  - Likely attractive economics compare to MHRs/LWRs
Session 1: Student Paper presentations

Risk Management and Radiolysis

Chaired by
Prof. T. Okita
Department of Quantum Engineering and Systems Science, UT

8:00 AM – 9:00 PM
January 8, 2008
ALLOCATION OF RESOURCES AGAINST STRATEGIC AND NON-STRATEGIC THREATS TO NUCLEAR ENERGY SYSTEMS

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University of California, Berkeley
Berkeley, CA 94720-1730
lancekim@berkeley.edu

ABSTRACT

Allocating limited security and safety resources at a facility with diverse and redundant systems is difficult under severe uncertainty of adversary intentions. Insufficient knowledge of the frequency of an attack and the adversary’s choice of targets limits the value of a risk minimization framework that demands such information. Game theory methods that better model the strategic interaction between the defender and adversary offer an alternative framework for the assessment and management of security. Conditioned on an attack occurring, a two-person, zero-sum game theory model is applied to find a Nash equilibrium defensive strategy when an adversary can select from multiple sets of targets with dependencies between sets. Plant safety is described with a quantitative risk analysis model for well-characterized non-strategic initiating events. Lacking information on the frequency of an attack, a Pareto efficient frontier describes the efficient combinations of safety and security given available resources and technological possibilities. A multi-attribute utility model of individual choice identifies the preferred distribution of resources from amongst the efficient solutions. The results found by mathematical programming demonstrate that a quantitative risk analysis approach to security can underestimate risk and can result in different resource allocation in comparison to the game theory solution. The efficient frontier identifies combinations of safety and security from which a decision-maker can choose and simultaneously identifies inefficient and infeasible strategies that a decision-maker should avoid.

The preferred allocation of resources depends on subjective preferences for safety, security, and opportunity cost and substitutes for information on the frequency of attack.

Key Words: Risk Assessment, Physical Protection, Game Theory, Multiobjective Optimization

1. INTRODUCTION

Revisions to the Design Basis Threat to which nuclear power plants must defend against raises the question of how to best manage allocate resources to manage threats to security while managing plant safety. Quantitative risk analysis (QRA) techniques have been successfully applied to manage safety risks at nuclear power plants. [1] Applying a similar risk minimization framework to security [2] is limited by severe uncertainties in adversary behavior and neglects the intelligent nature of the threat.

To better model the strategic interaction between defender and adversary, game theorists have identified a defensive strategy that minimizes the adversary’s maximum expected gain as a Nash equilibrium to a simultaneous, two-person, zero-sum game. [3] If such a model is appropriate, this minimax objective represents a shift away from risk management based on adversary intentions to a method based on adversary capabilities. [4]
In this work, a minimax objective is applied to assess and manage security for a system like a nuclear power plant where redundant and diverse subsystems give rise to dependencies between sets of targets that an adversary can choose to attack. Solutions to the safety and security models are found via mathematical programming. In the absence of information on the frequency of attack, the Pareto efficiency criterion identifies efficient allocations of resources between safety and security.

2. SAFETY AND SECURITY MODELS

A model is defined that captures many salient features of a nuclear power plant including a perimeter, buildings, components, and pathways. A fault tree represents the functional relationships between system components with one-out-of-k success logic for parallel subsystems. (Figure 1) The minimum cut-set (MCS) of the fault tree is derived via Boolean expansion. Sets of targets separated by OR gates in the MCS represent the choices available to an adversary.

![Diagram of System Model](image)

**Figure 1. System model with adversary pathway (left) and fault tree (right).**

Technological possibilities for each target and pathway are described by a decreasing convex function of resources, \( r \), expended. These technology functions reflect the conditional probability of failure given an initiating event (e.g. random events impacting plant safety or an attack by an adversary with defined capabilities).

\[
Pr(\text{failure} \mid \text{event}) = Pr e^{-h r}
\]  

2.1. Safety Model

A mathematical program is defined that allocates limited safety resources, \( R_{\text{safety}} \), amongst safety-related targets, \( T_{\text{safety}} \), to minimize safety risk, \( \rho \), against a set of random non-strategic initiating events that occur with known frequency. Safety-related targets are those targets that must be disabled to generate a source term and fail containment. The resource allocation vector, \( r_{\text{safety}}=[r_{P1}, r_{P1}, \ldots] \), whose sum cannot exceed \( R_{\text{safety}} \) and must satisfy the usual non-negativity constraint, describes the resources expended on safety-related targets. The rare event approximation is applied to calculate safety risk i.e. the expected value of a large release.
2.2. Security Models

Security is assessed with risk minimization and minimax objectives utilizing a pathway-based assessment of physical protection system performance conditioned on an attack occurring. [5, 6] Likely pathways are identified for a well-defined adversary intent on effecting a large release where exiting the system is unimportant. The performance of the system relies entirely on the intrinsic vulnerability of the targets and pathways. Four target-sets, \{TS1, TS2, TS3, TS4\}, are identified each of which are composed of security-related targets and pathways, \(T_{\text{security}}\), that must be attacked to effect a large release. These four target-sets represent the choices available to the adversary.

The risk minimization approach allocates security resources, \(R_{\text{security}}\), by assuming an adversary attack strategy, \(\alpha_i\), representing the probability that an adversary will choose to attack a target-set, \(TS_i\), conditioned on an attack occurring. The intrinsic vulnerability of each target-set, \(V_{TS_i}\), is a function of the resource allocation vector, \(r_{\text{security}}\), amongst the security-related targets and pathways. (Equation 3) A minimax security model allocates resources to minimize the maximum target-set vulnerability conditioned on an attack occurring. (Equation 4) As before, the resource allocation vector is constrained by security resources, \(R_{\text{security}}\), and a non-negativity requirement.

\[
\min_{r_{\text{security}}} \sum_{i \in TS_i} \alpha_{TS_i} V_{TS_i} \left( \bar{r}_{\text{security}} \right) \\
\text{s.t.} \\
\sum_{i \in T_{\text{security}}} r_i \leq R_{\text{security}} \\
r_i \geq 0, \ \forall i \in T_{\text{security}}
\]

\[
\min_{r_{\text{security}}} \max \left\{ V_{TS1} \left( \bar{r}_{\text{security}} \right), ..., V_{TS4} \left( \bar{r}_{\text{security}} \right) \right\} \\
\text{s.t.} \\
\sum_{i \in T_{\text{security}}} r_i \leq R_{\text{security}} \\
r_i \geq 0, \ \forall i \in T_{\text{security}}
\]
minimax approach where an intelligent adversary selects a target-set to impose the greatest
damage is a defensive strategy that introduces conservatism to risk assessment and management.

![Security Risk Reduction Profiles](image)

**Figure 2.** Security risk reduction profiles for risk minimization (QRA $[\alpha_{TS1}, \ldots, \alpha_{TS4}]$) and
game theory models.

### 3.2 Resource Allocation

Resource allocation depends on assumptions of adversary intentions in the risk minimization
model. If an adversary is assumed to attack a particular target set and no others, resources are
allocated to the target with the lowest marginal cost in the set. When there is uncertainty about
the adversary’s choice of targets, resources tend to be allocated to the perimeter as it is a
common element and is cost-effective to harden. Allocation to the perimeter is not necessarily a
robust strategy that is insensitive to assumptions of attack strategy. Such a conclusion
necessitates information on the relative likelihoods of the various attack strategies.

![Minimax Resource Allocation](image)

![Minimax Solution Results in Balanced Protection Between Target Sets](image)

**Figure 3.** Minimax allocation (left) and balanced protection (right)

In contrast, the minimax model initially allocates resources to the most vulnerable target set with
relatively low redundancy and diversity. For this model, all resources are initially allocated to
P1T7, the pathway between the perimeter and T7. As available resources increase, more
resources are allocated to the perimeter. (Figure 3, left) In doing so, the minimax model achieves “balanced protection” [5] by allocating resources to the most vulnerable target-sets such that all target-sets are less or equally vulnerable to the most vulnerable target-set. (Figure 3, right) For both models, the emphasis on the perimeter cannot be generalized to all systems as the perimeter may be relatively less cost-effective to defend in other systems.

3.3. Safety-Security Tradeoff

In the absence of information on the frequency of attack required for risk minimization, the Pareto efficiency criterion identifies nondominated combinations of safety and security given available resources and technological possibilities i.e. the efficient frontier. (Figure 4, left) The preferred solution on the efficient frontier is identified by maximizing a multi-attribute utility model (Equation 5) of individual choice describing preferences for safety, security, and opportunity cost with assumed constants and weighting factors. (Figure 4, right)

\[
U(s, \psi, R_{total}) = w_{se} e^{-k_{se} s} + w_{\psi} e^{-k_{\psi} \psi} - w_{oc} e^{-k_{oc} R_{total}}
\]  

(5)

Figure 4. Efficient frontiers for varying resource levels \(R=R_{safety}+R_{security}\) (left) and the utility maximizing location on an efficient frontier with lines of isoutility at a fixed level of resources (right)

4. CONCLUSION

A method for managing safety and security risk is explored for a system that achieves high reliability through redundancy and diversity when adversary behavior is unknown. The minimax objective represents a shift from risk management based on assumptions of an adversary’s intentions to one based on an adversary’s capabilities. Such an approach is consistent with the Generation IV Proliferation Resistance and Physical Protection methodology that advocates risk assessment conditioned on a reference threat occurring. [6] A risk management approach based
on adversary intentions converges with a capabilities approach if the adversary is assumed to attack the most vulnerable set of targets. In the absence of information on the frequency of an attack, the efficient frontier identifies efficient combinations of safety and security. Though the Pareto efficiency criterion is agnostic on the preferred allocation between safety and security, it excludes inefficient and infeasible solutions that a decision-maker should avoid. The multiattribute utility function that identifies the preferred allocation substitutes for knowledge of the frequency of an attack.

A number of extensions can be envisioned to the method described. An integrated model may better assess the effects of synergistic and antagonistic interactions between safety and security. Passive safety systems could be better treated with capacity-demand models describing functional failures. A security performance metric based on a timely detection better captures detection, delay, and response elements of a physical protection system. Information gap decision theory may identify a maximally robust allocation against uncertainty in attack frequency. In all cases, heuristic optimization algorithms may better accommodate more complex models.

ACKNOWLEDGMENTS

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REFERENCES

APPLICATION OF FLUORESCENT PROBE TO YIELD DETERMINATION OF OH RADICAL PRODUCED IN WATER RADIOLYSIS WITH HEAVY IONS OF ENERGIES UP TO 28 GEV.

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ABSTRACT

Fluorescent probe produced in scavenging reaction of OH by Coumarin-3-carboxyl acid (CCA) can be measured sensitively. It was aimed to apply CCA solution to estimation of the dependence of the OH yield on LET and scavenging time. Heavy ions of energies up to 28GeV from HIMAC were taken for irradiation.

Key Words: Water Radiolysis, Heavy Ion, Hydroxyl Radical, LET, G value, Fluorescent probe, Sensitive Quantification.

1. INTRODUCTION

Recently, heavy ion therapy of cancer has begun because it is much more effective for special cancers that are highly resistant for other typical radiations such as fast electron, $^{60}$Co γ-ray, and x-ray [1]. In such therapy, ion is accelerated up to 5 GeV to attain sufficiently long penetration depth into human body (normally 30 cm is necessary) to treat cancer in deep position. Note that there are only a few heavy-ion accelerators possessing abovementioned ability in the world, including HIMAC (Heavy Ion Medical Accelerator in Chiba) at NIRS (National Institute of Radiological Science), Japan. While advantages of heavy-ion therapy are well-known phenomenologically, details of mechanism in which heavy-ion irradiation leads to distinctive biological effectiveness have not been clarified yet. Then, understanding of water radiolysis with heavy ions is necessary because water is main component of human body.

Based on researches with low LET radiations such as fast electron and gamma rays, a general picture of water radiolysis is understood [2]. Highly reactive radicals such as hydrated electron ($e^{-}_{aq}$), hydroxyl radical (OH), and hydrogen atom (H) are produced after irradiation. These events are called as physical and physicochemical stages. Among the species produced in these stages, OH is known to be most responsible to DNA damage induced in indirect action between radiation and cell [3,4].

Generally speaking, there are two methods to determine yields of water decomposition radicals. One is direct observation with pulsed beam and the other is indirect measurement named scavenger method. In the latter method, scavenger, which is highly reactive toward a certain
radical product, is used to convert radical species, which is difficult to observe directly, into stable and easily observable one. It is known that absorption of OH is in ultra-violet and extremely weak, and the scavenger method is normally taken for its yield determination. Recently, the yield of OH in heavy-ion irradiation was reported by Taguchi and Kojima [5], Baldacchino et al [6], and Yamashita et al. [7], in which absorption analysis was used for their quantification.

In the present work, it was aimed to apply fluorescent prove to yield determination of OH because fluorescence spectroscopy is normally much more sensitive than absorption spectroscopy. Then, Coumarin-3-carboxylic acid (CCA) is used as scavenger toward OH. It is known that after scavenging reaction of CCA toward OH, some of them stabilize as fluorescent species, 7-Hydroxy CCA (7OH-CCA) [8]. However, details of reaction scheme in which 7OH-CCA is produced are not known well, then, they were also examined preliminary using electron pulse radiolysis and $^{60}$Co $\gamma$-ray. In short, following three things were done in the present experiment. First, reactivity of CCA toward OH and $e_{aq}$ was investigated using electron pulse radiolysis technique. Second, reaction mechanism in which 7OH-CCA is produced, was investigated using analysis of final product after $^{60}$Co $\gamma$-ray irradiation under several different conditions. Third, the CCA solution was applied to yield determination of OH produced in water radiolysis with heavy ions. In addition, comparison with earlier reports using absorption spectroscopy will be shown. In addition, $G$ value, which are expressed as the number of produced or decomposed species per 100 eV of absorbed energy, refers to as the production or decomposition yield in water radiolysis throughout this paper

2. EXPERIMENTAL

2.1. Irradiation system

Electron pulse of 35 MeV with pulse width of 10 ns from S-band LINAC in Tokai, University of Tokyo was taken for irradiation. Dosimetry was preliminary done based on absorption of $e_{aq}$ ($\varepsilon = 18500 \text{ M}^{-1}\text{cm}^{-1}$ at 720 nm) and dose per pulse was estimated as 20 Gy/pulse. $^{60}$Co $\gamma$-ray source was used at Department of Nuclear Engineering and Management, the University of Tokyo. Dosimetry was conducted with the Fricke dosimeter, and dose rate was 10 Gy/min. Absorbed dose was varied from 10 to 40 Gy.

Heavy-ion beams were provided from Heavy Ion Medical Accelerator in Chiba (HIMAC). Their maximum energies were 150 - 500 MeV/u, respectively. Energy absorber made of PMMA (polymethylmethacrylate) plates was put upstream of samples to decrease ion energy and to increase LET value in samples. Then, another condition of higher LET value was also taken for irradiation for each heavy ion. Dosimetry was performed using secondary electron monitors, of which signals were preliminary associated with dose measured with ionization chambers. Dose rate and absorbed dose were 2-12 Gy/min and 10-40 Gy, respectively.

2.2. Sample Preparation

All samples were prepared with ultra-pure water from the Milli-Q system, and their pH values were fixed at 6.8 with phosphate buffer. All chemical were purchased from Aldrich. CCA concentration was varied from 0.1 to 26 mM. In heavy-ion irradiation, only aerated CCA
solutions with no additive were used. Sample conditions used in electron pulse radiolysis and gamma-ray irradiation are summarized in Table I. As seen in the table, saturating gas and/or liquid additive were selected in order to change ratio(s) of $e_{aq}^\cdot$ and/or OH reacting with CCA. In general, scavenging reaction can be considered as pseudo first order reaction as far as dose is sufficiently low, normally, less than several hundred Gy. Then, objective radical, which is OH in this work, follow first order exponential decay with lifetime of inverse of scavenging capacity (SC). SC is defined as the inverse of multiplication of scavenger concentration, [S], and rate constant of scavenging reaction, $k_s$. Then, scavenging time scale (ST) can be estimated as the inverse of the SC. In short, ST can be controlled by choosing scavenger and by controlling its concentration.

Table I. Sample conditions used in gamma-ray irradiation. [9 - 13]

<table>
<thead>
<tr>
<th>Gas</th>
<th>Additive</th>
<th>Solute(s)</th>
<th>Ratio of radicals reacting with CCA</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$t$-BuOH</td>
<td>50 mM $t$-BuOH</td>
<td>2-84 %</td>
</tr>
<tr>
<td>$N_2O$</td>
<td>nothing</td>
<td>25 mM $N_2O$</td>
<td>100-199 %</td>
</tr>
<tr>
<td>Air</td>
<td>nothing</td>
<td>0.25 mM $O_2$</td>
<td>100 %</td>
</tr>
<tr>
<td>Air</td>
<td>acetone</td>
<td>3-300 mM acetone</td>
<td>60 %</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.25 mM $O_2$</td>
<td></td>
</tr>
</tbody>
</table>

Note that only CCA solutions with $t$-BuOH saturated with Ar and those saturated with $N_2O$ were used in electron pulse radiolysis.

2.3. Analysis method with HPLC

Samples irradiated with gamma-ray or heavy ions were analyzed using fluorescence spectrometer coupled with HPLC (High Performance Liquid Chromatography). Acetonitrile and 10 mM phosphate aqueous solution were mixed with the ratio of 7 to 3 as reverse phase elution, and column temperature was set at 40 degree Celsius. Fluorescence emitted by 7OH-CCA was detected at 445 nm with excitation wavelength of 365 nm.

3. RESULTS AND DISCUSSION

3.1. Electron Pulse Radiolysis

Transient absorption spectra obtained in electron pulse radiolysis under condition of Ar saturation and $t$-BuOH addition is shown in Fig. 1 (a). Under the condition, reaction of CCA toward $e_{aq}^\cdot$ was observed. Because $e_{aq}^\cdot$ has a strong absorption at 720 nm, decay of absorbance at this wavelength was followed to estimate kinetics of $e_{aq}^\cdot$ as shown in inset of the figure. It is clear that decay of $e_{aq}^\cdot$ become faster and faster with increasing CCA concentration, suggesting that CCA surely reacts with $e_{aq}^\cdot$. This is supported by the fact that absorption at 350 nm increase with time in CCA this solution while it is not seen in pure water. Then, fitting with exponential decay was conducted and SC at each CCA concentration (0.1-0.5 mM) was estimated. After that, rate constants of CCA toward $e_{aq}^\cdot$, $k_s$, was determined from the relationship between SC, $k_s$, and CCA concentration, [S].

On the other hand, Fig. 1 (b) shows results obtained under condition of $N_2O$ saturation, in short, only reaction of CCA toward OH was observed. As opposed to the case of reaction of CCA...
toward $e^\cdot_{aq}$ products of scavenging reaction of CCA toward OH gives species having strong absorption of which peaks were observed at 360 and 440 nm. Then, kinetics at these wavelengths were analyzed and fitted by exponential formation to obtain scavenging capacity at each CCA concentration. Then, rate constant of reaction between CCA and OH was determined by the same method mentioned in the previous paragraph. Finally, rate constants of these reactions were $2.2 \times 10^{10} \text{ M}^{-1} \text{s}^{-1}$ toward $e^\cdot_{aq}$, $6.0 \times 10^{9} \text{ M}^{-1} \text{s}^{-1}$ toward OH.

![Figure 1](image1.png)

Fig. 1. Transient spectra observed in electron pulse radiolysis of 0.1 mM CCA solution. Insets in panels (a) and (b) are kinetics in various concentrations (0.1-0.5 mM) of CCA solutions at 720 and 360 nm, respectively, corresponding to signals attributed to $e^\cdot_{aq}$ and (OH-CCA), respectively.

3.2. $\gamma$-Ray Radiolysis

$G$(7OH-CCA) obtained with $^{60}$Co $\gamma$-ray irradiation under 5 different conditions are shown as symbols in Fig. 2. In results of Fig. 2 (a), it is possible to the responsibility of only reaction between CCA and $e^\cdot_{aq}$ to production of 7OH-CCA. As can be seen, it is clear that $e^\cdot_{aq}$ is negligible for the production of 7OH-CCA in lower CCA concentration. However, the increase of $G$(7OH-CCA) with increase of CCA concentration is due to decrease of scavenging efficiency of $t$-BuOH toward OH.

![Figure 2](image2.png)

Fig. 3. Yields of fluorescent product, 7OH-CCA, in 0.1-26 mM CCA solutions under five different conditions as a function of scavenging time scale of CCA toward OH. The symbols and lines are experimental and simulation results, respectively. Conditions are as follows, (a): Ar saturated condition with $t$-BuOH addition, (b): $N_2$O saturated condition without any additive, (c): aerated condition without any additive, and (d): aerated condition with acetone addition.

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In results of Fig.2 (b), it is clearly seen that OH is much more responsible to production of 7OH-CCA than $e^-_{aq}$. Generally speaking, a larger number of OH exist at earlier time scale. However, 7OH-CCA production decrease at earlier time scale. This point can be explained competition between scavenging reactions of N$_2$O and CCA toward $e^-_{aq}$. This competition leads to decrease of conversion efficiency from $e^-_{aq}$ to OH, and then, total number of OH scavenged by CCA decreases with increasing CCA concentration.

In results of Fig.2 (c) and (d), concentration of acetone was varied to fix ratio of scavenging capacity of acetone to CCA. Specifically, acetone was added to scavenge constantly 90% of $e^-_{aq}$ and 15% of OH. It is clear that the amount of OH scavenged by CCA is related almost proportional to $G$(7OH-CCA), supporting that reaction between CCA and $e^-_{aq}$ does not affect production of 7OH-CCA.

Moreover, comparing the data shown as lines (b) and (c) at low scavenging time scale, $G$(7OH-CCA) values are comparable despite difference OH yields. It is known that OH yield under the condition (b) is twice higher than that under the condition (c) because $e^-_{aq}$, of which yield is almost same as that of OH under deaerated condition, is converted into OH by N$_2$O. Then, it is understood that 7OH-CCA is formed through two different reaction paths. One is elimination reaction which is reaction of two OH-CCA, and the other is disproportionation reaction through peroxy radical, OH-CCA-OO, which is decomposition of OH-CCA-OO into 7OH-CCA and HO$_2$.

The deduced reaction mechanism was introduced into simulation considering spur process to examine its adequacy. In the simulation, both rate constants of disproportionation and elimination reactions are assumed as $10^8$ M$^{-1}$s$^{-1}$. Obtained simulation results are shown as lines in Fig.2 At this moment, branching ratio between 7OH-CCA and other $n$OH-CCA products was necessary because it is known that there are five choices ($n = 4-8$) in the position at which hydroxy group is stabilized. Adjusting the branching ratio to 5.6%, simulation results agree quite well with experimental ones as seen in Fig. 2

3.3. Heavy-Ion Radiolysis

Based on the reaction mechanisms, it was concluded that aerated CCA solution without any additive is most suitable to estimate the yield of OH.

Figure 3(a) shows results of heavy-ion irradiation of the solution at HIMAC. As seen in the figure, $G$(7OH-CCA) value decrease with the increase of scavenging time scale, which corresponds to development of intra-track reaction with time going on from 6 ns to 2 µs. Comparing them with the data obtained with γ-ray, the yield of 7OH-CCA becomes 20% in the most significant case. Moreover, the yield decreased with increasing LET value, reflecting enhancement of intra-track reaction resulted from increased radical density in ion track. In order to discuss overall reaction rate of water decomposition radicals, each data series were normalized by the highest value at 6 ns of scavenging time scale as a standard, and the normalized yield is shown in Fig.3(b). It is clear from the figure that not only absolute yield but also normalized yield, which is an indicator of overall reaction rate of water decomposition radicals, decrease with increasing LET. This fact leads us to conclude that track of higher LET beam still keeps a denser track structure even in time range $10^{-9}$ to $10^{-8}$ s.
This page contains a scientific paper discussing the reactivity of CCA in the presence of different heavy ions. The paper includes a figure showing the scavenging time scale and normalized G values of 7OH-CCA measured after irradiation of five different heavy ions and OH. The absolute and normalized G values are plotted as a function of scavenging time scale for CCA toward OH in panels (a) and (b), respectively. The symbols indicate higher LET values for each ion.

4. CONCLUSION

First, reactivity of CCA was evaluated based on transient absorption spectroscopy obtained in electron pulse radiolysis experiment. Second, reaction mechanism in CCA aqueous solution leading to fluorescent product, 7OH-CCA, was investigated γ-ray radiolysis. By changing saturating gas and additive, it was deduced that there are two different ways, disproportionation or elimination, of stabilization from (OH-CCA) to 7OH-CCA, depending on existence of O₂. Finally, aerated CCA aqueous solutions were applied to yield determination of OH produced in heavy-ion irradiation. Decrease of OH yield with increasing LET and increasing scavenging time scale was revealed. Based on temporal behavior of normalized yield, it was concluded that track of higher LET beam still keeps a denser track structure even in time range 10⁻⁹ to 10⁻⁸ s. In addition, yield determination was thought to be possible with much lower dose such as several GY than typical value used in earlier works such as several ten GY.

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Development of New MicroStrip Gas Chambers for X-ray Applications

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ABSTRACT

To cope with the high intensities of new generation beams such as the X-ray Free Electron Laser, a novel detector design ensuring a steady performance is needed. One of candidate detectors is a MicroStrip Gas Chamber which is the first MicroPattern Gas Detector (MPGD). Although original MSGCs suffered from discharges and could not realize a high gas gain, we are developing Multi-Grid-Type MSGCs, which have grid electrodes between the anode and the cathode electrodes in order to obtain a stable electric field. M-MSGCs are fabricated in house by using recent Electron Beam lithography techniques. While studying these M-MSGCs, we have developed the concept of a fine-pitch MSGC for high counting rate applications. A very fine-Pitch MSGC or a “NanoStrip” Gas Chamber (NSGC) is promising for very high counting rate applications, as well as for fine resolution with high pressure heavy gas. The narrowest pitch between anodes has reached 30 µm with the anode width of 800 nm. Our first trial for 50 µm pitch plate showed a gas gain sufficient to detect 8 keV X-rays. We are currently studying the characteristics of these new fine-pitch MSGC with X-rays and seek the possibilities of MSGCs as a next generation detector.

Key Words: MSGC, NanoStrip Gas Chamber (NSGC), High counting rate and Fine pitch

1. INTRODUCTION

MicroStrip Gas Counters (MSGCs) are the oldest type of gas counters among micropattern detector developed by photolithography technique (Fig.1). With its narrow spacing electrodes, MSGCs overcome the limits of MultiWire Proportional counters such as the position resolution limit and the rate capability due to their finite wire-spacing. MSGCs are expected to be a good candidate for high intensity beam experiments performed for example at with synchrotron radiation and high energy facility. However the breakdown of electrodes by discharges was considered as one serious problem, which causes critical damage to the detectors. For this problem, we proposed a multi-grid-type MSGC (M-MSGC) [1-3]. M-MSGCs are equipped with many intermediate strips between the anode and the cathode strip. These strips are connected individually to a high voltage source to maintain intermediate potentials between the anode and the cathode. As a result, the electric field is stabilized by this grid potential, and therefore a very high gas gain can be achieved without discharges.
In addition to operating the M-MSGC in pulse mode, we are exploring the possibility of the charge integration mode, where pile-up pulses can also provide the intensity information of incident radiation as well. Although we lose the benefit of pulse counting, we can obtain a wide dynamic range in this operating mode. While studying these MSGCs, we have developed the concept of a fine-pitch MSGCs under high counting rate (Section 2). Based on our studies so far, the very fine-pitch MSGC or “NanoStrip” Gas Chamber (NSGC) appears very promising for very high counting rate applications, as well as for fine resolution with high pressure high-Z gas. We fabricated a first trial 50 µm pitch plate and tested its characteristics.

2. DYNAMIC RANGE MEASUREMENT

2.1. Charge-Integrating-type M-MSGC

For very high counting rate applications, pulse counting mode is not so suitable since the typical pulse width of MSGCs is about several hundred ns. Therefore we adopted the use of charge integrating readout of M-MSGC [4] where the performance at a high counting rate is only limited by a space charge effect. We designed an M-MSGC plate for charge integrating readout from cathode strips. Anode pitch was 400 µm and 4 grids were inserted between the anode and the cathode. The widths of anode, grid1, grid2, grid3, and grid4 were 10, 20, 25, 35, and 40 µm, respectively. Gaps between neighboring strips were set to 10 µm. Cathode strips were separated so that we can read current signals from individual cathode strips. We connected 32 cathode strips to input channels of integrating amplifier ASICs. For readout ASICs, we utilized two HX2 [5] chips for 32 cathode strips. HX2 contains an array of 16 integrating amplifiers, each with a 10pF feedback capacitor. The integration period can be varied over the range around 5 µs to around 100 ms by an external clock frequency, which provides enough dynamic range. The integration period is followed by the readout period where the 16 channels are multiplexed onto an analog output bus.
2.2. X-ray Irradiation from Top of the Plate

As we already reported in our previous paper [4], the charge integrating mode can assure the measured linearity of output current from the cathode as much as $10^9$ cps/mm² at the gas gain of 100, although the curve shows that saturations, called space charge effect start at around $10^8$ cps/mm². Since detailed investigation into the saturation region of such a high counting rate operation is not well reported in the literature, we carefully analyzed the characteristics of the detector. When we increased the gas gain, saturation occurred at a lower counting rate, which shows that the effect is due to the space charge.

2.3. X-rays Irradiation Parallel to the Strips

To overcome the saturation by the space charge, we are trying to enlarge an avalanche region per unit area. First we have tested with another geometry, where we irradiated 8 keV X-rays from the side of the plate (parallel to the strips, see Fig.1) so that we can take advantage of the full depth of the anode electrode effectively. We performed the experiment at Super Photon ring-8 GeV (SPring-8) [6] with 8 keV X-rays in the flowing gas mixture of Ar (70%) + CH4 (30%). The result is shown in Fig.2. We observed linearity of the detector response up to $10^{10}$ cps/mm² in the charge integrating mode.

![Fig.2(a)](image1)

**Fig.2(a)**

![Fig.2(b)](image2)

**Fig.2(b)**

Fig.2. The direction of the X-rays is shown with simplified schematic of the experimental set up. (a) Top view of the set up (The chamber is shown in dot rectangle. (b) Side view.

![Fig.3](image3)

**Fig.3.** Counting rate characteristics of a charge integrating operation.

![Fig.4](image4)

**Fig.4.** The comparison of the charge collection process for a conventional MSGC (left) and a finer pitch MSGC (right). Either plate is described without grids from the cross section view.
2.4. Towards Fine-Pitch M-MSGC

We have confirmed our idea to increase the dynamic range by enlarging avalanche area by the result at SPring-8. Linear region of the same plate was 3 decades wider by enhancing the avalanche region by changing the direction of X-rays irradiation as shown in Fig.3. We consider that a plate which has a smaller pitch length than electron range, about 400 µm in the gas mixture of Ar (70%) + CH4 (30%) at 1 atm, can work under much higher counting rate. We consider that narrow pitch of a finer pitch MSGC enhances the counting rate characteristics by dispersing charges of the cloud into several electrodes. The idea is shown in Fig.4 with comparing the process of conventional MSGC.

3. NanoStrip Gas Chamber

3.1. Plate Design

We have decided to develop a fine-pitch MSGC that can increase the area of the avalanche region and suppress the space charge effect. As a first trial, we have fabricated a 50 µm pitch M-MSGC. Its anode width is chosen as small as 800 nm. In the following we call this implementation a “NanoStrip” Gas Chamber (NSGC) in this aspect. Our first prototype 50 µm pitch of NSGC has 2 grids. Their widths are 3 µm. 40 sets of strips are aligned and each strip is 20 mm long (Effective area is 2 mm x 20 mm).

3.2. X-ray Test

The NSGC was tested with 8 keV X-rays at KEK, Photon Factory, BL-14A. A Gas Chamber is filled with an Ar (70%) + CH4 (30%) gas for all experiments. The detector is successfully operated. The gas gain of 280 is achieved. Fig. 5 shows waveforms obtained from the anode and the cathode at this condition. The rise times of signals are around 1 µsec. This rise time is not so fast as what we expected but it is because of the high resistance of narrow anode electrode. We measure its pulse height spectrum as well (shown in Fig.5). A relative energy resolution of 22% (FWHM) was achieved by employing a summing amplifier, we able to improve the relative energy resolution to 15% (FWHM).

3.3. Uniformity

We scanned the plate with collimated (100 µm in diameter) beam irradiation with 10 keV X-rays at KEK Photon Factory, BL-14A. We scanned the beam across the strips at the middle of the anode length in 100 µm pitch. The detector was operated at a gas gain of 220. We could observe signals at the effective area. We show the map of the scan in Fig.6. At the edge of the plate, the signal becomes larger because of the electrical field getting higher in this configuration.
4. CONCLUSIONS

We have made a very fine pitch M-MSGC as the first test of NanoStrip Gas Chamber (NSGC). The fabricated 50 µm pitch NSGC was successfully operated at a gas gain of 280. The energy resolution was 22 % (FWHM) for Fe-55 X-rays. We found that our NSGC design operates as conventional M-MSGCs. While our approach of a NSGC could have a significant impact high-intensity beam experiments, more detailed studies with refined and high-rate X-ray measurements are needed to optimize and demonstrate the full capabilities.

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Development of Geologic Repository Models for Site Selection and Design Optimization

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ABSTRACT

This paper presents a new radionuclide transport model to evaluate performance of a geologic repository for high-level radioactive waste. The model is based on a compartment model, and a Markov-chain model. A domain is divided into an array of compartments, and a transition probability matrix describes transport among the compartments. The model is demonstrated for a hypothetical repository in porous rock formations. A three-dimensional, non-uniform groundwater flow field is created numerically using the finite element method. The transition probability matrix is constructed based on the velocity field and hydraulic dispersion coefficients. The results show that this transport model can capture the change of hydraulic properties in the domain by the repository structure and material degradation. They also suggest that the impact of engineered barrier systems should be evaluated in comparison with the site conditions such as the ambient hydraulic head gradient.

Key Words: Geologic Disposal, Compartment model, Markov-chain process

1. INTRODUCTION

As a final disposal for high-level radioactive waste, many countries have proposed the deep geologic disposal under a water table. In Japan, the repository will be constructed in the water-saturated region at 300-500m deep from the surface. It consists of an array of tunnels containing the waste canisters, ranging over several square kilometers.

Previous performance assessments [1] are based on a simplified radionuclide transport model with conservative parameters and assumptions. It assumed a uniform flow of groundwater though repository, neglecting the repository structure. Although such a model is sufficient to evaluate the safety margin of the repository, it is not suitable to optimize the repository design, since each component should be analyzed in terms of its impact on the radionuclide transport.

For example, degraded concrete wall and excavation-damaged zone (EDZ) around tunnels could become a fast path of transport. In order to hinder the flow through the repository, several engineered barrier system will be introduced such that the tunnels will be filled by clay material, and concrete/clay plugs will be installed at the end of tunnels. The transport model must capture such differences of hydraulic properties, in order to compare the different designs. In addition, in order to analyze many different designs, the model has to be flexible and computationally inexpensive.
Previous works proposed compartment-model approach to describe the transport in the repository region [2]-[3]. In the compartment models, the domain is divided into an array of compartments. A Markov-chain process describes the transport by using transition probabilities of particles among the compartments. The advantage of this approach is its flexibility to include different geometries and various phenomena by probabilistic interpretation. It can also utilize results from the particle-tracking methods in a small-scale domain, to simulate the transport in larger domain with less computational power.

The previous models, however, only consider the uniform groundwater velocity field and one-dimensional domain. This paper presents the extension of compartment model in a non-uniform and three-dimensional domain. The transition probability is derived from more fundamental interpretation of transport process for a non-uniform domain.

We apply the model to a hypothetical repository in porous rock formations. Several repository designs and degradation of concrete wall are considered. A three-dimensional, non-uniform groundwater flow field is calculated numerically using the finite-element method for each set of the repository parameters. The transport is simulated to obtain a release rate to the downstream.

2. MODEL

2.1. Compartment Model

The domain is divided into an array of $n$ compartments ($i = 1, 2 \ldots n$). Let $N_{s,i}$ and $N_{a,i}$ denote the number of solute particles and sorbed particles in compartment $i$, respectively. The mass balance equations are written as a first-order differential equation with time as,

$$\frac{dN_{s,i}}{dt} = -r_{a,i} N_{s,i} + r_{d,i} N_{a,i} + \sum_{j=1 \atop i \neq j}^{n} q_{ij} N_{s,j} - \sum_{j=1 \atop i \neq j}^{n} q_{ij} N_{s,i} - \lambda N_{s,i},$$

$$\frac{dN_{a,i}}{dt} = r_{a,i} N_{s,i} - r_{d,i} N_{a,i} - \lambda N_{a,i}, \tag{1}$$

where $r_{a,i}$ and $r_{d,i}$ are the adsorption and desorption coefficient, respectively. $q_{ij}$ is the transfer rate of solute particle from compartment $i$ to compartment $j$. $\lambda$ is the radioactive decay constant. Although decay chains are not considered here, Equation (1) can be easily extended. The relationship between solute particles and sorbed particles is usually assumed to be linear as $N_{s,i} = \alpha N_{a,i}$.

Although Marseguerra et.al. derived the transfer rate $q_{ij}$ by the analogy to an advection-dispersion equation [3], they considered only a uniform domain. We take a different approach to derive the transport parameters, in a similar manner to Costa et.al.[4].

First, we consider the discrete-time frame by the forward Euler method as,
\[
N_{s,i}(t + \Delta t) = N_{s,i}(t) + \left( r_{a,i}/\alpha_p - r_{a,i} \right) \Delta t N_{s,i} + \sum_{j=1}^{n} q_{ji} \Delta t N_{s,j} - \sum_{i=1}^{n} q_{ij} \Delta t N_{s,i} - \lambda \Delta t N_{s,i}.
\] (2)

We define a transition probability, \( P_{ij} \), for a particle to move from compartment \( i \) to \( j \) in a time interval \( \Delta t \). Since the number of particles is large, the weak low of large numbers is applied. We consider “averaged” movement of particles so that the transition probability is interpreted to the transfer rate as \( P_{ij} = q_{ij} \Delta t \).

In the numerical calculation, the number of particles in each compartment at \( t = k \Delta t \) is stored in a vector \( a^{(k)} = (N_{s,1}(t), N_{s,2}(t), \ldots, N_{s,n}(t)) \). The transport in one interval is described by the matrix-vector operation as,

\[
a^{(k+1)} = a^{(k)}[P] + a^{(k)}[r] - \lambda \Delta t a^{(k)}. \] (3)

where \( [P] \) is an \( n \) by \( n \) transition probability matrix defined as \( [P] = P_{ij} \) and \( [r] \) is a sorption-term matrix defined as \( [r] = (r_{d,i}/\alpha_p - r_{a,i}) \Delta t \) for \( i = j \) and \( [r] = 0 \) otherwise.

### 2.2. Derivation of Transition Probability

First, we consider the transition probabilities from compartment \( i \) in a one-dimensional array. It is assumed that particles stay in the original compartment or move only to the adjacent compartments in a small time interval \( \Delta t \). The destination compartments are prescribed as \{\( i-1, i, i+1 \} \). Let \( X_i = \{-1, 0, +1\} \) be the magnitude of displacement of a particle in \( \Delta t \). We can define three transition probabilities as \( P_{i,i-1}, P_{i,i}, \) and \( P_{i,i+1} \). The expectation and the variance of \( X_i \) is obtained as,

\[
E[X_i] = P_{i,i+1} - P_{i,i-1},
\]

\[
\text{Var}[X_i] = (P_{i,i+1} + P_{i,i-1}) - (P_{i,i+1} - P_{i,i-1})^2
\] (4)

They can be directly determined by the particle-tracking method. Here we consider the transport in a non-uniform porous medium, where the hydraulic dispersion causes the distribution of particles. The hydraulic dispersion coefficient is the sum of a diffusion coefficient and a mechanical dispersion coefficient.

Consider three compartments \{\( i-1, i, i+1 \} \), having different length of compartments \{\( d_{i-1}, d_i, d_{i+1} \} \), porosities \{\( \varepsilon_{i-1}, \varepsilon_i, \varepsilon_{i+1} \} \), and dispersion coefficients \{\( D_{i-1}, D_i, D_{i+1} \} \). The concentration gradient is assumed to drive the dispersion, following common working hypothesis. The transition probabilities by dispersion from \( i \) to \( i+1 \) and from \( i \) to \( i-1 \) are obtained by multiplying the average dispersive flux by the time interval as,

\[
P_{disp,i,i+1} = \frac{\Delta t}{\varepsilon_i d_i} \frac{\varepsilon_i D_i + \varepsilon_{i+1} D_{i+1}}{d_i + d_{i+1}}, \quad P_{disp,i,i-1} = \frac{\Delta t}{\varepsilon_i d_i} \frac{\varepsilon_i D_i + \varepsilon_{i-1} D_{i-1}}{d_i + d_{i-1}}
\] (5)
The expectation value and variance are determined as $E_{\text{disp}}[X_i]$ and $\text{Var}_{\text{disp}}[X_i]$ from definitions. The total expectation and variance is obtained by adding the advection term $v_i \Delta t/d_i$ as,

$$E[X_i] = \frac{v_i \Delta t}{d_i} + E_{\text{disp}}[X_i]$$
$$\text{Var}[X_i] = \text{Var}_{\text{disp}}[X_i]$$

(7)

The transition probabilities are obtained by substituting Equation (8) in Equation (4). The time-step size and compartment size is restricted to avoid the probabilities to be negative. For multi-dimensional cases, the transition probabilities are obtained in the same manner by considering each direction independently.

### 3. DEMONSTRATION

Figure I shows a three-dimensional hypothetical repository. It consists of tunnels (inner cross-sectional area $2 \times 2 \text{m}^2$), which are surrounded by concrete wall (0.4 m thick) and EDZ (0.6m thick), and host rock. 6 waste forms ($0.8 \times 0.8 \times 0.8 \text{ m}^3$) are located in the tunnels. Clay Plugs (0.6m thick) are installed at the end of the tunnels.

![Figure I. Hypothetical Repository (a) Horizontal plane (b) 3-D view sliced by $x=0$ plane](image)

The domain is divided into $n = 86,112$ compartments. The number of compartments in each direction is $(n_x, n_y, n_z) = (78, 48, 23)$. The boundary condition is applied as an equi-potential plane at the right and the left side plane ($x = -15.8$ and $x = 15.8$) in Figure I (a). The ambient constant hydraulic gradient is set to be $c = 1 \times 10^{-3}$ or $1 \times 10^{-4}$ from left to right.

Table I shows input parameters, which are hydraulic conductivities $K$, porosities $\varepsilon$, and pore diffusion coefficients $D$. Each material is assumed to be a uniform porous medium. Since the repository scale is small, only effective diffusion coefficient is considered.
Table I. Hydraulic and transport parameters for each material in the repository region.

<table>
<thead>
<tr>
<th>Material</th>
<th>Rock</th>
<th>EDZ</th>
<th>Concrete</th>
<th>Backfill</th>
<th>Plug</th>
<th>Waste Form</th>
</tr>
</thead>
<tbody>
<tr>
<td>- logK (m/s)</td>
<td>8*</td>
<td>6*</td>
<td>5* / 13*</td>
<td>6*** / 12*</td>
<td>12*</td>
<td>K = 0***</td>
</tr>
<tr>
<td>Porosity</td>
<td>0.347*</td>
<td>0.366*</td>
<td>0.333*</td>
<td>0.333*</td>
<td>0.307*</td>
<td>0.100***</td>
</tr>
<tr>
<td>D (m²/yr)</td>
<td>4.37E-3**</td>
<td>2.37E-3**</td>
<td>4.37E-3**</td>
<td>4.62E-2**</td>
<td>4.37E-3***</td>
<td>4.37E-3***</td>
</tr>
</tbody>
</table>

* From Reference [5], ** From Reference [6], *** Assumed values

Different hydraulic conductivities are assumed for intact or degraded concrete ($1 \times 10^{-13}$ m/s or $1 \times 10^{-5}$ m/s), and clay backfill or rock backfill ($1 \times 10^{-13}$ m/s or $1 \times 10^{-6}$ m/s). The impact of plugs is also compared. The following cases are considered:

1. No Plug / Clay backfill/ Intact concrete
2. No Plug / Clay backfill/ Degraded concrete
3. No Plug / Rock backfill/ Degraded concrete
4. Plug / Clay backfill/ Intact concrete
5. Plug / Clay backfill/ Degraded concrete
6. Plug / Rock backfill/ Degraded concrete

The non-uniform groundwater velocity field is formulated by the Darcy’s law, assuming time-independent and incompressible flow. It is calculated numerically by the finite element method. Transport is simulated based on the model described in the previous section. It is assumed that particles are released at the waste form locations at time 0, diffuse out from the waste form and transport thought the repository region. Absorbing boundary conditions are prescribed at the six planes. Decay and sorption is neglected, in order to focus on the effect of hydraulic properties. Although the release from the domain occurs at all the six planes, only the release from the downstream plane ($x=15.8$) is evaluated, since it has the highest peak.

![Figure II Release rate from the domain to the downstream: $c = 1 \times 10^{-3}$](image1)

![Figure III Release rate from the domain to the downstream: $c = 1 \times 10^{-4}$](image2)

Figure II and III shows the release rate from $x=15.8$ plane to down-stream for the case with $c =$
1 × 10^3 and 1 × 10^4, respectively. It can be considered that Figure II is an advection-dominant case and Figure III is a diffusion-dominant case.

Figure II shows the effect of the plugs to reduce the peak release rate by factor of 1.5. Although the degradation of concrete increases the release rate, the impact is relatively small. The difference of backfill material does not have impact, since a fast transport path in the degraded concrete region dictates the transport.

In Figure III, peaks of the release rate are lower than those in Figure II. Since the diffusion is dominant, the release from the other planes is increased. In this case, the plugs and degradation of concrete, and the difference of backfill material do not show significant difference.

This could suggest that the engineered barrier system should be designed based on the site conditions. If the hydraulic head gradient is small, the hindrance of groundwater flow will not contribute to enhance the safety. In fact, this demonstration is quite simplified so that it does not take into account the change of chemical condition by concrete degradation, nor sorption by clay material. More realistic chemical and geochemical processes should be included in the model.

3. CONCLUSIONS

This paper has presented a new transport model concept based on the compartment model and the Markov-chain model. The transition probabilities are derived for the transport in the non-uniform porous medium, by conserving the expectation and variance of the displacement. The model has been demonstrated for a hypothetical repository with three-dimensional, non-uniform groundwater flow field. It has shown that this model can capture the difference of the repository design and material degradation. The results suggest that the impact of such difference also depends on the site conditions. The model should be further improved to include chemical and geochemical processes to be used for the actual optimization of the repository design.

REFERENCES

Study on a Sensor Network System with a Self-Maintenance Function for Plant Monitoring System

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ABSTRACT

This paper presents a new network-based concept of real-time plant monitoring based on the wireless sensor network. The wireless sensor network is a novel technology to perform distributed sensing tasks, especially for applications such as environmental monitoring, smart spaces, medical systems and etc. The advances in MEMS technology, sensing technology and wireless communications have enabled the development of real time, detailed, wide-range, low-cost, low-power and multifunctional wireless sensor network. On the other hand, semiconductor devices like ICs suffer from radiation damage. We must consider this effect for applying wireless sensor network devices to nuclear power plants. Radiation may produce one-time noise signals in the sensor network, or it may damage the wireless sensor network device. Therefore monitoring system will be unreliable for nuclear power plants in ordinary way of composing wireless sensor networks.

In this study, we develop a new concept of a robust wireless sensor network, that has a tolerance against both the partial node’s failure and packet errors; realized by a Self-Maintenance function. The Self-maintenance function is a function that enables an artifact to find, diagnosis and fix the trouble automatically and maintain itself. So far some approaches have been tried to realize robust monitoring system by applying the idea of multiplex system, based on “2 out of 3”, but this requires a large amount of the hardware and is not suitable for sensor network systems. Here, we designed a sensor network system with Self-Maintenance function based on qualitative reasoning technique for robust wireless sensor network system, and an instrument network based on ZigBee has been set up for investigations. It is found that ZigBee-based instrument networks are robust but susceptible to interference from traditional way of sensing.

Key Words: ZigBee, wireless instruments, sensor networks, health monitoring

1. INTRODUCTION

Structural health monitoring (SHM) [1] is a vast, interdisciplinary area of research whose literature spans several decades. Wireless sensor networks promise cheap and dense instrumentation for structural monitoring [2]. Recent work has demonstrated the feasibility of continuous structural data collection using a wireless network [3, 4, 5]. For these characteristics of wireless sensor networks, it could be a great application and solution for real time monitoring of nuclear power plant. Real time monitoring enables safe operation of nuclear power plants and
also, wireless sensor networks frees operators from worrying about fire caused by cable deterioration.

In other hand, since inside a nuclear power plant are under radiation field, ionizing radiation can cause unwanted effects in semiconductor devices such as flipping the state of memory cells. These unwanted changes are known as soft errors and are the most common type of single event upsets (SEUs). While statistically unlikely during the operating lifetime of even the largest field programmable gate array (FPGAs), circuit components within programmable logic devices (PLDs) such as configuration memory cells, user memory, and all those components made by semiconductors can be affected. When wireless sensor networks are implemented in nuclear power plant for structural health monitoring in ordinarily way (as we do in non-radiation field), soft errors may lead us to miss detection of failures.

In this paper, we describe the design and evaluation of self-maintenance sensor network, a sensor network system that has a tolerance against soft errors and partial nodes failure. The system has tolerance against SEUs, able to detect sensor node’s damage perfectly and avoid false-positives and false-negatives and realizes reliable real time monitoring.

2. SELF-MAINTENANCE ALGORITHM

Although rapid advancement of the technology, still living body is superior to artifacts in some respects. The difference frankly appears to the mechanism of “Production” and “Maintenance.” In other words, while living body has functions of form production and restoration, an artifact depends on outside of itself for this function. If an artifact had composition like the living body, composed of a numbers of cells, an artifact might be able to achieve the function like Self-Restoration. Therefore, for artifacts that are consisted of homogenous units, like sensor networks, we considered supplement will be one of a method for maintaining artifacts. The most fundamental function of sensor networks are maintained by supplementing damaged node with other nodes. Before implementing this self-maintenance algorithm into wireless sensor network in real world, we held a simulation in computers and designed wired device for testing.

2.1. Damage detection

Detecting a failure of a sensor node is the most important part of the self-maintenance sensor network system. It is impossible to estimate a failure from just looking at one sensor's value. We cannot tell whether the irregular value is caused by a damage of where it is monitoring (plant) or sensor itself. The system with accurate decision of node’s failure is strongly required for avoiding false-negative and false-positive. In our system, damage detection of sensor is based on the idea of model-based reasoning, that “nodes close to each other should some relationship.” Failure can be detected when sensors are connected into a network and handling those comprehensive output from the sensors. That is to say, verify whether sensor is working correctly or not by comparing value of it’s neighbor sensors and detect it’s failure. In other words, sensor's reliability is determined by sharing the information of each sensors and handling those information by network consisted of sensors. Consequently, we consider the system will be able to detect sensor's failure by giving a parameter “reliability index” to each sensor. Figure 1 describes the concept reliability index.
2.2. Reconfigurable Network

When broken sensor is detected, the system will exclude the sensor from the network and reconfigure the network with only healthy nodes. This function enables to minimize the damage of the network from a partial failure. Without this function, damaged sensor node will cause packet collisions and it will effect to entire network.

2.3. Simulation Result

In this simulation, we inspected it about the case that could happen frequently. A concrete example will be given to help clarify the system we designed so far and its result is shown in below. 16 nodes are placed and 1 of them is damaged by radiation (Fig.3-1). SEUs are occurring in the damaged sensor, and irregular value is coming. The red graph in Fig.3-2 is the output from damaged sensor, and it is outputting random values every time. Partial failure of network nodes will be a cause of false-negative the power plant.
Fig.3-3 is a graph after applying self-maintenance function to the network. Node in failure was detected correctly and wrong outputs are excluded from monitoring system. The result proved that it works when failure rate is 1/16 (6.25%). Since it is known that under 1000Gy of radiation, CMOS chip made from 0.15µm SOI process, the error bit rate will be 2.2% in average [5], the self-maintenance function is effective enough under those condition.

Fig.3-1 Layout                          Fig.3-1 Simulation of SEUs
Fig.3-2 Outputs from system

2.4. Wireless Device

From the simulation we held, we found there are enough possibility to place wireless sensor network into radiation condition by applying our self-maintenance algorithm. Therefore, we designed wireless sensor network device with ALTERA’s CPLD, 10bit digital temperature sensor, and Xbee wireless module. Since these are made through 0.18µm CMOS processes, that is larger process than 0.15µm, we consider that it has more tolerant against radiation.

Fig.4 Wireless Sensor Network Device
At this moment, we could only implement temperature sensor to the device, but we can also add vibration sensors, pressure sensors, strain gauges and noise sensor for health monitoring in many directions. Moreover, distributed processing system and this radio frequency module, called Xbee is manufactured under ZigBee protocol, the maximum size of the network will be up to 65000 nodes. The new health monitoring system followed by enormous number of sensors will enable a efficient innovational way of managing nuclear power plants.

3. CONCLUSIONS

In this paper we found a new fault tolerance algorithm for wireless sensor network. Our design and validation of self-maintenance system suggests that, sensor networks made of semi-conductors have enough possibility to be used for avoiding SEUs caused by radiations. We conjecture that the self-maintenance wireless sensor network architecture might be great applicable to nuclear power plant monitoring system, but have deferred an examination of this question to future work.

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Recent Developments of Compact Neutron Generators at LBNL

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ABSTRACT

We present recent work in the development of two new types of compact portable neutron generators, axial and field desorption, to be used for associated particle imaging and brachytherapy. The lightweight and small size of these neutron generators allow for greater versatility than current neutron sources that require heavy shielding and large spaces to operate. In these sources, neutrons are created via low energy (60-100 keV) nuclear reactions of deuterium with tritium or deuterium ions, yielding neutrons with very distinct energies. These generators produce neutrons via deuterium and tritium ions incident on a deuterium or tritium self-loaded Ti target where the nuclear reactions occur. Preliminary ion source testing demonstrates the ability of these sources to operate at pressures of 2 to 8 mTorr and 10^-6 to 10^-5 torr and yield as much as 10^7 and 10^5 neutrons per second for the axial and field desorption sources, respectively.

Key Words: Neutron generator; RF-induction ion source, field desorption ion source

1. INTRODUCTION

Various ion sources have been developed by the Plasma and Ion Source Technology (PIST) group at Lawrence Berkeley National Laboratory (LBNL) for use in the development of IC manufacturing, neutral beam injectors, and neutron generators. These neutron generator systems are useful for applications including boron neutron capture therapy (BNCT), neutron activation analysis (NAA), prompt gamma activation analysis (PGAA), and pulsed fast neutron transmission spectroscopy (PFNTS) [1,2]. Inductively coupled, RF-driven multicusp ion sources were used extensively to develop these generators. Plasmas with high current density, high atomic ion fraction of deuterium or tritium, and long lifetimes were generated using a 13.56 MHz RF power supply [3]. However, these neutron generator systems require the use of a high-powered RF-driven ion source and external water-cooling, making portability a significant issue. Therefore, development of a portable neutron generator that does not require the use of a large RF power source or water-cooling would be advantageous.

In this article, the development of two new types of compact, portable neutron generators, designed to perform in the areas of associated particle imaging (API) and brachytherapy, will be presented. The first source is an RF-driven ion source with CW operation at a very low RF power and small beam spot size for associated particle imaging, yielding as much as 10^7 D-T neutrons per second. The second type of neutron generator differs drastically from typical neutron generators, utilizing field desorption of ions from microfabricated tungsten coated silicon tips. Previous work confirms that desorbed gas atoms on the surface of a metal tip can be emitted as ions under strong electric fields [4]. Desorbed deuterium and tritium ions in the microfabricated needle tips are emitted and accelerated onto a titanium target at energies greater than 60 keV to produce up to 10^5 n/s. The microfabrication process used to create 10^7 tips on a 1 cm squared silicon wafer will be discussed.
The field desorption source is more compact than the axial neutron source, without the need for a separate RF supply, making it a good candidate for medical applications. However, this source needs further development. Design of the neutron tubes and various ion source schemes will be presented in the following chapters, along with the presentation of preliminary experimental results.

## 2. ASSOCIATED PARTICLE IMAGING NEUTRON GENERATOR

Associated particle imaging is an active neutron interrogation technique that makes use of the deuterium-tritium fusion reaction, Eq. 1, to generate a neutron and an associated alpha particle.

\[
\text{^2D + ^3T} \rightarrow \text{^4He + ^1n} \quad \text{E}_n = 14.1 \text{ MeV, E}_{4\text{He}} = 3.5 \text{ MeV}
\] (1)

The alpha particle is detected using a sensor consisting of a scintillator, fiber optic plate, and a thin aluminum coating to stop light and scattered ions from reaching the light detector. Information on the associated neutron, such as the neutron trajectory and position, can be determined via time of flight measurements using the alpha particle sensor. A gamma detector is set up near the examination object to detect characteristic gamma rays produced from neutron interactions with the object. The information from the alpha detector along with the geometry of the system determines the velocity of the associated neutron, while data from the gamma detector determines the object material and where the interaction took place within the object. The data can be analyzed in real time to generate 2D and 3D density contour maps of the object which can be used to determine its elemental composition [5]. This allows for single sided inspection of small sealed packages. A schematic of the API concept is provided in Figures 1.

![Figure 1. API geometry showing the alpha particle detection and gamma detection scheme](image)

### 2.1. Low power RF-driven multicusp ion source

In the past, inductively coupled RF-driven ion sources have been used to produce high current and high atomic ion ratios for neutron generators. However, these sources operate at high rf powers (1 to 5 kW) [6] and require a great deal of water cooling. The axial neutron source for API must be operated at a low enough power to be cooled by air and still provide sufficient beam current to achieve the required neutron yield of \(10^7\) neutrons per second with continuous-wave (CW) operation, lower than for typical D-T accelerator based neutron generators. Also, the neutron tube must be completely sealed and non-water cooled to avoid any tritium contamination and be easily portable. Therefore, further investigation of a low powered rf-driven ion source is necessary to meet axial neutron generator requirements for API.
Recent Developments of Compact Neutron Generators at LBNL

The neutron yield is affected by the ion beam current density, energy, and atomic ion fraction, so the design of an ion source must address these parameters. For example, a neutron yield of about $10^7$ n/s requires 30 to 40 $\mu$A of beam current for a beam energy range of 60 to 80 keV and 70-80% monoatomic ions [7]. Monoatomic ions carry the full beam energy, where as diatomic or triatomic ions have the beam energy spread among 2 or 3 ions, which results in lower fusion cross section per ion compared with monatomic ions. Previously, experiments using small cylindrical sources operated at low powers at pressure greater than 40 mTorr inside the chamber [8]. However, this pressure is too high for beam acceleration voltages of 60 kV or greater, due to sparking in the acceleration column, producing high electron current that can damage the scintillator. Therefore, the pressure must be kept low enough to prevent this discharge or dark current from being produced in the target chamber.

Many different ion sources with various antenna designs and frequencies were tested [9, 10]. It was found that a small source length-to-diameter aspect ratio of 1/3 or lower, with a flat spiral antenna and multicusp magnets provided the highest plasma density and atomic species fraction at pressures of < 10 mTorr. The length of the plasma source needs to be no more than 4-5 cm since the skin depth is on the order of a few cm [11]. Therefore, the RF fields will not penetrate much beyond that in the plasma, and the maximum plasma density will be in the first few cm, where the extraction takes place. The diameter of the source needs to be sufficiently large enough such that the mean free path is large enough for the primary electrons to gain enough energy to ionize the neutrals at pressures of less than 10 mTorr. The prototype source was constructed using a half closed quartz glass cylinder 12 mm in diameter and 4 mm tall with an aluminum plasma electrode. Figures 2(a) and 2(b) show a photo and schematic of the ion source, respectively.

The ion source has been tested in stable CW mode at a minimum rf power of 125 W and pressure of 8 mTorr, which is low enough to operate at under 80 kV of extraction voltage. Figures 3a and 3b show the current density and the atomic species fraction as a function rf power. From the experimental data, the measured extracted current at 150 W was approximately 30 $\mu$A, and the H$^+$ species was close to 70%, using a 1-mm diameter extraction aperture, which is more than sufficient to achieve the desired neutron yield at an 80 keV beam energy.

![Figure 2. (a) Picture of the preliminary ion source, (b) Preliminary ion source drawing.](image)
2.2 Neutron tube design

More measurements of the heat loading on the antenna and target must be collected to ensure stable operation without water cooling at long periods of time, before the final source can be constructed. Also, a small beam spot size is necessary for API, which can be improved by future work in ion optics. IGUN simulations for ion optics suggest the ability to achieve a beam focus size of 1.1 mm at the target for an aperture of 0.8 mm at 60 kV, for a pure H\(^+\) plasma and a total ion current of 36 μA. These calculations will be experimentally tested with the prototype in the future.

Plasma ignition poses another issue with the neutron generator assembly. For inductively coupled plasmas, high pressures are typically needed to start the plasma. After the plasma is ignited, the pressure can be turned down and the matching network tuned to the proper levels [10]. However, in a sealed tube environment, the pressure inside the source is difficult to adjust. A spark gap ignition technique of placing a sharp point biased at several kV next to a grounded metal surface can provide the necessary electrons needed to start the plasma. Tests using a sharp tungsten needle inserted through the plasma electrode and biased to 8 kV with respect to the plasma electrode, created a spark which generated enough primary electrons at 2 mT with 1000 W of rf-power, sufficient enough to ignite the plasma. The spark ignition technique will be further developed in future research.

3. FIELD DESORPTION ION SOURCE

Various radioactive sources such as Sr-90, Ra-226, and Ir-192 have been used to treat cancer, most notably prostate cancer. Current brachytherapy methods involve delivering a radioactive source inside the body to the cancer tumor via a catheter and leaving it there for a few minutes to a few hours, depending on the dose needed [12]. Using a neutron generator which is small enough to be handheld and which can be turned on and off would help to eliminate the use of radioactive sources and reduce the complication of injection during treatment. The neutron generator must be very small and reliable to be used in a hospital regularly. One of the more novel ion sources being investigated for a compact and portable neutron generators is the field desorption ion source [13]. This source utilizes tiny microfabricated “needle-tips” and high electric fields applied to the sharp points to desorb and ionize surface atoms, e.g. gas atoms, which have adsorbed into the metal tip surface. This type of source is radically different from other ion sources currently used in neutron generators. Figure 4a shows the prototype ultra-compact field desorption neutron generator schematic. The device can be preloaded with deuterium gas, tritium gas, or a mixture of both gases depending on the desired neutron energy and spectrum. For the case of the D-D reaction, the D\(^+\) atomic deuterium
ions are produced via electrostatic field desorption of deuterium that is adsorbed from the gas onto the surface of the tungsten coated field emitter tips. When a voltage is applied between the tips and the gate electrode, an electric field is applied to the tips. When the field is high enough to lower the potential barrier below the threshold for the ejection of an adsorbed gas atom from the metal surface, the gas atom becomes ionized and ejected away from the metal surface. The field strength needed for field desorption is in the range of 10-40 V/nm [14]. Depending on the voltage applied to the tungsten tips, the distance between the tips and the target can vary on the order of several millimeters. The generator is operated in pulsed mode, to allow time for tungsten tips and titanium target to replenish with deuterium or tritium atoms. The neutrons can be produced on-demand by turning the high voltage on and off, thus eliminating excessive shielding needed when the source is not operational. The amount of ion current that can be extracted from one tip is on the order of picoamps. However, the fabrication of a 1 cm² surface with over 10⁷ needles. Conservative estimations and calculations have shown that assuming a deuterium coverage density of 10¹⁵ D/cm² and a 0.1 μm tip a neutron yield of 10⁶ D-D n/s for ions accelerated to 100 keV and up to 10⁸ D-T n/s for ions at 100 keV [15] is realizable. This source will be operated in pulse mode with a frequency of 100 Hz, giving the tips enough time between pulses to replenish the deuterium atoms in a vacuum chamber filled with 10⁻⁵ Torr of deuterium. Figure 4b shows a picture of the prototype device that tests the one cm squared field emitter arrays.

The process flow for the field desorption fabrication has been developed. The substrate is a heavily doped conducting Si wafer. Next, a layer of oxide is grown to act as the hard mask for the silicon etch step which creates the tips. A thin layer of tungsten is deposited over the silicon, and another oxide layer is deposited on top of the tungsten, acting as the insulator. Finally, a layer of chrome film is deposited as the gate electrode, and the holes are exposed for the tips. The typical gate hole size is approximately 2 μm, and the radius of the needle tip is on the order of tens to hundreds of nanometers. The separation between each field tip is about 1 to 2 μm, resulting in a packing density of roughly 10⁷ tips per cm squared array. SEM micrographs of the first generation field emitter tips are shown in Figures 5a and 5b. Experimental results from the first generation field emitter tips yield a measurable positive current in the nanoamp range. However, the absolute ion current collected is currently unknown due to secondary electron emission from the target and gate electrode, as well as breakdown problems inside the test chamber.

Figure 4. (a) Concept of the field desorption source, (b) Picture of the field desorption source
3. SUMMARY

Two new types of neutron generators have been discussed. The first utilizes a low power, air cooled, inductively coupled, multicusp plasma source. Current densities and hydrogen ion species have been measured using the prototype plasma ion source. These new sources provide a compact, portable, completely sealed neutron generator that does not require water-cooling. The performance goal for the neutron generator is approximately $10^7$ n/s. Secondly, the field-desorption neutron source for brachytherapy was presented. It utilizes novel microfabricated tungsten coated silicon tips as the source and only requires compact low powered voltage power supplies to operate. The process flow for the microfabrication has been developed and the first sets of devices have been produced and tested. The neutron output goal for the field desorption source is $10^5$ n/s. Further work is underway to continue the fabrication of more field emitter arrays and test them under various pressures and voltages.

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A STEADY STATE MODEL FOR THE HEAT PIPE-ENCAPSULATED NUCLEAR HEAT SOURCE

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ABSTRACT

A major trend in the nuclear industry lately is towards more passive safety systems in power plants. Heat pipes are passive devices that have been used in many industries to remove heat without any electrical input. A design being examined at UC Berkeley is a fast-spectrum heat pipe reactor, called the Heat Pipe-Encapsulated Nuclear Heat Source (HP-ENHS), which utilizes sodium heat pipes to remove heat from a solid molybdenum reactor core fueled by uranium and plutonium nitride. This idea has been examined for space reactor applications before, and with a thermal output of 125MW and the potential to run the reactor with 20 year refueling cycles (from the neutronics data) the HP-ENHS could be suitable for remote sites that are far from the power grid. The focus of this talk will be the ability of the heat pipes to remove heat effectively using a model that predicts the maximum throughput based on capillary limits.

Key Words: Heat pipes, advanced reactors, reactor design, passive systems

1. INTRODUCTION

Heat pipes are used as heat transfer mechanisms in a variety of ways in many fields. A wide variety of working fluids could be used provided materials are chosen that are compatible and the proper temperature range can be achieved. For high temperatures liquid metals offer excellent heat transfer capabilities. Thus liquid metal heat pipes are well suited for high temperature passive heat removal environments. The particular application proposed is as the primary coolant in a fast-spectrum nuclear reactor. The Heat Pipe-Encapsulated Nuclear Heat Source (HP-ENHS) is a fast reactor that uses heat pipes filled with sodium to remove heat through natural circulation. The heat pipes and fuel rods (of equal diameter) are arranged horizontally in the core which consists of a monolithic rectangular block made of molybdenum TZM. The fuel is uranium nitride and plutonium nitride along with other transuranioms. The secondary coolant proposed is a molten salt (2LiF-BeF$_2$) known as FLiBe with very favorable heat transfer and expansion characteristics. This "battery" type nuclear reactor is proposed to operate for approximately 20 years before refueling and is ideal for developing countries, remote sites, and certain industrial applications. Due to its block / rectangular shape at the end of the 20 year cycle the relatively small reactor core could be replaced.

A number of heat pipe steady state operating limits have been documented including the sonic limit, the entrainment limit, the boiling limit, and the capillary limit. The capillary heat pipe operating limit is one of the most important of all of the heat pipe power limits. Without the necessary capillary pressure forces the liquid sodium will not be able to return to the evaporator
and the heat pipe will cease to function. With this in mind a model to predict this capillary limit was constructed for sodium heat pipes at high temperatures. Using this model, different wick materials and dimensions can be compared to see which yields the best results.

2. MODEL DESCRIPTION

2.1. Pressure and Temperature Analysis

This model simplifies the heat pipe by assigning for main points to examine. In this way a one dimensional analysis is possible by examining the heat transfer and pressure drops from one point to the next. A typical heat pipe has a wick structure for condensed liquid flow surrounding a vapor flow section. In the model there are four points of interest; two points for the wick and vapor passage sections in the evaporator section (denoted with subscripts ‘we’ and ‘ve’) and two points for the wick and vapor passage sections in the condenser section (denoted with subscripts ‘wc’ and ‘vc’) as shown in Figure 1.\(^\text{6}\)

The model does not examine the adiabatic section in any detail and calculates the effective length of the pipe, \(L_{\text{eff}}\), by adding half of the condenser and evaporator lengths to the length of the adiabatic section.\(^\text{1}\) The flow in the wick is modeled using Darcy’s law shown below.

\[
\Delta P_{\text{wc}} - \Delta P_{\text{we}} = \frac{\kappa_{\text{eff}} \rho_l A_{\text{wick}}}{\kappa} \frac{\Delta T}{\Delta x}
\]  

Here \(\kappa\) is the permeability of the wick – a number which is often determined experimentally. The mass flow, \(\dot{m}\), is derived straight from the power and the enthalpy of vaporization and \(\rho_l\) represents the density of liquid sodium. The pressure drop for vapor in the center passage comes from the standard equation for flow in tubes.
D_v is the vapor flow diameter in the center passage, V_v is the velocity of the vapor, and \( \rho_v \) is the density of sodium vapor. The friction factor, \( f \), is determined by first finding the flow regime through calculating the Reynolds number, \( Re_D \), in the vapor passage. Once this is determined the friction factor can be obtained through use of one of the following three equations:

\[
\begin{align*}
\text{(3)} & \quad f = \frac{64}{Re_D} \\
\text{(4)} & \quad f = 0.0791Re_D^{-0.25} \\
\text{(5)} & \quad f = 0.046Re_D^{-0.2}
\end{align*}
\]

An assumption is made that in the condenser the liquid and vapor pressure in the wick and vapor passage are equal such that

\[ P_{vc} = P_{wC}. \]  \hspace{1cm} (6)

In typical heat pipes the pressure drop from vapor to liquid in the condenser is much less than the pressure drop in the evaporator. The final equation comes from the Young-Laplace equation describing the pressure drop in the evaporator. Assuming each pore is a tiny cylinder is can be shown that

\[
P_{ve} - P_{we} = \frac{2\sigma}{r_c} \cos \theta
\]  \hspace{1cm} (7)

Here \( r_c \) is the radius of curvature and \( \sigma \) is the surface tension of the fluid. For simplification it is assumed that the contact angle between the fluid and the pore wall, \( \theta \), is zero and the fluid perfectly wets the wall. Equations (1), (2), (6), and (7) can be combined and \( r_c \) can be solved for as shown.

\[
r_c = \frac{2\sigma}{f\frac{\rho_v V_v^2}{2D_v} + \frac{\mu V_{eff}}{k_p A_{wick}}}
\]  \hspace{1cm} (8)

After all of the calculations are complete \( r_c \) is compared with the known pore radius, \( r_p \), to determine if the capillary limit has been reached. If \( r_c < r_p \) the capillary limit has been surpassed since this is not physically possible.

The saturation pressure of sodium as a function of saturation temperature can be reasonably approximated using two constants \( C_0 \) and \( C_1 \). These constants can easily be solved with a thermophysical property table of sodium. With this approximation the saturation temperature is given by

\[
P_{vc} = e^{C_0 - C_1/T_{vc}}
\]  \hspace{1cm} (9)

In the evaporator it is assumed that bubble nucleation occurs within a superheated liquid. This assumption works for evaporator regions closer to the center of the reactor, but since the
The evaporator is fairly long it is assumed to be valid. Now combining (9) with the homogenous nucleation equation and solving for $T_{ve}$ yields the temperature of the vapor in the evaporator.

$$T_{ve} = \frac{c_s + 2\gamma}{c_g - \ln (P_{ve})}$$

(10)

### 2.2. Heat Transfer Analysis

The heat transfer part of the model begins by varying the temperature of the liquid salt secondary coolant. Previous studies performed at UC Berkeley found that with an assumed power throughput of 16.87 kW per heat pipe in a bundle of 3704 (on one side of the reactor), the range of temperatures in the FLiBe will fall between 832 K and 945 K. The average bulk coolant temperature in the bundle was calculated as 888 K with an average heat transfer coefficient from wall to coolant of 4700 W/m$^2$K.

All physical data are interpolated when necessary; including properties of liquid and vapor sodium and the thermal conductivities for the various solid materials compared. The thermal conductivity is determined using a highly idealized model that does not account for the specific geometry of the wick. The conductivities are calculated assuming parallel and series heat flow in the solid and liquid portions of the wick separately.

$$k_{eff} = \frac{k_i k_s}{\phi k_s + (1-\phi) k_l}$$

(11)

$$k_{eff} = \phi k_i + (1-\phi) k_s$$

(12)

These two effective conductivities, $k_{eff}$, are then averaged to obtain an estimate of the effective thermal conductivity. Here $\phi$ is the porosity of the wick and is assumed to be 0.65. The conductivities $k_s$ and $k_l$ represent the conductivities of the solid and liquid respectively. In practice a thermal insulating layer could be employed outside the heat pipe in the condenser to ensure that the heat pipe coolant stays at the constant design temperature.

A simple one dimensional analysis is used to evaluate the conductive and convective heat transfer to and from the heat pipe in the condenser to start the calculation. The coolant temperature is assumed and thermal resistances are calculated based on the heat transfer coefficient to the coolant, the conductivity through the molybdenum wall, and the effective wick conductivity. Basic heat transfer calculations for the thermal resistance of a cylinder are employed to calculate the temperature of the sodium in vapor passage of the condenser. The physical properties are recalculated at this temperature and the process of calculating the various pressure drops and temperatures can proceed.

### 3. PRELIMINARY RESULTS

One of the most difficult parts of this model is obtaining data for certain wick characteristics including the pore radius and the wick permeability. For this reason actual wicks are not used in
this preliminary analysis, instead wicks are assumed to be able to be fabricated from several compatible materials with a porosity, permeability, and pore radius that is typical for wicks in this field. The plots are therefore somewhat hypothetical and more data on specific wick designs will certainly improve this comparison. For these reasons Figure 2 shows several cases with varying values for permeability with molybdenum as the wick material. This demonstrates just how sensitive the capillary limit is to changes in permeability. A porosity of 0.65 is very common for heat pipe wicks so it is used here as well. Previous work has discovered that the pore radius can be manufactured with lasers to as low as 2.5 μm. Since this work is for an advanced reactor design many years away from the manufacturing stage it is assumed that such a pore radius can be obtained. Physical characteristics of the heat pipe were chosen from previous studies at UC Berkeley: the outer heat pipe diameter is 1.56 cm, the inner heat pipe diameter is 1.36 cm, the wick thickness is 0.5 mm, the evaporator length is 75 cm, the adiabatic length is 50 cm and the condenser length is 48.8 cm. The temperature in the secondary coolant, $T_{FLiBe}$, varies from 800 K to 1000 K and the power limit is solved at each of these temperatures to create Figure 2.

![Figure 2. Capillary power limits with varying permeability](image)

4. CONCLUSIONS

The shape of the capillary limit curve is a bit suspect in comparison to some of the other capillary limit curves in the literature. However the power limit predicted for the heat pipe near 1150K seems reasonable. Since this is the region of interest in the HP-ENHS reactor this power limit will be acceptable until further more detailed studies can be done. Further studies will analyze the decay heat transfer from the heat pipes to the secondary fluid and to the Reactor Vessel Air Cooling System (RVACS) through the heat pipe block and the vessel walls. This heat pipe steady state model may be incorporated into these studies.
REFERENCES

8. E. Greenspan, “Solid-Core Heat-Pipe Nuclear Battery Type Reactor”, NERI Grant Number DE-FC07-05ID14706.
Session 2: Student Paper presentations

Design and Analysis for Nuclear Reactor

Chaired by
Prof. T. Fukuzaki
UTNEM

10:00 AM – 12:00 PM
January 8, 2008
NEUTRONIC DESIGN OF THE PB-AHTR

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ABSTRACT

This study (1) investigates the neutronic characteristics of the Pebble Bed Advanced High Temperature Reactor (PB-AHTR); (2) compares the PB-AHTR neutronic performance against those of the alternative design options of He-cooled high temperature reactors using either pebble (PBMR) or prismatic (VHTR) fuel and against a liquid-salt cooled design version of the latter (LS-VHTR); (3) studies the possibility of incinerating TRU in the PB-AHTR feeding TRU from LWRs spent fuel. It is found that the optimal features a graphite-to-heavy metal ratio of ~360 and its reactivity coefficients are all negative. A comparison with the helium-cooled pebble-bed reactor and with a prismatic-fuel reactor that is cooled with either flibe or helium is also presented. It is found that the PB-AHTR offers similar discharge burnup as the other three designs. As compared to the gas-cooled pebble bed, the PB-AHTR uranium loading and energy generated per pebble are ~2.5 times higher. When loading TRU in the pebbles it is found that they can reach a burnup as high as 685 GWD/tHM and the core average reactivity coefficients are all negative. About 70% of the initial load of HM is incinerated in a single pass.

Key Words: pebble, AHTR, VHTR, TRU

1. INTRODUCTION

The PB-AHTR is a Pebble Bed Advanced High Temperature Reactor that is cooled with the liquid salt flibe (LiF-BeF₂) rather than helium. Recent experiments have demonstrated that pebble beds can be formed with a liquid salt coolant and pebbles can be recirculated. This study presents a preliminary neutronic and depletion analysis for the PB-AHTR. Section II describes the computational model and the methodology developed to determine attainable burnup. Section III summarizes the parametric study conducted to search for the optimal uranium loading per pebble, for a given power density and total core power, that maximizes the attainable burnup. Reactivity coefficients are evaluated, too. Comparisons with the helium-cooled Pebble Bed Modular Reactor (PBMR) and with a prismatic-fuel reactor that is cooled with either flibe (Liquid Salt-cooled Very High Temperature Reactor – LS-VHTR) or helium (Very High Temperature Reactor – VHTR) are summarized in Section IV. A preliminary study for incineration of TRU in the PB-AHTR is presented in Section V.

2. MODEL AND METHODOLOGY

This section presents the model applied in the analysis and the methodology developed to determine the attainable burnup.
2.1. Pebble model

A simple unit-cell model is used. It consists of one pebble at the center of a prismatic cell surrounded by liquid salt. The volume fraction occupied by the pebbles in the core - 60% - is conserved. This model is simulated using MCNP5 Version 1.40. Each fuel particle is modeled discretely so as to properly account for the double heterogeneity effect. To save computational time only the fuel kernels are modeled distinctively while the kernel coatings and the graphite matrix are represented as a homogeneous mixture. This approximation was found to have negligible effect on the neutronic properties of the system. The fuel kernels are dispersed inside the pebble in a simple cubic structure; it was found that the neutronic characteristics are insensitive to the type of structure chosen. Reflective boundary conditions are applied to the unit-cell boundary so that our unit-cell model represents an infinite lattice of pebbles. The pebble diameter is 6 cm with a 0.5-cm thick shell. The uranium enrichment is fixed at 10%. Cross sections at the nominal operating temperatures are applied.

Due to the requirement that the pebbles float in the coolant at all operating and accident temperatures, the graphite matrix density is set at 1.60 g/cm$^3$ instead of the typical value of 1.74 g/cm$^3$. The coolant is LiF-BeF$_2$ (66-33 mole %). At the average coolant operating temperature of 655 °C the coolant density is 1.96 g/cm$^3$.

The coolant is initially assumed to be 99.995% enriched in $^7$Li initially, but the composition will change due to $^6$Li neutron capture while in the core$^{2,4}$. However, $^6$Li is also generated by (n,$\alpha$) reaction in the $^9$Be of the flibe. Considering that $^9$Be absorption cross section is very small (~2 mb) while $^6$Li cross section is very large (~200 b), $^6$Li concentration in the core will vary as a function of time until reaching an equilibrium point. It is estimated that about 2 years are required to reach this equilibrium composition of the salt. At this point $^6$Li concentration is less than 1/10 of its initial concentration.

As a result of replacing the salt initial composition with its equilibrium composition, there is a net gain in reactivity of about 5%. In the remainder of the study the equilibrium salt composition is assumed.

In order to determine the achievable burnup for the PB-AHTR a simple, though effective, methodology has been developed$^3$. It is based on a single pebble model and uses MCNP and MOCUP as computational tools so as to properly account for the effect of double heterogeneity. The objective is to estimate the discharge burnup for the equilibrium pebble core without considering how the equilibrium core composition is reached.

### III. DEPLETION ANALYSIS

#### III.1. Attainable burnup

The attainable burnup is determined as a function of the following design parameters: TRISO particles packing factor and fuel kernel diameter for given power density and total core power. For the reference design it is assumed that the total core power is 2,400 MW$_{th}$ and the power density 10.2 MW/m$^3$. The corresponding core dimensions, obtained by optimizing the core height-to-diameter ratio in order to minimize leakage, are 684 cm diameter and 640 cm active length. The neutron leakage probability was determined using a full cylindrical core model with
uniform BOL composition and was found to be 3%. The initial fuel enrichment is kept constant at 10%.

### III.1.1. Packing factor and fuel kernel diameter

Using a fuel kernel diameter of 425 µm, the TRISO particles packing factor is varied between 7.5% and 20% (C/HM between ~600 and ~200). The maximum burnup of 129 GWD/tHM is achieved at around 12.5% (C/HM ~360) and the corresponding residence time is about 670 days. When the fuel kernel diameter is reduced to 225 µm, the maximum burnup, 119 GWD/tHM, is achieved for TRISO particles packing factor of 30% (C/HM ~450) with a 500 days residence time.

Reactivity coefficients estimated for the maximum burnup case with 425 µm fuel kernel diameter, 12.5% packing factor and 10.2 MW/m³ power density, are all negative.

### III.1.2. Power density and total core power

The use of a molten salt as coolant can allow the PB-AHTR to reach much higher power densities than the gas cooled design. Therefore, the attainable burnup is determined for different power densities up to 40 MW/m³. The total core power is kept at 2,400 MWth while core dimensions and corresponding leakage probability are recalculated for each case. When increasing the power density the core becomes smaller and the neutron leakage increases, penalizing the achievable burnup that drops to 113 GWD/tHM when the power density is 40 MW/m³.

A similar effect is verified when decreasing the total core power to 600 MWth as for the PBMR: the leakage probability increases to 6% and the achievable burnup decreases to 113 GWD/tHM.

### IV. Alternative Designs

This Section compares selected characteristics of the main options for high temperature reactor design. The different possible combinations of coolant – flibe or helium, and fuel type – pebble or prismatic, are examined: (1) flibe/pebble – PB-AHTR; (2) helium/pebble – PBMR; (3) flibe/prismatic fuel – LS-VHTR; (4) helium/prismatic fuel – VHTR. The PB-AHTR performance has been described in the previous sections. Here the other designs are described and compared in terms of neutronic properties and attainable burnup.

### IV.1. Gas-cooled pebble bed design

The PBMR model is very similar to that of the PB-AHTR with the only differences that the flibe is replaced by helium and the graphite matrix density is set at the typical value of 1.74 g/cm³ since there are no limits on the pebble mass as in the PB-AHTR. A cylindrical PBMR core is assumed, with a smaller neutron leakage probability than in the actual annular PBMR core design.

### IV.1.1. PBMR attainable burnup

The attainable burnup for the PBMR is obtained using the methodology described in Section II. The total core power is 600 MWth and the power density is 6.6 MW/m³. The resulting core is 498
cm in diameter and 466 cm tall. The attainable burnup and pebble residence time are determined as a function of the TRISO particles packing factor (or C/HM) for fuel kernel diameter of 425 µm. The maximum burnup of 130 GWd/tHM is achieved for C/HM of 906; the corresponding residence time is 446 days. This C/HM ratio is much larger than that optimal for the PB-AHTR because the flibe contributes to the neutron slowing-down and because a reduction in the HM loading increases the fraction of neutrons absorbed in the salt. The limiting factor for burnup in the PBMR becomes the neutron leakage.

IV.2. Prismatic fuel designs

Prismatic fuel designs that are cooled by either liquid salt\textsuperscript{10,11} or helium\textsuperscript{10} are evaluated. The computational model simulates a single fuel element with reflective boundary conditions. The fuel element is a prismatic graphite block (36 cm flat-to-flat, 79.3 cm tall) that is penetrated by 210 fuel holes (active length 78.15 cm), 108 coolant channels and 6 holes for burnable poisons\textsuperscript{10,11} (here these holes are modeled as empty). The fuel holes are filled with columns of fuel compacts that consist of a mix of graphite and fuel TRISO particles. The main difference between the LS-VHTR and the VHTR model, besides the coolant, is the coolant channel diameter that is reduced to 0.953 cm for the LS-VHTR from 1.5875 cm for the VHTR\textsuperscript{10,11}.

IV.2.1. Liquid salt cooled – LS-VHTR

The coolant used for the LS-VHTR is flibe – like that for the PB-AHTR; its composition is assumed to be the equilibrium composition. The LS-VHTR compared to the PB-AHTR features a significant smaller inventory of salt – \(\sim 7\%\) vs. \(\sim 40\%\). As a consequence, the salt fractional absorption is significantly higher in the pebble bed design. The total core power for the LS-VHTR is set at 2,400 MW\textsubscript{th} and the power density at 10.2 MW/m\textsuperscript{3}. The core configuration considered is cylindrical with 265 fuel columns. The estimated neutron leakage is 3\%. The attainable burnup is determined from the depletion analysis of the single element assuming uniform power across the core. The maximum burnup, 131 GWd/tHM, is obtained for 15\% packing factor, corresponding to C/HM of 846 and a fuel residence time of 525 days.

This analysis did not consider the buoyancy issue of the fuel element in the salt for the LS-VHTR. The overall performance could be noticeably degraded if ballast is needed in the fuel elements to hold them in place.

IV.2.2. Gas cooled - VHTR

The inferior thermal properties of helium compared to flibe require the VHTR to use larger coolant channels, smaller power density (6.6 MW/m\textsuperscript{3}) and smaller total power (600 MW\textsubscript{th})\textsuperscript{9,10}. The core configuration is annular; the fuel columns are disposed in three rings for a total of 102 columns. Attainable burnup is determined applying the same technique as for the LS-VHTR. The neutron leakage calculated is 7\%. The maximum burnup obtained – 125 GWd/tHM, is achieved in 567 days for a packing factor of 11\% (C/HM 1033).
IV.3. Comparison

Table I compares the characteristics of the designs that were found to offer the maximum burnup for each of the four systems investigated. It is observed that the discharge burnup of all four reactors examined is similar, with that of the VHTR tending to be somewhat lower. Compared to the PBMR, the PB-AHTR can operate at higher power density, larger total core power, and therefore lower leakage probability. The power generated per pebble is 1.30 MWd for the PB-AHTR vs 0.52 MWd for the PBMR reflecting the fact that the HM per pebble in the PB-AHTR is 2.5 times that in the PBMR. The higher fuel loading in the PB-AHTR is made possible by the relatively large volume fraction of liquid salt that is an effective neutron moderator.

V. TRU incineration

The possibility of designing the PB-AHTR to be fed with TRU fuel instead of enriched uranium is also investigated. The pebble and core design are not modified while the TRISO particles feature small fuel kernels (200 µm diameter) to enhance actinides transmutation. The TRU composition is that of spent fuel from PWRs after 50 GWd/tHM burnup, 5 years cooling and complete removal of uranium, curium and fission products.

The attainable burnup is determined as a function of TRISO packing factor. The maximum is achieved for packing factor of ~11% (C/HM 1993) and is as high as 685 GWd/tHM. This burnup correspond to a destruction of ~70% of the initial load of TRU, 67% reduction of Pu inventory, 96% reduction of fissile Pu and 89% reduction of $^{237}$Np and precursors inventory.

All core average reactivity coefficients remain negative.

The transmutation performance is compared to that of the VHTR fed with the same type of fuel. In this case the maximum burnup is reached for a packing factor of 14% and is set at ~621 GWd/tHM. The limiting factor for the burnup is mainly set by the leakage probability.

The PB-AHTR compared to the VHTR reaches a significant larger destruction rate for $^{237}$Np and precursor inventory, mainly due to the larger destruction rate of $^{241}$Pu.

TABLE I. Maximum attainable burnup designs comparison with 425 µm fuel kernels.

<table>
<thead>
<tr>
<th>Property</th>
<th>PB-AHTR</th>
<th>PBMR</th>
<th>LS-VHTR</th>
<th>VHTR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>Flibe</td>
<td>He</td>
<td>Flibe</td>
<td>He</td>
</tr>
<tr>
<td>Total power [MWth]</td>
<td>2,400</td>
<td>600</td>
<td>2,400</td>
<td>600</td>
</tr>
<tr>
<td>Power density [W/cm³]</td>
<td>10.2</td>
<td>6.6</td>
<td>10.2</td>
<td>6.6</td>
</tr>
<tr>
<td>Leakage probability [%]</td>
<td>3</td>
<td>12</td>
<td>3</td>
<td>7</td>
</tr>
<tr>
<td>Fuel kernel packing factor [%]</td>
<td>12.5</td>
<td>5</td>
<td>15</td>
<td>11</td>
</tr>
<tr>
<td>C/HM</td>
<td>363</td>
<td>960</td>
<td>846</td>
<td>1033</td>
</tr>
<tr>
<td>Specific HM inventory [kg/MWth]</td>
<td>5.23</td>
<td>3.23</td>
<td>4.07</td>
<td>4.59</td>
</tr>
<tr>
<td>Burnup [GWd/tHM]</td>
<td>129</td>
<td>130</td>
<td>131</td>
<td>125</td>
</tr>
<tr>
<td>Fuel residence time [days]</td>
<td>672</td>
<td>418</td>
<td>525</td>
<td>567</td>
</tr>
<tr>
<td>Energy generated per pebble [MWd]</td>
<td>1.30</td>
<td>0.52</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>
VI. Conclusions

The attainable burnup for the PB-AHTR fueled with 10% enriched uranium was evaluated as a function of the following design variables: UC\textsubscript{0.5}O\textsubscript{1.5} fuel kernel diameter, TRISO particles packing fraction, power density and total core power. The maximum burnup of 129 GWd/tHM is achieved when the packing fraction is 12.5% and the kernel diameter is 425 µm. The corresponding fuel residence time is 672 days. Fuel temperature and coolant void reactivity coefficients for the equilibrium core are both negative.

The maximum discharge burnup of the PB-AHTR is very similar to that of the other three design options for high temperature reactors examined. Having a large (~40%) volume fraction, the liquid salt contributes significantly to the neutron moderation in the PB-AHTR. As a result, the peak-burnup uranium loading and the energy generated per pebble of the PB-AHTR are nearly 2.5 times those of a helium-cooled PBMR. Relative to the PBMR, the PB-AHTR features a smaller leakage probability but larger parasitic neutron capture in the coolant. Relative to the LS-VHTR, the PB-AHTR has a significantly larger liquid salt volume fraction and therefore needs a larger specific fuel inventory – 5.23 versus 4.07 kg/MW\textsubscript{th}.

PB-AHTR can be designed to incinerate TRU while maintaining reactivity coefficients negative. The attainable burnup reaches up to 685 GWd/tHM reducing the initial TRU inventory by about 70%.

REFERENCES

MODELING AND TRANSIENT ANALYSIS FOR THE MODULAR PEBBLE-BED ADVANCED HIGH TEMPERATURE

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ABSTRACT

The Pebble Bed - Advanced High Temperature Reactor (PB-AHTR) is a liquid-salt cooled reactor that uses conventional TRISO fuel. Due to the high thermal inertia of the liquid salt coolant, the PB-AHTR allows for a much higher power-density core than traditional gas-cooled high temperature reactors which has a favorable economic impact. In 2007, the baseline design 2400Mwth PB-AHTR was modified to incorporate a modular core with smaller pebbles (between 3 to 4 cm in diameter) flowing inside a large number of separate channels, inside a set of graphite reflector blocks. This configuration has a number of potential advantages over the large, homogenous core that was studied previously. The addition of graphite reflectors allows for a further increase in heavy metal loading, therefore reducing the number of pebbles required and lowering fuel costs and spent fuel volume.

A series of PB-AHTR transients were analyzed using the thermal-hydraulics code RELAP5-3D with the intent of determining an acceptable core power-density range while still meeting design accident criteria. The range under consideration (between 15 to 20 MW/m$^3$) is much higher than the power density of a typical gas-cooled reactor core (6.5 MW/m$^3$). The initiating events considered were Loss of Forced Cooling (LOFC) transients as well as an Anticipated Transient Without Scram (ATWS) which consists of a LOFC transient without scram. Results presented in this paper show that the PB-AHTR response to the LOFC is very promising. Peak temperatures during an ATWS are also within the code temperature range for high temperature alloys such as Alloy 800H. Results from a parametric studies determining the optimal design for the Heat Removal system and subsequent configurations for the core and pebbles are presented.

1. INTRODUCTION

The Pebble Bed Advanced High Temperature Reactor (PB-AHTR) is an innovative reactor design that uses conventional TRISO high temperature fuel, but with a low-pressure liquid salt coolant rather than high-pressure helium [1]. This report presents design and analysis information on a modular pebble-fueled variant of the originally conceived AHTR design.

One of the primary advantages of the AHTR includes the ability to operate at higher power density than helium cooled high temperature reactors while achieving comparable power conversion efficiency, which creates the potential for substantial reductions in the plant capital cost. Likewise, the lower neutron leakage provided by the large PB-AHTR core allows improved fuel utilization, reduced spent fuel generation, and lower fuel cycle costs than those for modular helium reactors.

The earlier PB-AHTR work was on a baseline 2400 MWth PB-AHTR design [2] with a 704°C core outlet temperature, a well understood and qualified fuel (TRISO-based pebble fuel) and available ASME code qualified materials for all high-temperature components (Alloy 800H clad with Hastelloy N), to prevent the need for any materials and fuel development programs. In this work, neutronics simulations [3] demonstrate that negative void reactivity can be achieved by increasing the heavy metal loading of the pebbles and RELAP-3D simulations showed that
the increase of the core outlet temperature was small during a loss of forced cooling (LOFC) transient. Based on these results, a new configuration is being explored with higher power density between 20 to 30 MW/m$^3$, compared to the typical values of 4.8 to 6.5 MW/m$^3$ for the modular helium cooled reactors. This is achieved by using smaller pebbles (between 3 to 4 cm in diameter). This work presents and analyses a modular design with pebbles located in large numbers of separate channels inside a set of graphite reflector blocks, referred to as Pebble Channel Assemblies (PCAs). The goal is to provide a preliminary assessment of the response of the modular PB-AHTR during LOFC transient and to guide in the meantime the design improvements.

2. THE PEBBLE-BED ADVANCED HIGH TEMPERATURE

2.1. Modular PB-AHTR Design

The Advanced High Temperature Reactor is a high-temperature reactor that uses conventional TRISO coated particle fuel together with a liquid fluoride salt coolant. The flow diagram in Fig. 1 provides a simple overview of the initial 2400MWt baseline PB-AHTR design. The primary loop is represented by a line flowing between the core and the Intermediate Heat exchanger (IHX) modules. The annular space between the reactor vessel and the guard vessel is filled with a low-cost buffer salt, sodium fluoroborate, which minimizes primary salt inventory loss if the reactor vessel is faulted.

![Figure 1: Simplified Flow Diagram of the AHTR](image)

The key modifications introduced to develop the 900 MWth, high power density modular design were to switch to smaller pebbles than used in modular helium reactors, and to introduce a large number of separated channels in the core instead of a large cylindrical core configuration. The main advantages of this design include:
2.2. Initiating Event Selection and Regulatory Design Criteria

Probabilistic Risk Analysis (PRA) methods are used to assess overall plant risk by first predicting the frequency of a particular event occurring and then determining the subsequent consequence(s) or end state of this event. The first component of PRA is beyond the scope of this work however assessing the reactor performance and associate consequences of particular events or transients is the main objective of this report. The two transients considered for this paper were selected due to their familiarity in the light water reactor licensing landscape.

During a LOFC transient, a natural circulation flow loop is formed between the hot core and a set of Direct Reactor Auxiliary Cooling System (DRACS) heat exchangers (DHX modules). The DRACS heat exchangers transfer heat by natural circulation flow from the buffer salt to heat exchangers cooled by outside ambient air. Under forced circulation the bypass flow through the DHX is minimized by a fluidic diode. This event is typically initiated by a pump trip due to hardware failure.

In the event of an anticipated transient without scram (ATWS), the reactor is unable to shutdown through conventional means during an anticipated transient. The focus of this event is how the fuel responds during the event without the insertion of reactivity control. This event is typically initiated by a typical reactor trip followed by the failure of the control rod system to function.

In both these events considered, the two key design parameters are the temperature of the fuel and the core outlet temperature. If either of these parameters exceeds design limits, the proposed design may violate governing regulatory design criteria (RDC). Key PB-AHTTR RDC affected during these transients includes the need to control heat generation and remove core heat. Fashioned after 10CFR100 Design Criteria for the MHTGR [4], RDC for the PB-AHTTR are developed such that top-level regulatory criteria established by the licenser are met. Ensuring the ability to control heat generation in the core and the ability to cool down safety-relate systems, structures and components is essential to assessing plant performance during the analyzed transients.

2.3. RELAP5-3D Modeling

The modeling restrains the design to a single hexagonal block core, as is the case for the Pilot Plant. The main parameters of the full design and of the modeling are listed in the Table 1.

- Increase of heavy metal loading in the pebbles due to the moderation provided by the graphite reflectors. This reduces the number of pebbles required and the spent fuel volume.
- Reduction of the coolant void fraction in the core by a factor of 2.
- Creation of locations for insertion of control elements in the solid reflectors
Table 1: Comparison of main parameters for the integral and modular AHTR and for the one-block core modular modeled in RELAP

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Integral design</th>
<th>Modular design</th>
<th>One-block modular</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pebble diameter (cm)</td>
<td>6.0</td>
<td>3.0/4.0</td>
<td>3.0/4.0</td>
</tr>
<tr>
<td>Thermal Power (MWTh)</td>
<td>2400</td>
<td>900</td>
<td>128.6</td>
</tr>
<tr>
<td>Power density in channel (MW/m³)</td>
<td>10.3</td>
<td>60</td>
<td>60</td>
</tr>
<tr>
<td>Active height of core (m)</td>
<td>6.4</td>
<td>2.2</td>
<td>2.2</td>
</tr>
<tr>
<td>Core mass flow rate (kg/s)</td>
<td>9670</td>
<td>3625</td>
<td>518</td>
</tr>
<tr>
<td>Average coolant velocity (m/s)</td>
<td>0.14</td>
<td>0.46</td>
<td>0.46</td>
</tr>
<tr>
<td>Bed packing fraction</td>
<td>60%</td>
<td>60%</td>
<td>60%</td>
</tr>
</tbody>
</table>

In the modular design, the complexity of the core configuration requires the use of a few additional modeling approximations. Currently RELAP cannot handle the hexagonal configuration of the channels in the graphite. The need of having only two connected surfaces led to the choice of an annular design with a succession of pebble channels and graphite areas. Because it contains the same total volume of graphite reflector material and pebble channels as the prototypical system, the model properly reproduces the thermal inertia of these materials in addition to the flow resistance and pressure loss across the core. However, this nodalization generates a distortion in the transient heat transfer between the pebble channels and reflector material.

2.4. Modeling Results

Results from RELAP-5 are presented in Figures 2 and 3 for the LOFC and ATWS transients for the current reference 900 MWth modular PB-AHTR design with 3cm diameter pebbles. Figure 2 shows the modular PB-AHTR thermal response to a LOFC transient initiated at 500 seconds. The most important parameter is the coolant outlet temperature, which determines the peak temperature reached by the primary pumps, IHX’s, and DHX’s during the transient. For the PB-AHTR, the LOFC transient is very mild, with the coolant outlet temperature reaching 735°C at 1100 seconds. As we can see on the previous figure, the LOFC transient is gentle for this reactor. These results can be compared to the GT-MHR response to depressurized LOFC transient with SCRAM during which the fuel reaches temperatures above 1400°C and the vessel temperature increases more than 150°C [5].
Figure 3 shows the transient thermal response of the PB-AHTR to the ATWS transient. As with other liquid cooled reactors, the ATWS transient is among the most severe transients that the reactor can be expected to potentially encounter. In the PB-AHTR, the negative Doppler feedback from the fuel ultimately drives shutdown for an unmitigated ATWS transient. The maximum fuel temperature in the pebble bed is 1175°C. Even though this value is much larger than the peak temperature under LOFC with SCRAM, the TRISO particle failure limit of 1600°C limit is not exceeded. Additionally, the temperature rise of the core outlet is also more severe in the ATWS accident; climbing up to 873°C, which is still below the Alloy 800H temperature limit.

Figure 2: Temperature histories during a LOFC for the new 900 MWth modular reactor

Figure 3: Temperature variations during the ATWS
Parametric studies show that several design parameters affect the maximum coolant outlet temperature in an ATWS transient, including the fuel geometry. These studies show that the peak outlet temperature is lower when using smaller pebbles (3 instead of 4cm) and an annular geometry (which stores less energy than the homogeneous design). However, issues with the pressure drop across the core which required a thicker vessel wall and with fuel particle power fraction lead to modifications of the Pebbles Channels Assemblies (PCAs). By increasing both the flat-to-flat dimensions of the PCAs and the diameter of the channels, lower pressure drops and acceptable fuel particle power fraction can be attained.

3. CONCLUSIONS

This study shows that high powers are attainable with the modular PB-AHTR baseline. The 900Mwth modular design has proved to be viable choice. The LOFC transient is very mild with the peak core outlet temperatures never exceeding 750ºC. The ATWS transient has the potential to be more severe but the temperatures reached are within the code range for high temperature alloys such as Alloy 800H. The impacts of several parameters have been studied to optimize the design both in terms of peak temperatures during transients and pressure drops across the core. Near-term R&D for the PB-AHTR will focus in part on an integral experiment to confirm RELAP5-3D results for the LOFC transients, but also on the modeling of the closed primary loop to study pump trip, reverse flows and intermediate loop trip, which could potentially be more severe.

REFERENCES

Intermediate Heat Exchanger Dynamic Thermal Response Model

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ABSTRACT
The thermal hydraulics of a compact heat exchanger are modeled as an effective porous medium. The approach implements volume averaged momentum and energy equations and significantly reduces computational time. A finite volume approach is used to find the transient temperature variation during flow transients and the resulting temperature distribution is used to calculate an effective stress distribution using more conventional FEA techniques.

Key Words: porous media offset strip fin heat exchanger

1. INTRODUCTION
This paper presents UCB progress in developing a comprehensive thermal and fluid dynamics model for the NGNP intermediate heat exchanger (IHX) and other compact heat exchangers. For improved efficiency electricity generation and for nuclear hydrogen applications, an IHX is required to transfer heat from high temperature and high-pressure primary helium coolant to a power plant or hydrogen production process. An intermediate heat transfer loop is used for the purpose. Under these conditions, plate-type heat exchangers with small flow channels, such as the well known Heatric designs, are a major candidates because they can achieve high power densities with small amounts of material, and can be fabricated using a diffusion bonding process so that the entire heat exchanger has the strength of the base material. However, these types of heat exchangers can be susceptible to very large stresses during thermal transients, for example when the flow of one fluid is interrupted abruptly. UCB has proposed a capillary shell and tube IHX configuration that could have lower susceptibility to thermal shock. For all IHX options accurate analysis of global and local thermal stresses are critical to evaluating the heat exchanger reliability and safety.

In order to estimate the stresses in compact heat exchangers a comprehensive thermal and hydraulic model is needed. The model developed here uses an effective porous media (EPM) approach because the evaluation of the detailed global flow with computational fluid dynamics (CFD) as well as finite element methods (FEM) for the mechanical analysis, at the resolution scale of the flow channels involves prohibitive computational time. The EPM fluid dynamics and heat transfer computational code developed at UCB is called the compact heat exchanger

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thermal and hydraulics (CHEETAH) code. CHEETAH solves for the transient temperature-distribution in the IHX. This temperature distribution can then be imported into a commercial finite element analysis (FEA) code for mechanical stress analysis using the EPM methods developed earlier by UCB for global and local stress analysis [2]. These simulation tools will also allow the designer to optimize the heat exchanger design, to minimize the pressure drop while maximizing the IHX’s thermal effectiveness, as well as to optimize the mechanical performance of the IHX particularly as it relates to creep deformation and transient thermal stresses.

2. FLUID DYNAMICS

The fluid mechanics in the IHX are solved for both fluid phases by breaking up the IHX into over 20 flow zones. Each of these zones has unique physical properties that govern the flow for each of the two phases.

The IHX is built from a diffusion bonded stack of alternating gas and liquid (or gas) plates. Due to the stacked assembly, the flow areas on the liquid and gas plate overlap all over the IHX and create zones in the solid material with complex thermal properties. The zones in the liquid plate are completely separate form the zones in the gas plate, so the fact that the zones overlap has no effect on the fluid mechanics. The overlap intimately affects the heat transfer, however.

2.1 Fluid Dynamics Equations

Beginning with the equation of continuity it is known that mass must be conserved in any representative control volume in the field of flow such that:

\[
\frac{\partial u}{\partial x} + \frac{\partial w}{\partial z} = 0 \quad \text{Equation 1}
\]

Then, it is known that for very small Reynold’s number flows, Darcy’s transport equation yields excellent accuracy with a medium dependant permeability. For laminar flows with a Reynold’s number larger than 1 the permeability coefficient can be adjusted so that it becomes a function of the Reynold’s number. This is done by solving for the Fanning friction factor, \(f_f\) and using this to find the effective permeability, \(k_x\) & \(k_z\).

Darcy’s transport equation for 1D flow in porous media is

\[
u = \nu_{int} \cdot \Phi_x = -\frac{k_x \rho g}{\mu} \frac{dh}{dx}\quad \text{or} \quad u = -\frac{k_x}{\mu} \frac{d\Phi}{dx} \quad \text{Equation 2a}
\]

\[
w = \nu_{int} \cdot \Phi_z = -\frac{k_z \rho g}{\mu} \frac{dh}{dz}\quad \text{or} \quad w = -\frac{k_z}{\mu} \frac{d\Phi}{dz} \quad \text{Equation 2b}
\]

Here, \(\Phi\) is the flow potential [Pa] and \(u\) is the Darcy velocity.
Permeability $k$, and dynamic viscosity $\mu$, vary in space such that the combination of equations 1 & 2 gives the following elliptic differential equation:

$$\frac{1}{\mu} \frac{\partial k_x}{\partial x} \frac{\partial \Phi}{\partial x} + \frac{k_x}{\mu^2} \frac{\partial \mu}{\partial x} \frac{\partial \Phi}{\partial x} - \frac{k_x}{\mu} \frac{\partial^2 \Phi}{\partial x^2} = 0 \quad \text{Equation 3}$$

This equation was solved with appropriate boundary conditions using finite differencing with Taylor-series approximations for each phase and it provided the steady state pressure fields for each fluid. This pressure field readily gave the velocity field for the areas of constant properties. However, while this set of assumptions worked well within the OSF portion of the IHX due to its constant permeability, the assumptions were clearly invalid in the inlet and outlet manifolds where the geometry changes frequently and abruptly.

In order to estimate the velocity field in these areas the more complicated Equation 4, above, had to be solved. However, when it too was approached using the finite difference scheme with Taylor-series approximations, large instabilities resulted in the zones where the permeability varies from node to node such as in the diffuser, reducer and also at the interface between two zones of different permeability such as between the OSF region and the inlet and outlet manifolds. It seemed the instabilities were coming from the first derivatives of permeability with respect to space, $\frac{\partial k_x}{\partial x}$ and $\frac{\partial k_z}{\partial z}$.

3. HEAT TRANSFER

After the fluids module resolves the velocity distribution in the axial (x) and cross-flow (z) directions, the thermal module of the IHX code applies an energy balance using a similar finite volume analysis (FVA) or control volume analysis as discussed previously in the fluid mechanics sections. Here too, the differential equations stemming from the energy equation were initially solved using the FDM with Taylor-series approximations until the CHEETAH code encountered instabilities when modeling the consolidated IHX, which includes many zones with properties that vary spatially and abruptly. In fact, the differential equations below result from the assumption that only temperature varies spatially and that variables such as velocity, density, specific heat, conductivity, convection coefficient, and porosity are all spatially independent. Clearly, this is not the real case, but still the following equations are valid within the independent zones of the IHX where it is reasonable to assume the above mentioned properties as constant and the FDM was used to solve this system of equations before the development of the consolidated CHEETAH code. The equations are also included here because they illustrate the dominant physical phenomena that govern the energy transport in the IHX for each phase.

3.1 Heat Transfer Equations

An important dimensionless number to consider here is the Peclet number, which provides the ratio of the advective energy transport (thermal energy transport due to mass transport) and the conductive thermal energy transport. The Peclet number for the helium and liquid salt flow rate
is over 7000. This clearly means that the heat is mostly due to mass transport while the conductive contribution is negligible.

Neglecting Conduction in the Hot Fluid the energy balance is reduced to:

\[
-u_{fh} \rho_{fh} c_{pfh} \frac{\partial T_{fh}}{\partial x} - w_{fh} \rho_{fh} c_{pfh} \frac{dT_{fh}}{dz} - h_{fh} a'_{fh}(T_{fh} - T_s) = \phi_{fh} \rho_{fh} c_{pfh} \frac{\partial T_{fh}}{\partial t}
\]

Equation 4

Solid (s):

\[
h_{fs} a'_{fs}(T_{fh} - T_s) - h_{fs} a'_{fs}(T_s - T_{fc}) + k_s a_k \frac{\partial^2 T_s}{\partial x^2} + k_s a_k \frac{\partial^2 T_s}{\partial z^2} = \phi_s \rho_s c_{ps} \frac{\partial T_s}{\partial t}
\]

Equation 5

Cold Fluid (fc):

\[
-u_{fc} \rho_{fc} c_{pfc} \frac{\partial T_{fc}}{\partial x} - w_{fc} \rho_{fc} c_{pfc} \frac{\partial T_{fc}}{\partial z} + h_{fc} a'_{fc}(T_s - T_{fc}) = \phi_{fc} \rho_{fc} c_{pfc} \frac{\partial T_{fc}}{\partial t}
\]

Equation 6

4. RESULTS AND CONCLUSIONS

The main concern with the IHX is the mechanical performance during rapid thermal transients. The CHEETAH code has been developed to model the transient temperature distribution during such events. In modeling this scenario, it is assumed that the power plant is operating at steady state at full power when a transient event takes place. Among other things, these scenarios may include an almost instantaneous tripping of one of the coolant pumps. From a thermal stress perspective the most likely, relatively severe thermal transient would involve either the primary or intermediate coolant pump tripping while the other continues to operate. The thermal response for this scenario was recently modeled by CHEETAH, and the results for both a liquid salt and helium pump trip are provided in Figure 1. These are example results. Validation and benchmarking have not yet been performed. The top row of Figure 1 shows the IHX with a temperature distribution that is very near what the steady state or operations temperature distribution in one of the IHX plates. At time equals zero the left column shows the temperature evolution as time advances from a liquid salt (cold fluid) pump trip. The hot temperature front can thus be seen propagating through the IHX. Likewise, the right column shows the IHX experiencing a helium (hot fluid) pump trip and the cold front can be seen moving through the IHX as time advances. These transient temperature distributions can then be used as the basis for a mechanical stress analysis using FEA software such as ANSYS.

Through an iterative process such as this one the IHX designers can quickly optimize the design and update CHEETAH to reflect the new dimensions or shape of the IHX. Eventually, when no obvious problems arise, the temperature distribution can be exported to a mechanical FEA software (likely ANSYS) for detailed thermal/mechanical stress analysis using effective mechanical properties to calculate strains and to then use these strains to back calculate the local stresses as demonstrated in previous work using effective mechanical properties at UCB [1].
Liquid Salt Pump Trip

Helium Pump Trip

0 s

4 s

8 s

12 s

18 s

50 s

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5. NOMENCLATURE

\( a' \) – surface area per unit volume \([m^2/m^3]\) in conv terms or \([m^3/m^3]\) as a porosity in cond terms

\( D_h \) – hydraulic diameter \([m]\)

\( f_f \) – Fanning friction factor

\( h \) – convection coefficient \([W/m^2]\)

\( k_x \) – conductivity in x direction \([m^2]\)

\( k_z \) – conductivity in z direction \([m^2]\)

\( k_{px} \) – permeability in x direction \([m^2]\)

\( k_{pz} \) – permeability in z direction \([m^2]\)

\( \mu \) – dynamic viscosity \([Pa*s]\)

\( \Phi \) – flow potential \([Pa]\)

\( \phi \) – porosity

\( t \) – time \([s]\)

\( T \) – temperature \([K]\)

\( u_D \) – Darcy fluid speed in x direction

\( u_{int} \) – interstitial fluid velocity

\( \bar{u} \) – averaged fluid speed

\( u \) – fluid speed (Darcy) in x direction \([m/s]\)

\( w \) – fluid speed (Darcy) in z direction \([m/s]\)

\( x \) – primary flow direction \([m]\)

\( y \) – direction of plate stacking \([m]\)

\( z \) – cross flow direction \([m]\)

6. REFERENCES


7. ACKNOWLEDGMENTS

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PARTIAL SAFETY ANALYSIS FOR A REDUCED URANIUM ENRICHMENT CORE FOR THE HIGH FLUX ISOTOPE REACTOR

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ABSTRACT

A computational model of the reactor core of the High Flux Isotope Reactor (HFIR) was developed in order to analyze non-destructive accidents caused by transients during reactor operation. Such a model was built based on the available description parameters as provided by the latest version of the nuclear analysis software package called Program for the Analysis of Reactor Transients (PARET). Analysis performed with the model constructed was compared with previous data obtained with other tools in order to benchmark the code. Finally, the model was used to analyze the behavior of the reactor under transients using a different nuclear fuel with lower enrichment of uranium (LEU) than the fuel currently used, which has a high enrichment of uranium (HEU). The study shows that the presence of fertile isotopes in LEU fuel which increases the neutron resonance absorption reduces the impact of transients on the fuel and enhances the negative reactivity feedback, thus making LEU fuel a safe alternative fuel for the reactor core.

Key Words: transient, enrichment, PARET, safety.

1. INTRODUCTION

An initiative to investigate the feasibility of Low-Enriched Uranium fuel as a potential alternative fuel for the High Flux Isotope Reactor originated from the Reduced Enrichment for Research and Test Reactors (RERTR). The proof of principle of this project was performed by R T. Primm [3] in a study where the feasibility of this project was demonstrated. A particular design which employs LEU fuel was identified. Please see reference [3] for the detailed description of this design.
After this preliminary study was performed, the safety of this potential fuel was studied. The desired safety analysis would then provide information on the behavior of this new fuel design under anticipated transients. A nuclear safety computer code called PARET was chosen as the appropriate research tool. Only two anticipated transients were performed: Loss of Flow and Reactivity insertion accidents, but only the first one is reported in this document due to size constraints.

The work presented here was performed during the summer of 2007 as part of the Oak Ridge National Lab – Nuclear Engineering Science Laboratory Synthesis internship program under the mentorship of R. T. Primm III from the Research Reactors Division.

2. HFIR CORE MODEL CONSTRUCTION

2.1. Geometric Properties

The geometric properties for both fuel designs HEU and LEU fuel are equal, as this is the fundamental constraint of the project. The current design of the reactor core of the HFIR is composed of two concentric cylindrical fuel assemblies that consist of fuel plates. The HFIR fuel plates are curved, with an involute shape. In the HFIR core there are 171 fuel plates in the inner fuel region and 369 fuel plates in the outer fuel region (refer to Fig. (1). The fuel thickness within the plates is not constant along the width of the plate.

Figure 1. HFIR Fuel Elements top and lateral cross sectional view

2.2. PARET Nuclear Safety Analysis Software

PARET is a computer code which iteratively solves for the neutronic-hydrodynamic-heat transfer aspect of the reactor under steady state and transient behavior. This code uses the point reactor
kinetics relations for the neutronic aspect of the analysis; furthermore there are multiple laminar and turbulent correlations for the hydrodynamic part and an array of heat transfer correlations as well. The user has the capability to select the appropriate correlation based on the operational parameters and the data available for needed for each correlation.

The computer code allows the user to define multiple core regions with its own specified power generation levels as well as its own reactivity feedback intensity factors for the moderator and the fuel, as described by Obenchain [2]. Each of these core regions are independent from each other in terms of the hydrodynamic and heat transfer aspect, that is, the code solves the thermal hydraulics in 1-D. However, the code couples all regions together for a collective reactivity feedback effect. While the spatial dependence of the power generation is explicitly not taken into account, it is important to specify the appropriate equivalent diameter for the core region being defined, since this will determine the coolant flow velocity and regime in that particular region.

The HFIR core horizontal cross section will be represented by 17 core regions. Axially, the core will be discretized into 20 uneven meshes. Both of these discretization criteria are due to the non-uniformity of the power level along the reactor core also as reported by Primm [1].

2.3 HEU / LEU Model Neutronic – Thermal Parameters

The new fuel design must have equal overall dimensions as the current design, since the reactor vessel is not sought to be changed. However these designs will differ in composition. The current fuel is made with an U$_3$O$_8$ – Al alloy. The new fuel design is composed of a U-Mo alloy.

The PARET input deck is made up of three different sections, the core and fuel geometric data section, the thermal and neutronic parametric data section, and the relative power density description section. The only section common to both fuel design is the geometric data section. Please refer to Appendix A for a listing of the main neutronic and thermal properties and Ref. [3] and Ref. [8]

3. RESULTS

3.1. Loss of Flow Accident

This family of transients refers to the sudden reduction or loss of cooling capacity in the reactor core. The identified transient is documented and detailed in the USAR section 15.3.3. This transient assumes an immediate reduction in primary coolant flow, rather than a gradual loss as it is the case of the pump coast down scenario. This case represents the most severe case of flow reduction, and is therefore chosen to be modeled with PARET.

The figures below show the data output by PARET. The initial conditions for the transient were the specified steady state conditions for the reactor core (see Appendix A). At t=10 sec, the primary mass flux rate was decreased to ¾ of its nominal value, and this causes a sudden temperature rise of the fuel (Fig. 2). It is important to notice from the graph below that the temperature of LEU fuel is higher than the temperature of HEU fuel at steady state, that is, even before the transient.. This is due to the low thermal conductivity of the LEU fuel design
compared to the HEU fuel (see Appendix A). During the transient, both fuel designs rise to a temperature proportional in both cases, and they remain within an acceptable range, thus not compromising safety. The negative values of the mass flow rates indicate downward flow of coolant in the core.

![Primary Mass Flow Rate of Transient](image1)

![Max Fuel Temperatures during Transient](image2)

**Figure 2.** Primary coolant mass flux during transient (left), maximum fuel temperature during transient for both fuel designs (right)

Clearly, the rise in temperature during the transient for both fuel designs causes an insertion of negative reactivity in the core, given that both of these fuel designs have a negative temperature reactivity coefficient. It can be inferred from Fig. 3 that the LEU fuel has a larger temperature reactivity coefficient (more negative) than the HEU design. This can be easily explained due to the fact that LEU fuel has a large amount of U-238, which has a large Doppler reactivity coefficient. In turn, HEU has small amounts of this isotope present, and the Doppler coefficient for U-235 is greater lesser in magnitude, thus reducing the effectiveness of the temperature reactivity feedback mechanism, in case of transients. The power profile shown below also demonstrate this principle, as it can be seen, since the LEU coast down power is lesser for the LEU fuel.

![Reactivity profile of Transient](image3)

![Transient Power Profile](image4)

**Figure 3.** Reactivity insertion profile during transient for both fuel designs (left), profile of reactor core during transient for both fuel designs (right)

3. CONCLUSIONS
Both transients that were analyzed demonstrated that LEU fuel behaves as expected. The reactivity insertion transients prove that under large temperature excursions, there is certain amount of negative reactivity self inserted by the fuel. This is due to the larger Doppler reactivity of the LEU fuel compared to the HEU fuel. The reduction of primary coolant flow transient proved that under a sudden and large temperature change, the insertion of negative reactivity due to temperature feedback both of the fuel and the coolant causes the LEU fueled core level off at a lower power than the HEU fuel, which is due again to the large self insertion of negative reactivity caused by the temperature excursion.

Also, based on the tests performed, it can be observed that the fuel plates of the LEU model reach temperatures higher than those reached by the HEU, this is due to the poor thermal conductivity of the LEU fuel. While this decreases the margin of safety of the reactor core, the highest temperatures reached (~250 °C) are still acceptable; therefore the safety of the fuel is not compromised. Ref. [8] reports that the maximum temperatures under the reactivity insertion transient, are also acceptable for the case of the LEU fuel.

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6. K. A. Smith, *HFIR SAR 15.3.4 Reactivity Transients*, ORNL/HFIR/USAR/2344/Rev. 6
   Approved August, 2006


## APPENDIX A

### HFIR Operating data

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power</td>
<td>85 MW</td>
</tr>
<tr>
<td>Pressure</td>
<td>3.4 MPa</td>
</tr>
<tr>
<td>Reference Coolant Temperature</td>
<td>49 deg C</td>
</tr>
<tr>
<td>Reference Coolant Density</td>
<td>993.0 kg/m$^3$</td>
</tr>
<tr>
<td>Coolant primary mass flux</td>
<td>2.088E4 kg/cm$^2$/s</td>
</tr>
</tbody>
</table>

Table I. HFIR Operating parameters

### HFIR HEU fuel plate geometric data

<table>
<thead>
<tr>
<th>Component</th>
<th>Magnitude [unit]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plate half thickness</td>
<td>6.350E-4 m</td>
</tr>
<tr>
<td>Fuel half thickness</td>
<td>2.460E-4 m</td>
</tr>
<tr>
<td>Fuel width</td>
<td>6.30 E-2 m</td>
</tr>
<tr>
<td>Active fuel length</td>
<td>0.508 m</td>
</tr>
<tr>
<td>Center of fuel – center of coolant channel distance</td>
<td>1.27E-2 m</td>
</tr>
<tr>
<td>Total core area of flow</td>
<td>0.0498 m$^2$</td>
</tr>
</tbody>
</table>

Table II. HEU fuel plate geometric properties

### HFIR Reactor kinetics data: HEU fuel

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>No. of delayed groups $\beta_i$</td>
<td>6</td>
</tr>
<tr>
<td>Effective delayed neutron fraction $\beta_{\text{eff}}$</td>
<td>0.0076</td>
</tr>
<tr>
<td>Mean generation time $\Lambda$</td>
<td>35.00E-6 s</td>
</tr>
<tr>
<td>Fuel Doppler temperature coefficient</td>
<td>-3.711E-6</td>
</tr>
<tr>
<td>Moderator temperature feedback coefficient</td>
<td>-6.65E-3 S/deg C (at T = 50 C deg C)</td>
</tr>
<tr>
<td>Moderator void feedback coefficient (inner fuel element)</td>
<td>-0.24 3$/%$ void</td>
</tr>
<tr>
<td>Moderator void feedback coefficient (outer fuel element)</td>
<td>-0.101 3$/%$ void</td>
</tr>
</tbody>
</table>

Table III. HEU fuel neutronic properties
PWR CONTROL ROD EJECTION
ANALYSIS WITH THE MOC CODE DE CART

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ABSTRACT

During the past several years, a comprehensive high fidelity reactor core modeling capability has been developed called the Numerical Nuclear Reactor (NNR) for detailed analysis of Light Water Reactors. The project was initially sponsored as part of a US-ROK collaborative DOE I-NERI project and the past two years has been sponsored by EPRI to address specific fuel performance issues as part of EPRI BWR Fuel Reliability Program. The NNR achieves high fidelity by integrating whole-core neutron transport solution and ultra-fine-mesh computational fluid dynamics/heat transfer solution. During the past several months there has been interest in taking advantage of the NNR to improve the fidelity for LWR transient analysis. The work described in this paper is a preliminary demonstration of the ability of the whole core neutron transport code, DeCART, to provide an accurate core average power during a control rod ejection. The current state of the art in analysis of this event is to rely upon the assembly averaged power from a whole core nodal neutronics to provide a conservative analysis.

Key Words: DeCART, Control Rod Ejection, Numerical Reactor

1. INTRODUCTION

The Numerical Nuclear Reactor (NNR) is a software system for high-fidelity multi-physics LWR core simulations [1,2]. During the past few years in support of the EPRI BWR Crud Deposition Analysis program, the principal focus of NNR applications has been to steady-state LWR problems, with the most recent emphasis on depletion analysis of BWR fuel assemblies.[3,4] As part of that effort, the neutronic analysis module, DeCART, was extended to include consideration of geometric heterogeneities of BWR fuel assemblies and a depletion analysis capabilities was implemented and tested. A new Computational Fluid Dynamic (CFD)-based boiling heat transfer model was also incorporated in thermal-hydraulic analysis module, STAR-CD, capable of simulating high void fraction boiling regimes encountered in BWRs and the predicting the critical heat flux in PWRs.

During the past several months, there has been interest in taking advantage of the NNR to improve the fidelity for LWR transient analysis. In response, the ability of the whole core neutron transport code, DeCART, to provide an accurate average power during a control rod ejection has recently been investigated. The state-of-the-art in analysis of this event is to rely upon the assembly averaged power from a whole core nodal neutronics simulator to provide a

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conservative analysis. The benefits of the NNR is a full physics simulation of the coupled fields during the rod ejection transient, which could then provide the detailed intrapin power temperature distributions during the transient. The primary goal of the work summarized in this paper is to demonstrate the capability of the NNR neutronic module DeCART to accurately predict the core average power during a rod ejection accident, the reference is PARCS, the core simulator used by the US-NRC to study this kind of accident.

2. MODEL DESCRIPTION

The geometry used to study a rod ejection accident is a “mini-core” consisting of a 3x3 array of typical PWR fuel assemblies. A control rod is partially inserted into the central assembly which is fresh fuel and is adjacent to once and twice burned fuel as shown in Figure 1.
A reflective boundary condition is assumed at the exterior of the 3x3 array. The "mini core" is assumed to be at a hot-zero power state: the fuel and coolant temperatures are at their nominal values but the power generated is set to zero. It is typical of a reactor in standby position (control rods in) and its most sensitive state with respect to reactivity initiated accidents (RIA) (rod ejection).

3. METHODOLOGY

The current methodology to study a rod ejection accident, is to use U.S. NRC core simulation code PARCS. PARCS uses the standard nodal methods with assembly homogenized cross sections. For the purpose of having a consistent comparison with the DeCART solution, the nuclear data used by PARCS (macroscopic cross sections, kinetic parameters) are generated with DeCART.

3.1. Generation of the nuclear data for PARCS

In order to perform transient calculations, PARCS needs two energy group macroscopic cross sections, assembly discontinuity factors, six group precursor's delayed neutron fraction, and two group neutron velocities. DeCART was used as a lattice code (e.g. CASMO) to generate the cross section data for PARCS. For each materially different assembly, fresh unrodded, fresh rodded, once burned and twice burned, a 2D single assembly model is build with DeCART and run to extract the nuclear data needed by PARCS.

3.2. Verification of the nuclear data

In order to check the consistency of the nuclear data generated with DeCART for PARCS, a set of steady state calculations are performed with PARCS and their results is compared to DeCART. The first set of calculations is done for each of the materially different assembly. The infinite multiplication coefficient are recorded. The results are shown in the Table 1.

<table>
<thead>
<tr>
<th>Assembly</th>
<th>k infinity</th>
<th>∆k in pcm</th>
</tr>
</thead>
<tbody>
<tr>
<td>fresh unrodded</td>
<td>1.11891</td>
<td>0</td>
</tr>
<tr>
<td>fresh rodded</td>
<td>0.75454</td>
<td>0</td>
</tr>
<tr>
<td>once unrodded</td>
<td>1.07594</td>
<td>0</td>
</tr>
<tr>
<td>twice unrodded</td>
<td>0.95760</td>
<td>-1</td>
</tr>
</tbody>
</table>

The good agreement of PARCS with DeCART shows that the two group macroscopic cross sections generated with DeCART are correct. The next step is to check the quality of the assembly discontinuity factors by comparing the effective multiplication coefficient of PARCS and DeCART for the 3D "mini core" in steady state. Two cases are considered, one with the
control rod inserted (control rod in) and the other one with the control rod withdrawn (control rod out). Comparing the k-effective of both calculations give access to the reactivity inserted upon withdrawal of the control rod. The results are shown in the Table 2.

The agreement between PARCS and DeCART is not as good as in the previous single assembly calculations. This is expected since the nuclear data, and particularly the assembly discontinuity factors are generated at the single assembly level assuming reflective boundary condition. In the real case, the assembly is next to a materially different assembly and the reflective boundary condition assumption made during the cross section generation stage doesn't hold anymore. This assumption is typical of the current generation of methods. Its effect is not important in this case because the worth of the control rod which is the driver for the transient calculation is close for both PARCS and DeCART calculation. The other important finding of this set of calculations is the control rod worth is over a dollar of reactivity, around 1.15 dollar. The transient resulting of a rod ejection is then a super prompt critical transient.

Table 2. Steady State Comparison of DeCART and PARCS for 3D "mini core"

<table>
<thead>
<tr>
<th>Case</th>
<th>k effective</th>
<th>Δk in pcm</th>
</tr>
</thead>
<tbody>
<tr>
<td>Control Rod In</td>
<td>1.15936</td>
<td>1.15880 49</td>
</tr>
<tr>
<td>Control Rod Out</td>
<td>1.16840</td>
<td>1.16813 23</td>
</tr>
<tr>
<td>Rod Worth in pcm</td>
<td>667</td>
<td>689</td>
</tr>
</tbody>
</table>

Since the nuclear data generated by DeCART have been verified to be accurate, the next step is to look at the transient calculation.

4. TRANSIENT CALCULATION RESULTS

The control rod is ejected from the central assembly and the transient calculation is performed with DeCART and PARCS. Two sets of calculations are done with each code. In the first one (blue curves), the thermo hydraulic feedbacks are turned off. The purpose of this calculation is to evaluate the kinetic parameters generated with DeCART for PARCS. In the second set of calculations (red curves), the thermo hydraulic feedbacks are turned on. Both DeCART and PARCS used a similar simple internal thermo hydraulic solver to simulate the fuel and moderator temperature response during the course of the transient. The only difference between both solvers is that DeCART provides temperature feedback at the fuel rod level whereas PARCS does the same but at the assembly level.

The average power of the mini-core obtained by each code for both calculation is shown in Figure 2. The agreement between PARCS and DeCART during the set of transient calculation without feedbacks is very good showing that the procedure to generate kinetics parameters with DeCART is correct. The agreement between PARCS and DeCART for the set of transient calculations with thermo hydraulic feedback is good at the beginning of the transient and tends to degrade toward the end of the transient due to difference in the way the thermo hydraulic feedback are obtained. There is two main reasons that explain the difference between PARCS
and DeCART. The first reason is different neutronic solver. DeCART handles the exact core geometry without homogenization while in the PARCS code, the geometry within a computational node is homogenized, resulting in significant differences as seen in Section 3. The second reason that accounts for part of the differences between PARCS and DeCART lay in the handling of the thermal hydraulic feedback. The internal solvers used by PARCS and DeCART are based on the same equations however, those equations are solved at the fuel rod level for DeCART and at the assembly level for PARCS. Both reasons detailed above explain the different power behavior of PARCS and DeCART during the rod ejection accident. Nonetheless, the goal of this study, the assessment of the transient capability of the DeCART code has been reached: the agreement of the DeCART results with the US-NRC code PARCS is good. The DeCART approach to perform transient calculations is validated.

![Figure 2. Evolution of the core power during a rod ejection accident](image)

### 3. CONCLUSIONS

A control rod ejection transient has been analyzed with both DeCART and with the U.S. NRC core simulation code PARCS. The cross sections are generated with DeCART in order to provide a consistent comparison of the calculation here with the current state of the art in analysis of the control rod ejection accident. The agreement in terms of core average power between DeCART and PARCS is very good and the small differences can be attributed to the different thermo hydraulic solvers and to the assumptions used to generate nuclear data for PARCS. The benefit of the DeCART approach is to have access to very detailed data like intrapin power density throughout the transient. This kind of data is not available with the current methodology and could be used to reduce some of the conservatism in the fuel design limit for reactivity initiated accident.
REFERENCES


INCOMPRESSIBLE FREE SURFACE FLOW ANALYSIS USING MOVING PARTICLE SEMI-IMPLICIT METHOD

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ABSTRACT

Computational fluid dynamics is widely used to evaluate incompressible flow. However it is difficult to analyze complex motion of the free surface. Particle methods are studied for such flow. It is because motion of the free surface is the same as that of particles. An incompressible condition is to be satisfied to analyze the fluid like water. Using Moving Particle Semi-implicit (MPS) method, we can calculate incompressible free surface flow. In MPS method, particle interaction models are used to approximate the differentials like gradient, divergence and Laplacian. The interactions are localized using a weight function to save the computation time. An implicit pressure correction procedure like the Marker and Cell (MAC) method is employed. However the calculated pressure oscillates, which is not preferable to obtain stable solution. There are two conditions to express incompressibility. The first one is the constant density condition and the other is null density change condition. These conditions are the same in continuum equations, but are not the same in discretized equations. The first condition, which is adopted in the original MPS, is excessively strict so that the pressure oscillates. On the other hand, if the second one is adopted, the pressure oscillation is suppressed but the density gradually changes. This is because the incompressible condition is not perfectly satisfied in the discretized formulation. To stop the gradual density change, we adopted an artificial pressure. With both the second incompressible condition and the artificial pressure, the pressure oscillation is suppressed without gradual density change. We obtained smooth pressure and density distributions in a dam break analysis using the present method.

Key Words: Computational Fluid Dynamics, Particle Methods, Moving Particle Semi-implicit, Free Surface Flow, Incompressibility

1. INTRODUCTION

Particle methods are widely studied to analyze dynamic motion of free-surface flow. Morris et al. [1] and Monaghan[2] used a method for weekly compressible flow in SPH (Smoothed Particle Hydrodynamics). However, strictly incompressible flow analysis is of importance in terms of accuracy and efficiency. MPS (Moving Particle Semi-implicit) method can calculate strictly incompressible flow [3]. MPS adopted a fractional step algorithm like SMAC, in which Poisson’s equation of the pressure is to be solved. A similar algorithm is also applied to other particle methods[4,5,6,7]. However, pressure suffers oscillation. This oscillation is a kind of
numerical oscillation. Therefore it is not preferable. To obtain the smooth pressure fields, pressure is re-calculated in [8,9]. However this needs high computational cost. For the incompressibility, constant density condition is used in [3,4,5], while null density change condition is adopted in [6,7]. Although the pressure oscillation is suppressed with the second condition, the density changes gradually. The mixture of the two is adopted by Zhang et al. [10], Hu and Adams[11] use the both condition. However the problems still remain. Kondo and Koshizuka[12] adopted the null density change condition with artificial force to keep the density constant. This combination enabled us to suppress the pressure oscillation without gradual density change.

In this study, the formulation of [12] is shown. And the pressure is compared with the original MPS[3] using dam break analysis. We also introduce analyses of a fuel pool sloshing by an earthquake and glass melting using MPS method.

2. MOVING PARTICLE SEMI-IMPLICIT METHOD

2.1. Governing Equation of the Incompressible Flow

The governing equations of the incompressible flow are given as:

\[ \nabla \cdot \mathbf{u} = 0 \]  
\[ \frac{D\mathbf{u}}{Dt} = -\frac{1}{\rho} \nabla P + \nu \nabla^2 \mathbf{u} + \mathbf{g}. \]

Equation (1) is the incompressible condition and equation (2) is Navier-Stokes equation. The terms in the right hand side of equation (2) are the pressure, the viscosity and the gravity terms, respectively. Pressure in Navier-Stokes equation is determined to satisfy the incompressible condition.

The right hand side of Navier-Stokes equation (2) is divided into two:

\[ \left( \frac{D\mathbf{u}}{Dt} \right)_{\text{press}} = -\frac{1}{\rho} \nabla P \]
\[ \left( \frac{D\mathbf{u}}{Dt} \right)_{\text{other}} = \nu \nabla^2 \mathbf{u} + \mathbf{g}, \]

which are pressure term and the other terms. Poisson’s equation of the pressure:

\[ \nabla \cdot \frac{1}{\rho} \nabla P = \nabla \cdot \left( \frac{D\mathbf{u}}{Dt} \right)_{\text{other}} = \frac{D}{Dt} (\nabla \cdot \mathbf{u})_{\text{other}} = \frac{D}{Dt} \left( -\frac{1}{\rho} \frac{D\rho}{Dt} \right)_{\text{other}} \]

is obtained using the mass conservation equation. The pressure of the incompressible flow is calculated with this equation.

2.2. Partial Differential Models

To calculate the motion of the free surface flow with particles, governing equations are discretized to the momentum equations of the particles. The differential operators are replaced by particle interaction models. The interaction of the particles is limited to a finite domain using the following weight function:
\[ w_{ij} = \begin{cases} \frac{r_e}{|r_{ij}|} - 1, & 0 \leq |r_{ij}| < r_e, \\ 0, & r_e \leq |r_{ij}|, \end{cases} \quad (6) \]

where \( |r_{ij}| \) is the distance between two particles and \( r_e \) is the effective radius (Fig. 1). To calculate a weighted average, particle number density is calculated as,

\[ n_i = \sum_j w_{ij}. \quad (7) \]

Gradient and Laplacian are represented by the following particle interaction models:

\[ \langle \nabla \phi \rangle_i = \frac{d}{n_i} \sum_j \frac{\phi_j - \phi_i}{|r_{ij}|^d} w_{ij}, \quad (8) \]

and

\[ \langle \nabla^2 \phi \rangle_i = \frac{2d}{\lambda n_i} \sum_j [(\phi_j - \phi_i) w_{ij}] \quad (9) \]

respectively, where \( d \) is the number of spatial dimensions. In equation (4), parameter \( \lambda \) is evaluated as

\[ \lambda = \frac{\sum_i |r_{ij}|^2 w_{ij}}{\sum_j w_{ij}}. \quad (10) \]

Since the particle number densities are kept constant in incompressible fluid dynamics, we can use the constant value denoted by \( n_0 \) instead of \( n_i \) in equations (8) and (9). For \( n_0 \), we use the particle number density at the initial state.

**2.3. Semi-implicit Algorithm**

To satisfy the incompressible condition, a semi-implicit algorithm is used in the MPS method (Fig. 2). This algorithm has two steps. At the beginning of each step, we search neighbor particles using background cells.

The first is the explicit step in which the forces except for pressure are calculated as:
\[
\left( \frac{du}{dt} \right)_i^{\text{other}} = \nu \frac{2d}{\lambda n^0} \sum_{j \neq i} (u_j - u_i) w_{ij} + g_i. \tag{11}
\]

These are viscosity and gravity in this study. In this step, the temporal acceleration and movement are also calculated as:

\[
\mathbf{u}^*_i = \mathbf{u}^t_i + \Delta t \left( \frac{D \mathbf{u}}{Dt} \right)_i^{\text{other}} \tag{12}
\]

and

\[
\mathbf{x}^*_i = \mathbf{x}^t_i + \Delta t \mathbf{u}^{(1/2)}_i, \tag{13}
\]

where \( \Delta t \) is a time step.

In the second step the pressure term is calculated implicitly. After the explicit step, the right hand side of Poisson’s equation (9) can be approximated as:

\[
\frac{D}{Dt} \left( - \frac{1}{\rho} \frac{D \rho}{Dt} \right)_i^{\text{other}} = - \frac{1}{\Delta t^2} \frac{n^* - n^k}{n^0}. \tag{14}
\]

The particle number density has two meanings. One is the normalization factor of the weighted average and the other is the value proportional to the fluid density. Therefore the particle number densities are used to evaluate the density change.

The left hand side of Poisson’s equation (5) is discritized using the Laplacian model including the coefficient \( 1/\rho_{ij} \) as:

\[
\nabla \cdot \frac{1}{\rho} \nabla p = \frac{2d}{\lambda n^0} \sum_{j \neq i} \left[ \frac{1}{\rho_{ij}} (P_j - P_i) w_{ij} \right], \tag{15}
\]

where

\[
\rho_{ij} = \frac{2 \rho_i \rho_j}{\rho_i + \rho_j}. \tag{16}
\]

Solving discrete version of Poisson’s equation :

\[
\frac{2d}{\lambda n^0} \sum_{j \neq i} \left[ \frac{1}{\rho_{ij}} (P_j - P_i) w_{ij} \right] = - \frac{1}{\Delta t^2} \frac{n^* - n^k}{n^0}, \tag{17}
\]

we obtain the pressures of particles. Since the matrix equation (8) is symmetric, it is solved by the conjugate gradient method.

---

**Fig. 2 Semi-implicit algorithm of MPS**

The matrix equation is indefinite if equation (8) is given to all particles. Therefore we give zero pressure to the surface particles, which are judged using the particle number densities.

With this pressure calculation density of the fluid is not kept constant. It is because the incompressibility is the null density change condition. The density should be constant in incompressible fluid dynamics. Therefore, the particle number density is required to be constant. The constant is denoted by $n^0$. We introduced the artificial pressure as

$$p_{i,\text{art}} = -\frac{\alpha \rho}{\Delta t^2} \sum_j \frac{w_{ij}}{n^0_j} \left( n^k_j - n^0_j \right) ,$$

(18)

where $\alpha$ is a coefficient. We use $\alpha = 0.1$ in this study. The whole pressure is the sum of the pressure obtained from Poisson’s equation and the artificial pressure.

$$p_{i,\text{whole}} = p_{i,\text{press}} + p_{i,\text{art}}$$

(19)

Using the whole pressure, the acceleration due to the pressure term is calculated as

$$\left( \frac{du}{dt} \right)^{\text{press}} = -\frac{1}{\rho} \frac{dn^0}{dt} \sum_{i=1}^{N} \left[ p_{i,\text{whole}} + p_{i,\text{press}} \right] \frac{\rho^0}{|\mathbf{r}_{ij}|^2} \mathbf{r}_{ij} w_{ij} .$$

(20)

This model enables us to keep the momentum of all particles constant. This kind of conservation is important in the numerical calculation. If the distribution of the particles is uniform, we can say that the gradient model (equation (20)) is equivalent to the equation (3).

In the implicit step, the particles are accelerated and moved as

$$\mathbf{u}^{k+1}_i = \mathbf{u}_i^k + \Delta t \left( \frac{\mathbf{D} \mathbf{u}}{\Delta t} \right)^{\text{press}} ,$$

(22)

and

$$\mathbf{x}^{k+2}_i = \mathbf{x}_i^k + \Delta t \mathbf{u}_i^{k+1}$$

$$= \mathbf{x}_i^k + \Delta t^2 \left( \frac{\mathbf{D} \mathbf{u}}{\Delta t} \right)^{\text{press}} .$$

(23)

### 2.4. Stabilization

We eliminate the negative pressure after solving Poisson’s equation of the pressure. Since the particle number densities of the particles near the surface are small, the pressures of the particles become negative, which causes instability. We set zero pressure, if the calculated pressure is negative. In the case that the pressure is seldom negative away from the surface, this treatment does not affect the calculation result so much.
3. CALCULATION EXAMPLES

3.1. Dam Break Analysis
Using both the original MPS method[3] and the proposed MPS method[12], a dam break is analyzed. Pressures at wall particles A and B are compared in Fig. 3. While the pressure oscillates in the original model, the oscillation is suppressed in the proposed method.

![Fig. 3 Dam break analysis](image)

![Fig. 4 Pressure at particle A](image)

![Fig. 5 Pressure at Particle B](image)

3.2. Fuel Pool Analysis
A fuel pool sloshing is analyzed under the earthquake wave of Nigata-Chuetsu Oki. The size of the pool is 10 x 10 x 10[m]. Instead of moving the pool itself, we give inertia force to each particle based on the acceleration data, which is opened to the public by Japan Meteorological Agency. The motion of the fluid surface and the pressure is obtained(Fig. 6). The wall of the pool is not rendered so that the pressure can be seen.
3.3. Glass Melter Analysis

Simulations are performed for a glass melter, which is used to vitrify high level-liquid waste (HLLW) from nuclear fuel reprocessing in nuclear fuel cycle. Pouring the molten glass from nozzle into a canister is analyzed. The glass flow simulation is important because the glass melter conditions can be optimized. In the glass flow, large deformation and breakup of fluid appeared, where the Eulerian approach using grid or mesh is difficult to analyze. On the other hand, the Lagrangian approach like MPS is relatively easier to deal with them.

As far as the molten glass properties, e.g. density, viscosity and surface tension are concerned. They depend upon the temperature and noble metal content which is included in the HLLW. In this study, temperature dependency is considered in calculating the glass flow. To take surface tension into consideration, we adopted the model of Kondo et al. [13]. The glass flow which is calculated by particles is visualized by not only creating isosurface but also considering luminescence from high temperature glass (Fig. 7).

4. CONCLUSION

Using MPS method, incompressible free surface flow can be calculated. The pressure oscillation in the original MPS[3] is suppressed without gradual density change. The formulation adopting the combination of the null density change condition and the artificial pressure is shown. In the dam break analysis pressure is compared to the original method. The pressure is less fractured with the present method. We also show the results of the fuel pool sloshing by the earthquake and the glass melter analyses, which indicate the possibility that MPS method provides us the visual observation to understand these phenomena.
REFERENCES

MAXIMIZING POWER OF HYDRIDE FUELLED PRESSURIZED WATER REACTOR CORES

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ABSTRACT

This paper presents the thermal hydraulic methodology and results for grid-supported pressurized water reactor cores, loaded with UZrH$_{1.6}$ fuel arranged in a square lattice, and with fuel rod diameter and pitch optimized for maximum burnup. A parametric study over core length and core pressure drop shows a maximum attainable power of 4,330 MWth for a core radius of 1.68 m, with flow velocity as the most limiting constraint. This result is comparable to the power of 4,210 MWth for a core geometry designed for peak power, at standard core pressure drop and length.

Key Words: Hydride fuel, PWR, thermal hydraulic analysis, VIPRE.

1. INTRODUCTION

Recently, there has been an interest in the use of hydride fuels in light water reactors. This is due in part to the potential of hydride fuels to increase power density, discharge burnup, minor actinide consumption and capability to utilize thorium. This paper builds upon work that has been carried out at UC Berkeley under Prof. Greenspan$^1$ and at MIT under Prof. Todreas, looking at the neutronic and thermal hydraulic design to maximize burnup and power output$^{2,3}$.

Neutronic analysis done at UC Berkeley by Ganda et al. found that a pitch, P, to diameter, D, (P/D) ratio of approximately 1.2 enables maximum burnup (BU) of UZrH$_{1.6}$ fuel having 12.5% enriched uranium$^1$.

Shuffler et al. investigated maximum attainable power for a fixed fuel length (L) of 4.6 meters and for a limited range of pressure drop. Their studies show that pressure drop across the core is the dominating constraint for small D and small P/D values. Shuffler obtained a core power of 4,210 MWth, at the standard pressure drop of 0.20 MPa and standard core length of 4.6 m, and a core geometry of P/D = 1.49 and D = 6.5mm. The core power at the geometry that maximizes burnup, of P/D = 1.2 and D = 9mm, and standard core length and pressure drop is 3,390 MWth$^2$.

In the present work, we perform a parametric study, which considers the core length as a design variable, and which considers a wider range of core pressure drops in the search for the maximum attainable core power. Our goal is to identify the peak attainable power from a pressurized water reactor (PWR) core loaded with UZrH$_{1.6}$ fuel that is designed for peak burnup (BU).
This paper presents a methodology for PWR steady state thermal hydraulic analysis, which reduces computational time, allowing for parametric studies over larger design parameter spaces. Using this methodology we evaluate the maximum attainable power of tight lattice small height cores, for square lattice rod geometry.

2. PROPOSED ANALYSIS

We propose a parametric study of attainable core power as a function of core height and pressure drop, for tight lattice small height UZrH_{1.6} cores. The fixed parameters are P/D of 1.2 and rod diameter of 9.0 mm (this geometry maximizes BU), coolant inlet temperature of 294°C, and core radius of 1.68 m.

The design constraints are the Minimum Departure from Nucleate Boiling Ratio (MDNBR) at steady state operation, which must remain above 2.17; peak fuel temperature T\textsubscript{CL}, which must remain below 750°C; and coolant flow velocity, which must remain below 8 m/s.

The MDNBR constraint is determined by Shuffler et al., based on South Texas reference PWR UO\textsubscript{2} core, and operating conditions of UZrH\textsubscript{1.6} fuel\textsuperscript{2,3}. The peak centerline temperature is limited by the stability of the hydride fuel; and the average velocity in the hot channel is limited by fuel rod vibration and wear. A fixed velocity limits is a conservative approach, considering that we will investigate shorter cores, but keep a constant number of grid spacers.

3. METHODOLOGY

3.1. Single Channel Thermal Hydraulic Analysis

We used the VIPRE-01 mod02 thermal hydraulic code\textsuperscript{4} to solve the steady state single channel conservation equations\textsuperscript{5} for mass, momentum, and energy, with a chopped cosine axial power distribution of 1.55 peaking factor.

3.2. Cross-Flow Correction

In order to perform a parametric study of core power over a wide parameter space, we used a simplified computational model. Rather than a full-core thermal hydraulic analysis, we perform a single channel analysis of the hot channel to obtain the average permissible linear heat rate in the hot channel – q\textsuperscript{hot}. In terms of the constraints listed in Section 2, the channel with the highest average linear heat rate is most limiting, and it sets the maximum attainable power of the core. We estimate the average linear heat rate in the core to be q\textsuperscript{hot}/1.65, where 1.65 is the assumed core radial peaking factor.

The single channel analysis approach is more conservative than a full-core thermal hydraulic analysis because the latter accounts for the coolant cross flow and heat exchange among channels. The radial heat flow from the hot channel increases the permissible total heat flux from the hot channel for a given MDNBR constraint. As a result of this phenomenon, Shuffler observes a radial enthalpy peaking factor of 1.71 for a hydride core with radial linear heat rate peaking factor of 1.65.
To account for cross-flow and turbulent mixing among channels, we use an iterative single channel analysis methodology. This methodology is outlined in Figure 1, and detailed below:

**Figure 1. Thermal hydraulic analysis methodology.** We propose an iterative single channel thermal hydraulic analysis, which uses radial peaking factors for enthalpy rise and linear heat rate of $F_{q'} = 1.65$, and $F_{\Delta h} = 1.71$, respectively. We compare the results of the proposed methodology with those from a lumped channel model of $1/8^{th}$ of the core.

Step 1: The MDNBR and pressure drop are set and the maximum permissible linear heat rate for the hot channel is determined using VIPRE single channel analysis. Step 2: The $q'_\text{hot}$ from Step 1 is used to calculate the linear heat rate of the average channel: $q'_\text{av} = q'_\text{hot} F_{q'} = q'_\text{hot} \frac{1.65}{1.65}$. Step 3: The $q'_\text{av}$ from Step 2 and the pressure drop are set, and the mass flow rate of the average channel is determined using VIPRE single channel analysis. Step 4: The $m_{\text{av}}$ from Step 3 is used to calculate the effective mass flow of the hot channel: $m_{\text{hot}} = m_{\text{av}} F_{q'} F_{\Delta h} = m_{\text{av}} \frac{1.65}{1.71}$. Step 5: The $m_{\text{hot}}$ from Step 4 and the MDNBR are set, and the final linear heat rate of the hot channel is calculated by VIPRE single channel analysis. This $q'_\text{hot}$ is the maximum achievable hot channel linear heat rate for a given MDNBR and core pressure drop ($\Delta P_{\text{actual}}$).

To validate our methodology, we compare our results with the results obtained by Shuffler at al² using a $1/8^{th}$ core lumped channel thermal hydraulic analysis for the following geometry: $P/D = 1.49$, $D = 6.5$, $L_{\text{core}} = 4.6m$. The results of this comparison are shown in Figure 1.

Because the single channel model does not allow for cross flow, the differences in the axial velocity profile may lead to an overestimation of the friction losses and a reduced mass flow at a given pressure drop, as the results in Figure 1 indicate. As a result, our model is to a small degree more conservative in predicting linear heat rate and fuel peak temperature and less conservative in predicting average velocity in the hot channel. For a given MDNBR and pressure drop, these two effects balance each other out.
3.3. Core Geometry

The total rod length scales with the fuel length. The fuel starts 8.89 cm from the inlet of the core; this distance is required for instrumentation and that it does not scale with the fuel length. We scale with fuel length the space at the top of the fuel rod, which is allotted for the accumulation of fission products.

\[ L_{\text{core}} = 8.89 \text{cm} + L_{\text{fuel}} (1.057) \]  

(2)

The number and types of grid spacers are the same as in the reference PWR: two non-mixing vane grids and eight mixing vane grids. The spacing between the grid spacers is scaled with the length of the core:

\[ Grid_i = Grid_{i,\text{ref}} \frac{l_{\text{core}}}{l_{\text{core,ref}}} = = Grid_{i,\text{ref}} \frac{L_{\text{core}}}{4.6 \text{m}} \]  

(3)

The grids of the 4.6 m reference core are located at axial heights of 0.000 m, 0.148 m, 0.529 m, 1.051 m, 1.573 m, 2.356 m, 2.878 m, 3.139 m, 3.661 m, and 4.183 m.

Because core pressure drop is one of the design parameters, the power consumption by the coolant pumps needs to be accounted for. This power scales with the core pressure drop:

\[ Q_{\text{pump}} (hp) = \frac{\dot{V} \Delta P \times 2.41}{3960 \times SG \times \eta_{\text{pump}} \times \eta_{\text{el}}} = \frac{\dot{V} \Delta P \times 2.41}{3960 \times 1 \times 0.6 \times 0.33} \]  

(4)

Where \( \dot{V} \) is the volumetric flow rate in gpm, \( \Delta P \) is the core pressure drop in psi, SG is the specific gravity and it is conservatively assumed to be 1, \( \eta_{\text{pump}} \) is the pump efficiency, and \( \eta_{\text{el}} \) is the thermal to electric energy conversion efficiency.

The core power is calculated from the linear heat rate of the hot channel as follows:

\[ Q_{\text{core,net}} = Q_{\text{core}} - Q_{\text{pump}} = q'_{\text{av}} \times L_{\text{fuel}} \times N_{\text{fuel}} = \frac{q'_{\text{hot}}}{F_{q'}} \times L_{\text{fuel}} \times N_{\text{fuel}} - Q_{\text{pump}} \]  

(5)

4. RESULTS

Steady state thermal hydraulic analysis shows that the maximum permissible core power is 4330 MWth obtained at \( \Delta P = 0.36 \text{ MPa, } L_{\text{core}} = 4.6 \text{ m, } P/D = 1.2 \text{ and } D = 9 \text{ mm.} \)

If the flow velocity constraint could be disregarded, the temperature limited core power increases with pressure drop up to a value of 7720 MWth at a pressure drop of 1.38 MPa, as shown in Figure 2.
Figure 2. Maximum Permissible Core Power over a range of Lcore = 1.4 to 4.6 m. P/D = 1.2, D = 9 mm, MDNBR = 2.18, T_{CL} < 750 °C.

5. DISCUSSION

The maximum attainable power at a given pressure drop represents the maximum power as a function of core length. At fixed core pressure drop, shorter cores lead to lower friction losses, and allow for higher mass flow. Therefore shorter cores correspond to higher MDNBR-limited linear heat rate. Because total core power is a product of the linear heat rate and the fuel length, core power will present a peak, when plotted as a function of core length. These results are shown in Figure 3. It is to be noted that at higher pressure drops this peak shifts to longer core heights.

Figure 3. Core Power. MDNBR = 2.18, P/D = 1.2, D = 9 mm.
For reasons explained above, flow velocity has a strong direct dependence on core pressure drop, and a weaker inverse dependence on core length. For MDNBR-limited designs, we expect peak fuel temperature to increase for shorter core lengths and higher pressure drops, because these designs correspond to higher linear heat rates. Figure 4 shows the coolant flow velocity and the fuel temperature as functions of core length and pressure drop.

![Figure 4. Peak Fuel Centerline Temperature and Flow. MDNBR = 2.18, P/D = 1.2 and D = 9 mm. The fuel temperature constraint is 750 oC. The flow velocity constraint is 8 m/s.](image)

6. CONCLUSIONS

We have shown that for a hydride core pressurized water reactor optimized for peak burnup (P/D = 1.2, D = 9 mm), the maximum permissible core power is 4330 MWth, at a core length of 4.6 meters, and core pressure drop of 0.36 MPa. This design is limited by flow velocity of 8 m/s, and steady state MDNBR of 2.18, and it meets the UZrH$_{1.6}$ fuel peak temperature limit of 750°C. We have also shown that if the flow velocity constraint is relaxed, core powers of up to 7,720 MWth can be obtained.

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OPTIONS FOR HYDRIDE FUELS REPROCESSING

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ABSTRACT

This paper proposes a possible method for reprocessing of hydride fuels with several options and limitations which must be overcome by future research efforts. Based on the well-established Zirflex process with certain modifications, the fuel composition can be dissolved resulting in the feed solution for conventional solvent extraction in TBP. Appropriate conditions for the dissolution and extraction processes are specified.

Key Words: Reprocessing, Hydride fuels, Modified Zirflex, TBP extraction

1. INTRODUCTION

The zirconium-hydride based compositions are proposed as nuclear fuel and are currently under investigation within the context of a collaborative effort by the UC Berkeley Nuclear Engineering Department among other participants [1]. Such fuels have proven to possess a valuable combination of efficiency and safety factors that make them useful for various reactor applications. Much attention is now being devoted to the possibilities for substitution of traditional oxide fuels with hydride compositions in new design LWR-type reactors; moreover other potential applications are mentioned in the literature – such as, for example, in accelerator-driven systems for actinide transmutation. The research is supported by decades of the successful operation of uranium-zirconium hydride as a fuel or zirconium hydride as a moderator in research reactors.

Another expected advantage of the of the hydride-fuel based reactor systems is the possibility of recycling the trans-uranium component, particularly Pu, which allows for the control of its stockpile and the ability to approach the “closed” nuclear fuel cycle. This is consistent with the overall global trend aimed at the more effective fissile material utilization, such as the INPRO concept supported by IAEA.

The proposed hydride fuels have a very specific composition and characteristics that are distinctly different from the traditionally-used oxide fuels. The treatment of these fuels at the reprocessing stage will require the introduction of unconventional methods, and the overall feasibility and effectiveness of hydride fuel reprocessing and recycling is far from clear. Even though industry has been utilizing nuclear fuel reprocessing for over 60 years now, there is still little or no information on the industrial reprocessing of fuels other than ceramic-based ones.
2. HYDRIDE FUEL COMPOSITIONS

Several types of hydride fuels have been researched, as described in [1]. All of them are members of a family of a composite hydride fuel that can be denoted as U-(ThnPuZr)Hx, and the primary composition concerned is uranium-zirconium hydride (U-ZrHx). For power LWRs the weight percent of the uranium ranges from about 30 wt % to about 60 wt %, and is preferably about 45 wt % uranium [2]. As a variation of this design, the uranium-thorium hydride (U-ThHx) composition is considered. For LWRs the uranium content of this fuel is supposed to be approximately 25 wt % [1]. Additionally, similar hydride fuel systems were proposed for the incineration of plutonium and trans-uranium elements. These compositions can be denoted as PuH2-ZrHx and PuH2-ThHx for plutonium fuel and TRUH2-ZrHx and TRUH2-ThHx for trans-uranium elements utilization [1].

For LWR application, zircaloy is proposed as the fuel cladding material. Several methods were proposed to prevent Zr clad hydration during the campaign, with priority given to the liquid metal gap filling. The LM is a low melting temperature (~120°C) alloy of lead, tin and bismuth at 33 weight % each [3].

3. HYDRIDE FUELS DISSOLUTION

Based on the analysis of the available dissolution methods for highly inert hydride fuel compositions, it is possible to select a process that will satisfy all the parameters required for industrial application.

In order to recover target components from the zirconium or thorium-based hydride fuels by solvent extraction, the fuel and zircaloy cladding must be dissolved and suitable feed solution prepared. Chemically, one is limited in the choice of reagents to dissolve hydride fuels. The practical aspects of the problem include rate of dissolution, resistant equipment materials, solubility limitations, minimum explosion hazards, products extractability, and a low unit volume of waste.

Hydride fuel compositions are very inert materials and in aqueous media could be dissolved at significant rates only by fluoride species. The most obvious reagent for this is hydrofluoric acid, however, both have obvious difficulties associated with their high corrosion activity, and the problem of extensive hydrogen emissions during the dissolution in HF make it inapplicable. An alternative approach is to use fluoride salts to produce fluoride ions in situ allowing one to realize dissolution at the industrial scale. The most developed and widely practiced method is the Zirflex process and its various modifications. It was used at the industrial scale both for selective zircaloy cladding and zirconium-uranium fuel dissolution [4].

Modified Zirflex can be considered as the most suitable dissolution process for hydride fuels. The dissolution is then realized in the NH4F-NH4NO3-H2O2 system, and allows the complete and rapid dissolution of zircaloy cladding and hydride fuel with soluble uranium and TRU nitrates as the ultimate product. In this process, ammonium fluoride NH4F serves as the source of fluoride ions, which gives relatively low corrosion rates and zirconium is obtained as (NH4)2ZrF6.
Ammonium nitrate NH$_4$NO$_3$ serves as an oxidant and as a hydrogen depressant. It is important to consider that the corrosion of structural materials increases when free fluoride and nitrate ions are simultaneously present in the solution. Therefore, the process of adding ammonium nitrate could be omitted, producing a less corrosive environment, if some precipitants (for example tin from the zircaloy) and hydrogen in the off-gas stream are permitted. In case of hydride fuels, ammonium nitrate is not objectionable, because the final solution has to be stabilized in nitric acid resulting in the feed for the PUREX extraction.

Hydrogen peroxide H$_2$O$_2$ is used as an oxidant and to reduce corrosion, but it would also oxidize any U (IV), preventing uranium precipitation. If this additional oxidant is not present, tetravalent uranium would precipitate before solution stabilization in HNO$_3$. Thus, the continuous addition of H$_2$O$_2$ is required, although a specific rate is not critical as long as a threshold oxidant concentration is maintained. According to [4], the excess of hydrogen peroxide can only dilute the solution and contribute oxygen to the off-gas, without any negative effects.

The dissolution product should be stabilized in the nitric acid solution which is a requirement for the PUREX solvent extraction feed. However, the resulting system would still contain free fluoride ions that are extremely corrosive in the presence of the nitrate ion. This will require additional processes in order to decrease the fluoride activity in the solution before stabilization.

For the obtained solution, it would be most appropriate to bind the excessive fluoride in the complex with aluminum, a process that is often described in the literature [5, 6]. According to this method, the hydride fuel solution could be adjusted with aluminum nitrate Al(NO$_3$)$_3$ in order to produce the complex with residual fluoride and make the solution non-corrosive to stainless steel. Nitrate ion added with aluminum provides additional stabilization for uranium and other components. Fluoride complexes of uranium, plutonium, thorium, as well as most of the fission products are generally less stable than those of aluminum. Zirconium, however, is an exception; therefore in the presence of aluminum, all available fluoride will be distributed between the aluminum and zirconium complexes:

\[
\begin{align*}
\text{Main process:} & \quad \text{Zr}^{4+} \rightarrow \text{ZrF}^{3+} \rightarrow \text{ZrF}_2^{2+} \rightarrow \text{ZrF}_3^+ \\
\text{Stability constant [5]:} & \quad 10^{9.8} \quad 10^{17} \quad 10^{29} \\
\text{Adjustment:} & \quad \text{Al}^{3+} \rightarrow \text{AlF}^{2+} \rightarrow \text{AlF}_2^+ \rightarrow \text{AlF}_3 \rightarrow \text{AlF}_4^- \\
\text{Stability constant [7]:} & \quad 10^7 \quad 10^{12} \quad 10^{16} \quad 10^{20}
\end{align*}
\]

The constraints posed by the solubility limits in the Zr – Al system should be observed. These factors can be considered at the dissolution stage or controlled by the dilution with stabilizer – nitric acid and water. The final H$^+$ – Al – Zr – F composition should be chosen for optimal solution stability, salting strength and corrosion protection in the acid environment. The final stabilization and feed adjustment should be done by adding nitric acid to the desired concentrations, typically around 3 mol/l for TBP extraction.

The gaseous reaction products from the hydride fuels dissolution by the Modified Zirflex process consist of ammonia NH$_3$, hydrogen and oxygen [8]. Ammonia from the off-gas can be removed by water or dilute nitric acid scrubbing. The remaining mixture of hydrogen and oxygen can
potentially reach flammable concentrations, but their volume is expected to be small and can be
diluted with air. Therefore, dissolution can be considered safe.

The proposed dissolution process based on Modified Zirflex followed by stabilization in the
aluminum nitrate and nitric acid is potentially applicable to the hydride fuels of any composition.
It provides feed solution suitable for conventional extraction and permits common construction
materials for both dissolution and solvent extraction equipment.

However, the primary disadvantage of this method is fluoride transfer to further reprocessing
stages where they can become a source of intensive corrosion. It can affect all subsequent
processes – extraction, off-gas treatment, waste disposal, and especially evaporation and volume
reduction. This problem is inherent to all methods that involve fluoride compounds.

Another concern of the dissolution process in the case of zirconium hydride fuels is the
introduction of large amounts of inert material into the feed flow. This results in the formation
of high saline solutions that require special adjustments of the solvent extraction process and
affect the overall efficiency of the separation. It also considerably increases the volume of the
liquid radioactive wastes compared with the conventional chop-and-leach process where zircaloy
cladding is removed in the form of compact solid wastes. An overview of the potential
decladding methods for hydride fuels did not, however, reveal an effective approach. This
difficulty is inherent in the proposed compositions because of the similarity in chemical and
physical properties between clad and fuel materials. This requires one to consider options for
hydride fuel pre-treatment that will allow for the separation of zirconium cladding into a
different flow.

4. FEED ADJUSTMENT AND THE FIRST EXTRACTION CYCLE

The dissolution process described above provides a feed solution that is suitable for solvent
extraction in TBP. Therefore, overall extraction partitioning could be realized within the PUREX
process system (or its modifications) similar to conventional oxide fuels. However, several
important differences should be considered and adjustments to the head extraction cycle applied.

The most important factor for the hydride fuel feed solutions is the significant zirconium content
as a result of the combined dissolution of fuel matrix and zirconium cladding. These solutions
will have high saline content and low concentrations of the target components (uranium,
plutonium, thorium). Additionally, zirconium is one of the few elements whose behavior in the
solvent extraction is extremely complicated. Even in oxide fuel extraction, when zirconium is
present as a fission product and the content is low, its separation to the waste flow is problematic
and sometimes unsuccessful, requiring additional refinement at the later stages.

In nitric acid solutions, zirconium forms a number of complex compounds [9]:

\[
\text{Zr(OH)}_4 \rightarrow \text{Zr(OH)}_2\text{NO}_3 \rightarrow \text{Zr(OH)}_2(\text{NO}_3)_2 \rightarrow \text{ZrOH(NO)}_3 \rightarrow \\
\rightarrow \text{Zr(NO}_3)_4 \rightarrow \text{Zr(NO}_3)_6^{2-}
\]

* The term “inert material” refers to any reagents included or introduced into the process and eventually included into the
radioactive wastes flow. In the case of hydride fuels, it is substantial amounts of aluminum and zirconium.
These complexes equilibrate very rapidly and the dominant form depends on the zirconium and nitrate ion concentrations. Zirconium is highly extractable in tetranitrate form, thus contaminating fissile material flow according to the reaction:

$$\text{Zr}^{4+} + 4\text{NO}_3^- + 2\text{TBP} \rightarrow [\text{Zr(NO}_3)_4 \cdot 2\text{TBP}]_{\text{org}}$$

In conventional oxide fuel reprocessing, the contribution from this reaction is significant and the zirconium decontamination coefficient can vary significantly depending on the conditions.

Remarkably, for the feeds obtained as a result of zircaloy dissolution in the Zirflex process, significantly higher decontamination factors from zirconium were achieved. In fact, as it was noted before, the introduction of the fluoride ion at the dissolution stage forms extremely stable complexes with zirconium reducing the amount of Zr in the extractable form. This effect was labeled as the “Zirflex Tonic Effect” [10] and was observed even when aluminum was added for corrosion control. This is an important basis for extraction processing of hydride fuel solutions.

Additionally, since zirconium extractability depends directly on the nitrate ion concentration, the weak acid scrubbing should further increase its decontamination factor. The acid lowest concentration in the scrubbing flow is limited by the need to prevent plutonium hydrolysis and could be as low as 0.5 mol/l.

Another serious problem associated with zirconium content in the feed solution is its tendency to initiate the “third phase” formation. At certain conditions, zirconium forms insoluble complexes with tributyl phosphate and its decomposition products – dibuthyl and monobutil phosphates. This results in the formation of emulsions on the aqueous-organic phase interface with a decrease in decontamination factors and extraction effectiveness.

In zirconium rich solutions, the third phase is mainly formed by the formation of $\text{Zr(NO}_3)_4 \cdot 2\text{TBP} \cdot \text{HNO}_3$ and was observed at high concentrations of nitric acid when TBP is saturated with HNO$_3$. This effect is also sensitive to the nature of TBP organic diluent and is considerably diminished by the use of special additions of aromatic diluents as well in pulse extractors [11]. Therefore, for third phase control from the hydride fuels solution feed, lower concentrations of nitric acid and TBP in head extraction cascade could be recommended.

A very important factor for the extraction processing of hydride fuel solutions is the significant fluoride content in the feed after the Zirflex dissolution. Generally, the presence of anions other than NO$_3^-$ negatively affects the extraction of target elements. The fluoride in the solution can inhibit the extraction of plutonium and thorium due to the formation of fluoride complexes of these elements. However, in the hydride fuel feed solution fluoride ion is combined in much more stable complexes with zirconium and aluminum, and this effect is not expected to occur. The presence of aluminum in the feed solution has a positive effect on the extraction since it serves as a salting-out agent for uranium which increases decontamination factors. In addition, as mentioned before, aluminum forms stable complexes with free fluoride ions reducing corrosion and its effect on extractability.
5. CONCLUSIONS

The method most suitable for aqueous hydride fuels reprocessing is based on the fuel dissolution in the \( \text{NH}_4\text{F} - \text{NH}_4\text{NO}_3 - \text{H}_2\text{O}_2 \) system which allows for the complete dissolution of the inert fuel matrix. Subsequent stabilization in \( \text{NH}_3\text{O}_3 - \text{Al(NO}_3)_3 \) results in a stable feed for conventional TBP extraction. The processes, are not limited by the reagents corrosiveness, therefore ordinary stainless steels are applicable at the dissolution and extraction stages. The off-gas consists of \( \text{NH}_3 \) and small amounts of \( \text{H}_2 \) and \( \text{O}_2 \), which with appropriate off-gas treatment guarantees the safety of the process. Considering the compositions of the feed solutions, some adjustments at the first extraction cascade are required. However the general separation can be achieved within the traditional PUREX process scheme.

For further research in hydride fuels reprocessing, continued experimental research is required. The priority within this context must be to clarify the optimal conditions of the dissolution process, solubility constraints of the zirconium-aluminum feed solution, and the parameters of the first extraction cycle.

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DEVELOPMENT OF PORTABLE X-BAND LINAC X-RAY SOURCE FOR NON-DESTRUCTIVE TESTING

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ABSTRACT
We are developing a portable X-ray non-destructive testing (NDT) system using 9.4GHz X-band linear accelerator (linac) with 250kW magnetron. This X-ray source energy is less than 950 keV for Japanese regulation. The 950 keV X-band linac X-ray generator is under development for corrosion wastage of tubes and condition based maintenance of impeller pumps. We have chosen the 250kW magnetron so that the RF heat loss is remarkably reduced. This design yields compactness and portable. This system consists of the X-band magnetron, modulator, thermionic 20kV electron gun, X-band linac, and metal target of X-ray generation. We aim that X-ray spot size is less than 1mm. We designed the linac structure of the $\pi$ mode at low energy parts and the $\pi/2$ mode at high energy parts, and analyzed the electromagnetic field by SUPERFISH and the electron beam dynamics by GPT. We finished cold test, RF aging and electron beam emission. Now, we are performing electron accelerating test and X-ray generation. Details of the total system and demonstration experiment are presented.

Key Words: X-band linac X-ray source, Non-destructive test, Condition based maintenance

1. INTRODUCTION
Generally, non-destructive testing (NDT) is carried out by using ultrasonic, radiation, neutron, eddy-current, and X-ray. Nondestructive testing by using X-ray is particularly the most useful technique to inspect with higher resolution. We can especially evaluate corroded pipes of petrochemical complex, nuclear- and thermal-power plants by the high energy X-ray NDT system. General X-ray NDT system is based on the S-band linear accelerator (linac), but it is rather large and the electron beam spot size and the spatial resolution are about 3 mm. On the other hand, we design a mobile “Suit-case-sized” X-band 950keV linac for NDT applications. The NDT system with X-band linac uses 1 MW magnetron, where the RF heat loss is serious.
We can inspect inner imperfections of many remarkable industrial products if we use this portable NDT system.

2. THE DETAIL OF X-RAY SOURCE SYSTEM

2.1. Schematic View of X-ray Non-destructive Testing System

We have developed a compact X-ray NDT system using 9.4 GHz X-band linac driven by a tunable 250 kW magnetron. Fig. 1 is the schematic illustration of our compact NDT system. Instead of a large 1.5 MW magnetron conventionally used in 1 MeV X-band linacs, we have chosen a low power, marine radar magnetron to significantly reduce system cost and enhance portability. Our system consists of the magnetron, microwave components, pulse modulator, thermionic 20 kV electron gun, X-band linac, target for X-ray generation, and control system. The total system size consists of two boxes of 50 cm x 30 cm x 30 cm for power supply, 50 cm x 30 cm x 30 cm for magnetron, linac, cooling system and metal target of X-ray generation. The operation temperature of the system is 35 degree.

2.2. X-ray Non-destructive Testing Applications

The 950 keV X-band linac X-ray generator is under development for corrosion wastage of tubes and condition based maintenance of impeller pumps. We can carry out on-site evaluation of...
industrial products at various plants and petrochemical complexes with this NDT system. Considering X-ray energy and spot size, this system can evaluate corrosion and wastage of tube. As you can see Table 1, the corrosion and wastage size are more than 1mm.

Table I. Objects of evaluation

<table>
<thead>
<tr>
<th>Flaw</th>
<th>Corrosion</th>
<th>Wastage</th>
<th>Small crack</th>
</tr>
</thead>
<tbody>
<tr>
<td>Size (mm)</td>
<td>&gt; 1</td>
<td>&gt; 1</td>
<td>&lt; 1</td>
</tr>
</tbody>
</table>

Fig.2 is examples of NDT system use. We will use this NDT system for corrosion check at chemical complex tower, Kashima, Japan. As condition based maintenance, this X-ray system is able to check impeller pump in rotating condition.

3. ACCELERATOR DESIGN

To optimize the accelerating structure, we used some commercial software. SUPERFISH-code is the calculation of 2D electromagnetic field to design the cavities. MICROWAVE STUDIO is a 3D calculation code of electromagnetic field for the waveguide coupler. D-GUN and General Particle Tracer (GPT) are for the design of electron gun and electron beam dynamics, respectively. This section explains accelerating structure design.

3.1. Electromagnetic Field Calculation by SUPERFISH

The accelerating tube is the on-axis coupled standing-wave cavity. At low energy parts of the accelerator tube, the $\pi$-mode is adopted and at high-energy parts, the APS type $\pi/2$-mode is used for efficient electron beam acceleration. The $\pi$-mode is standing wave and the $\pi/2$-mode is the synthesis of the forward and backward traveling wave. The resonant frequency is 9.4 GHz. By these components, electron beam is accelerated from 20 keV to 950 keV. Fig.3 shows electric field of designed accelerating tube structure.
3.2. Beam Dynamics Simulation by GPT

We could calculate beam energy shift by GPT, and confirmed this accelerator structure is effective. The energy density and histogram are shown in Fig.4.

![Beam dynamics simulation by GPT](image)

Figure 4. Beam dynamics simulation by GPT (left: energy density, right: histogram)

4. EXPERIMENT RESULT

We completed the measurement of resonant frequency and axial electromagnetic fields using the bead-pull method. We have also checked the design parameters. Now, the whole system included beam diagnostic section is under construction at Nuclear Professional School, the University of Tokyo. Fig.6 is the photograph of X-band linac tube.

![Accelerator tube](image)

Figure 6. Photo of accelerator tube
4.1. RF Aging and electron emission experiment

We completed RF aging and electron emission test without RF. On RF aging, E-field could be put in 250 kW with 50 pps. The left side of Fig.7 is typical waveform of oscilloscope. You can see over-coupling reflection. After RF aging, in the condition of no-RF, electron emitted. In the result of electron emission test, we could get 20 keV electron beam with 400 mA (See the right side of Fig.7).

![Figure 7. Waveforms of oscilloscope (left: RF aging, right: electron emission without RF)](image)

4.2. Electron Accelerating Test

Now, we are constructing an experimental setup for electron beam acceleration and X-ray generation experiment. Fig.8 shows the schematic configuration of the experimental setup for beam current measurement. We observed the accelerated beam current to be 80 mA.

![Figure 8. Beam current measurement (left: experimental setup, right: waveform of oscilloscope)](image)

5. FUTURE WORKS

Now, we experiment electron energy spectrum test and will experiment spot size and emittance. 950 keV X-ray source can be only used for thin tubes of ~20 mm thickness. Therefore, we start to design 3.95 MeV X-band linac X-ray NDT systems. The range of application is broader by more high energy. We think that this X-ray NDT system will be used for corrosion wastage and cracking in thicker tubes at nuclear plants and larger impeller pumps. This system consist of X-band linac, thermionic cathode electron gun, magnetron and waveguide components etc.

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3.95 MeV X-band linac structure is adopted side-coupled structure for higher electric field works out. This structure is more effective acceleration than 950 keV Linac with alternating periodic structure. We adopt 1.3 MW magnetron for RF source. This accelerator system is ~30cm long. The beam current is ~150 mA, and X-ray dose rate is 10 Gy@1m/500 pps.

6. CONCLUSIONS

We have designed and developed a compact, non-destructive X-ray evaluation system using the 9.4 GHz X-band linac with a 250 kW magnetron. By using X-band linac and low power magnetron, the accelerator length becomes shorter, and the RF heat loss is remarkably reduced. Therefore, the cooling system becomes smaller, and the total system size becomes more compact and portable. With this NDT system, we can carry out on-site evaluation of industrial products at various plants and petrochemical complexes.

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A STUDY OF REFLECTION AND CONNECTION MATERIALS USED FOR TRANSMITTING AND CONDENSING SCINTILLATION LIGHT BY MEANS OF OPTICAL FIBER

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ABSTRACT

The commonly used γ-ray measurement method employs a technique to transmit scintillation light to a photo-multiplier tube (PMT) via an optical fiber. However, it is said that the light transmission efficiency in this technique reduces significantly depending on the transmission properties, particularly on the critical angle of the optical fiber. The light entering the optical fiber from all directions, such as scintillation light, is not suitable for the optical fiber transmission. As most of the incident light escapes from the fiber, the use of optical fibers results in a significant loss of light. We performed experiments and Monte Carlo simulations to examine the extent of decrease in the light transmission efficiency of a discrete γ-ray detector, which uses optical fibers for the connection of PMT with scintillator, relative to an integrated γ-ray detector, which connects PMT directly with scintillator. We also examined the effect of the reflection and connection materials on the light transmission efficiency. The results show that in improving the light transmission efficiency, it is significant to apply specular or diffuse reflectors appropriately to the side and bottom surfaces of the scintillator, and also to provide space (air layer) between the output surface of the scintillator and the input surface of the optical fiber without using grease.

Key Words: Reflector, Plastic Optical Fiber, Scintillator, Grease, Monte Carlo Simulation

1. BACKGROUND OF THE STUDY

The light entering the optical fiber from all directions, such as scintillation light, is not suitable for the optical fiber transmission. As most of the incident light escapes from the fiber, the use of optical fibers results in a significant loss of light. The light loss is estimated to be almost 90%.1,2) If the light loss is lessened, the weakness of optical fiber transmission can be eliminated. Thus we performed experiments and simulations to evaluate the effects of using an optical fiber to transmit light from the scintillator to the PMT in detecting an annihilation γ-ray using an LSO scintillator.

2. EXPERIMENT 1: EVALUATION OF REFLECTION MATERIAL PROPERTIES
2.1. Materials And Methods

Figure 1 shows the configuration of the experimental setup. $^{22}\text{Na}$, LSO (7 mm dia. x 20 mm thick) and HAMAMATSU_H3378-51 were used as positron source, scintillator and PMT, respectively. In Case 1, the scintillator was optically connected to the PMT with grease. In Case 2, a light guide was placed between the scintillator and the PMT. The light guide was a PUNEUM polymer light guide, with 0.63 as NA (the number of apertures). Energy spectrum was measured for Cases 1 and Case 2, and the relative intensity of light was obtained by comparing the peak values of the photopeak (511 keV) of the annihilation $\gamma$-ray.

![Figure 1. Configurations of detectors.](image1)

Figure 2 shows the energy spectrum for Cases 1 and Case 2, in which a Teflon tape (3M) and TiO$_2$ paint (Saint Gobain) were used as diffuse reflectors and AL tape (3M) as a specular reflector. To briefly explain about specular reflection and diffuse reflection (the detailed explanation provided later), the former is the reflection of light at the same angle as the incident angle, while the latter is the reflection of light at all angles regardless of the incident angle. Diffuse reflectors are usually white objects.

![Figure 2. Spectra of normal type and fiber type at 511 keV.](image2)
2.2. Results

We obtained the results that indicated the effect of the properties of diffuse or specular reflectors on the transmission rate of the light. Table I shows the results of optical fiber transmission rates. The optical output was normalized taking Case 1 with Teflon tape as 100%. The highest transmission rate was obtained for Teflon in Case 1 and for TiO₂ in Case 2. In terms of the light loss due to the optical fiber, Case 2 shows different results, i.e., 11.0% for Teflon, 16.8% for TiO₂ and 13.5% for Al. Both Teflon and TiO₂ are diffuse reflectors. The opposite results in Case 1 and Case 2 regarding the transmission efficiency suggest the effects of other properties than reflectivity, because the opposite results cannot be attributed only to the difference in reflectivity between the two diffuse reflectors. For the specular reflective Al tape, the light output was small as a whole.

<table>
<thead>
<tr>
<th>Reflective Surface</th>
<th>Case 1</th>
<th>Case 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Teflon Tape [Lambert]</td>
<td>100</td>
<td>11.0</td>
</tr>
<tr>
<td>TiO₂ Paint [Lambert]</td>
<td>82.6</td>
<td>13.9</td>
</tr>
<tr>
<td>Al Tape [Mirror]</td>
<td>43.2</td>
<td>5.8</td>
</tr>
</tbody>
</table>

Taking notice on this refractive index of Teflon and TiO₂ as another important property than reflectivity, we performed calculations taking the effect of this into account.

2.3. Discussion

A simulation was performed to discuss the experimental results, using a mathematical model for the experiment. In the simulation, the experimental results were reproduced and the characteristics of a hypothetical experimental system were studied.

2.3.1. Summary of the simulation

We simulated the phenomena that occur in the process from γ-ray detection by the scintillator to the transmission of scintillation light to the PMT. There are two types of reflection, diffuse and specular. In specular reflection, light is reflected at the same angle as the incident angle against the reflecting surface. In diffuse reflection, light is reflected in all directions regardless of the incident angle. Lambert’s cosine law was used for diffuse reflection.

When the diffuse reflector was used, probably Fresnel reflection would have occurred due to the difference in refraction index between the reflector and the scintillator. Therefore, a thin layer
(about 0.1 mm thick) to represent the refractive index of the reflector was placed in front of the Lambert surface and to cause Fresnel reflection on the surface of the layer.

In the simulation study, the direction of emission at the light emitting point, the direction of reflection and the loss by reflection were determined by the Monte Carlo method, using Excel macro programming.

### 2.3.2. Simulation conditions

The simulation used a scintillator size of 7 mm dia. x 20 mm thick, and a reflector reflectivity of 98 %, 92.5 % and 80% for Teflon, TiO₂ and Al models, respectively. A refractive index of 1.35, 2.5 and 2.5 was used for Teflon, TiO₂ and Al models, respectively. A refractive index of 1.82, 1.40, 1.48 and 1.35 was used for the scintillator, the grease, the light guide core and the light guide clad, respectively.

However, in a calculation with the thin layer (about 0.1 mm thick) representing the refractive index of the reflector, light may have been reflected several times within the layer, and consequently may have entered the Lambert surface with a certain probability, causing a decrease in apparent reflectivity. In order to perform a general evaluation of reflection characteristics, a correction was made to calculate multiple reflections within the layer as one reflection. Statistical results were obtained from a calculation of 100,000 photons.

### 2.3.3. Calculation results

Table II shows the calculation results. The simulation results are generally in agreement with the experimental results, i.e. the optical fiber transmission rate is the highest for the Teflon model in Case 1 and for the TiO₂ in Case 2. This suggests the effect of the refractive index of the reflectors. However, numerically this hardly agrees with the experimental results. This disagreement is probably due to the fact that the reflector reflectivity used in the calculation was not a measured value. Also, it is necessary to examine how close the reflectivity in the air is to that in the crystal. The relative values for Case 2 to those for Case 1 are larger than the experimental results, the reason for which is to be discussed later.

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
<td>54.8(100)</td>
<td>35.4(64.6)</td>
<td>24.4(44.5)</td>
</tr>
<tr>
<td>Case 2</td>
<td>8.1(14.8)</td>
<td>10.5(19.1)</td>
<td>5.4(9.9)</td>
</tr>
</tbody>
</table>

( ): Number normalized taking Case 1 with Teflon®tape as 100%
[ ]: Reflection property

Table II. Simulation result of optical fiber transmission ratios for different types of reflectors
3. EXPERIMENT 2: EFFECT OF OPTICAL CONNECTION MATERIALS

3.1. Materials and method

The light output was measured for Case 1 and Case 2, with/without an optical connection material. The basic measurement configuration was the same as in Fig. 1. The light intensity was compared between the two cases with/without grease on the optical connection surface. Table IV shows the measurement results. The values in parentheses are the results for the case with grease (Case 1). Comparison between the cases with/without grease shows that the light output decreased for all reflectors in Case 1, while in Case 2 without grease, it increased from 11.0 to 14.7 for Teflon (by 34 %) and from 13.9 to 16.3 for TiO₂ (by 17%). This shows that when a diffuse reflector is used, the results of light output were better without grease than with grease.

<table>
<thead>
<tr>
<th>Table IV. Measurement result of optical fiber transmission ratios for optical connection without grease</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
</tr>
<tr>
<td>Case 2</td>
</tr>
</tbody>
</table>

( ): Optical connection with grease [ ]: Reflection property Numbers are normalized taking Case 1 with Teflon®tape as 100%.

3.2. Results

A simulation was performed in the same manner with Table III, which showed similar trends to the experimental results.

<table>
<thead>
<tr>
<th>Table V. Simulation result of optical fiber transmission ratios for optical connection with/without grease</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
</tr>
<tr>
<td>Case 2</td>
</tr>
</tbody>
</table>

( ): Optical connection with grease [ ]: Reflection property
shows the simulation results. A calculation for Case 2 with Teflon and TiO$_2$ as reflectors verified the experimental result that the transmission rate was better without grease. The improvement rate was 71.6% for the Teflon model and 9.5% for the TiO$_2$ model, which were much higher than the improvement rate of 1.9% for the Al model.

3.3. Discussion

The improvement of the transmission rate by using a diffuse reflector can be explained as follows. For scintillators with a high refractive index such as LSO scintillators, total reflection based on Fresnel formula occurs on the output surface of the scintillator, restricting the incident angle of the optical fiber. Using grease as an optical connection material gives a wider incident angles for the light to reach the light-receiving surface of the optical fiber than having an air layer. Therefore, the light entering the optical fiber increases when the grease is used. However, the light entering the optical fiber at a large angle is not necessarily transmitted through the optical fiber with the total reflection conditions retained.

If an air layer is used as an optical connection, the light that has reached the output surface of the scintillator at a large angle repeats reflecting inside the scintillator, without entering the optical fiber. Then it reaches the output surface again at a different incident angle by the diffuse reflector. Thus the diffuse reflection in the scintillator changes the light directions, contributing to reducing the light loss in the light guide, which consequently improved the transmission rate. Whereas in a specular reflector, this does not occur because the light retains the same incident angle even if it repeats reflection.

4. CONCLUSIONS

We evaluated the effect of reflector properties by experiment and simulation. It was found that reflection properties of the reflector, as well as the relative refractive index of the scintillator and the reflector, affect the light output. The simulation also showed that no use of grease for optical connection is effective in optical fiber transmission if diffuse reflectors are used.

ACKNOWLEDGMENT

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REFERENCES

Transient Hydride Fuel Behavior in LWRs

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ABSTRACT

A Parametric study is performed for the transient response of a uranium zirconium-hydride (U_{0.31}ZrH_{1.6}) fuel element to boundary conditions typical to a light water reactor (LWR) accident. The diffusion of hydrogen within the fuel is treated as a function of the changing temperature profile. Temperature and hydrogen concentration dependence of thermal properties of the fuel is also considered. The set of coupled equations describing the transient conduction of heat and diffusion of hydrogen are solved simultaneously with a Crank-Nicolson time-discretization scheme for heat diffusion and an explicit scheme for hydrogen diffusion.

Key Words: Hydride Fuel, Transient Heat Conduction, Transient Hydrogen Diffusion, Stress

1. INTRODUCTION

Hydride nuclear fuels (uranium-zirconium hydride) have been successfully utilized in the past in many research and test reactors as well as space programs. The added presence of hydrogen in the fuel provides neutron moderation within the fuel in addition to the traditional moderator. This allows displacement of moderator with fuel, effectively increasing power density. Hydride fuels also enjoy a higher thermal conductivity than oxide fuels and possess thermally-induced hydrogen up-scattering that accompanies Doppler feedback. Uranium-zirconium hydride fuel consists of metallic α-U phase dispersed in a δ-ZrH_{1.6} matrix. Maximum heavy metal loading inside the fuel is limited to 45 vol% uranium which corresponds to the fuel composition of U_{0.31}ZrH_{1.6}. During operation of the reactor, the temperature gradient across the fuel drives the hydrogen to the cooler regions due to the large heat of transport of hydrogen in δ-ZrH_{1.6} phase (T_Q = 640K) [1]. The thermal conductivity, hydrogen diffusivity, and volumetric heat capacity are all treated as mixed polynomial functions of temperature and hydrogen concentration. The effect of stress on hydrogen diffusivity is however neglected. It is therefore necessary to couple the heat conduction to the hydrogen diffusion in order to achieve accurate results in predicting the temperature and hydrogen concentration profiles both under steady state and transient operating conditions. Accurate modeling of the coupled transient behavior will provide detailed information of the stress across the fuel as well as the necessary information for predicting the possibility of excessive hydrogen release from the fuel during accidents. Please refer to Appendix A for the list of all notations accompanied with the definitions and units.
2. METHODOLOGY

The transient one-dimensional radial heat equation with internal heat generation and variable properties is solved simultaneously with the diffusion equation. The coupling scheme for a single time iterate is presented below along with the pertinent differential equations:

\[ \frac{\partial}{\partial t} \rho C_p T = \frac{1}{r} \frac{\partial}{\partial r} \left( kr \frac{\partial T}{\partial r} \right) + q^* \]  

(1)

\[ J_r = -DN \left( \frac{dC}{dr} + \frac{T Q C}{T^2} \frac{dT}{dr} \right) \]  

(2)

Figure 1. A single time-step in the solution algorithm.

A semi-implicit Crank-Nicolson scheme is used for the time-discretization [2], while the hydrogen diffusion is solved for explicitly. The heat equation is solved for the current temperature using properties from the previous time-step and extrapolated properties for the current time step. Next, the hydrogen concentration is calculated for the current time-step using parameters only at the previous time-step. Third, the diffusivity, thermal conductivity, and volumetric heat capacity are updated with the current temperature and hydrogen concentration. This process is shown in Figure 1 where arrows denote inputs, circles are variables (dashed lines denote the previous time step), rectangles are equations, triangles are boundary conditions, and the hexagon is power (assumed independent of other variables).

Stress distribution across the material also needs to be addressed. The two sources of strain in the material arise from temperature and hydrogen concentration gradients across the fuel. Using constitutive equations coupled with the equilibrium condition and also assuming a plane strain scenario in the axial direction, a differential equation governing the radial stress across the fuel is determined (Eqn. 3). The two necessary boundary conditions are radially symmetric stress at the fuel centerline and zero radial stress at the fuel surface.

\[ \frac{1}{r^2} \frac{d}{dr} \left( r^3 \frac{d\sigma_r}{dr} \right) = -E \left[ \frac{d}{dr} (\alpha T) + \beta \frac{dC}{dr} \right] \]  

(3)

The radial equilibrium condition in cylindrical coordinates is used to calculate the azimuthal stress across the fuel based on the radial stress. To determine the distribution of axial stress across the fuel, the axial stress is first calculated assuming complete restraint in the axial
direction \((\varepsilon_z = 0)\). Then the difference from the average of this quantity across the fuel is denoted as the actual magnitude of axial stress [4].

3. RESULTS

3.1. Steady State Results

The steady-state temperature, H/Zr ratio, and axial stress distributions were computed. The results for both a liquid metal (LM) (infinite conductance assumed) and helium (He) (conductance of 0.57 W/cm\(^2\).K) fuel/cladding gap are shown at linear heat rates (LHR) of 100, 200, and 300 W/cm.

![Figure 2. Steady-state temperature, H/Zr ratio, and axial stress distributions for a LM gap and He gap at LHRs of 100, 200, and 300 W/cm.](image)

As expected, the fuel temperature gradient increases with LHR. The corresponding fuel temperatures are significantly higher with the He bond than the LM bond. The hydrogen concentration gradients are also steeper with increasing LHR. However the hydrogen redistribution is more severe in the case of LM bonded fuel. This is apparent by inspecting of the governing equation for the flux (Eqn. 2). The \(T^{-2}\) dependence of the temperature gradient term enhances its impact at lower temperatures. The largest component of stress is the axial stress, whose value is influenced by the temperature and hydrogen concentration gradients in an opposing manner. However, hydrogen-induced stresses are the dominant component, as is evident from the steady state results. Generally, the fuel surface experiences severe compression from axial and azimuthal components of stress, while all three components of stress are tensile at the central region of the fuel.

In general, the magnitude of the stresses increases with LHR due to more severe hydrogen redistribution. An exception to this is the He bonded fuel which experiences smaller stresses at a LHR of 300 W/cm than 200 W/cm. This is due to the higher average temperature (affecting the coefficient of thermal expansion of the material) and somewhat steeper thermal gradient which further counteracts the hydrogen-induced stress.
3.2. Transient Results

Four parametric transient case studies were completed with a LM gap, but only two will be discussed (cases 2 and 3). The nominal LHRs are 200 W/cm and the nominal coolant temperatures are 575K. Each case has its own characteristic power and surface temperature trace that is meant to represent a simplified accident that might occur in a LWR. The imposed transient boundary conditions and resultant transient temperature and axial stress distributions are shown. The hydrogen redistribution, although present, is miniscule since the hydrogen diffusivity is orders of magnitude smaller than the thermal diffusivity (\(2 \times 10^{-12} \text{ m}^2/\text{s}\) compared to \(6 \times 10^{-6} \text{ m}^2/\text{s}\)).

3.2.1. Case 2

In case 2, the power is maintained at the nominal value while the fuel surface temperature is mildly ramped up 100K over a 50s duration. This transient is similar to what could happen during a loss of flow accident (LOFA) in an LWR. The temperature and axial stress responses are shown below:

![Case 2 diagram](image)

Figure 3. Left to Right: Power/fuel surface temperature, temperature profile, axial stress distribution during the transient.

The temperature profile remains constant but shifts upwards with the gradual increase of surface temperature. The corresponding stress distribution is only slightly lowered. This is again due to the higher temperature and steeper thermal gradient that reduces the hydrogen-induced stress.

3.2.2. Case 3

For case 3, the power is pulsed to twice the nominal value for 2.5s and thereafter dropped to 5% while the fuel surface temperature remains constant. This case is an extreme example of a reactivity insertion accident (RIA) with a large pulse height and long pulse width, where the surface temperature response is neglected. The response of the fuel temperature and axial stress are shown below.
Figure 4. Left to Right: Power/fuel surface temperature, temperature profile, axial stress distribution during the transient.

The fuel temperature follows the power pulse, rapidly peaking and dropping down to the final state of case 1. The stress response of the fuel is interesting in that the stress is actually lowered and flattened during the power pulse. However, as the fuel cools, severe stresses corresponding only to the hydrogen concentration gradient develop. Figure 5 shows the maximum fuel temperature for various pulse heights and durations.

Figure 5. Maximum fuel temperature during power pulse.

4. CONCLUSION

Steady state and transient behavior of several aspects of the fuel operating performance have been investigated, taking into account the temperature and hydrogen concentration dependence of the fuel properties.

Steady state temperature, hydrogen concentration, and stress profiles for various linear heat rates and gap materials have been calculated. The LM bonded fuel experiences smaller average temperature, however the extent of hydrogen redistribution in the fuel is more extreme. Strains
in the fuel occur from thermal and hydrogen concentration gradients, with latter being the dominant contributor. Therefore LM bonded fuel is subject to higher stresses. Axial and azimuthal stresses are both compressive at the surface and tensile at the fuel centerline. All of these results are in agreement with what was previously shown by Olander [3].

In order to gain some understanding of the transient response of hydride fuels, several simplified scenarios were studied, for which power and fuel surface temperature were artificially altered as boundary conditions. The thermal response of the fuel to the changing boundary conditions is very quick (in order of few seconds) due to the small fuel rod and large thermal diffusivity. There is no discernable movement in the transient hydrogen profile. The steady-state distribution on the other hand is important to know for the initial conditions. The stress and properties respond instantaneously to temperature and concentration changes.

An interesting result is that during power pulses the stress across the fuel is actually reduced. The temperature-induced stresses counteract the hydrogen-induced stress, so the fuel is in its most relaxed state during this stage in transients. The fuel experiences maximum stresses when temperature gradients diminish and the hydrogen displacement remains.

REFERENCES


APPENDIX A

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Quantity</th>
<th>Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>k</td>
<td>Thermal conductivity</td>
<td>W/cm.K</td>
</tr>
<tr>
<td>ρ</td>
<td>Mass density</td>
<td>kg/cm³</td>
</tr>
<tr>
<td>C_p</td>
<td>Specific heat capacity</td>
<td>J/°C.K</td>
</tr>
<tr>
<td>T</td>
<td>Temperature</td>
<td>K</td>
</tr>
<tr>
<td>q''''</td>
<td>Volumetric heat generation rate</td>
<td>W/cm³</td>
</tr>
<tr>
<td>J_r</td>
<td>Radial flux</td>
<td>cm²/s</td>
</tr>
<tr>
<td>D</td>
<td>Macroscopic diffusion coefficient</td>
<td>cm²/s</td>
</tr>
<tr>
<td>N_Zr</td>
<td>Zirconium number density in δ-ZrH1.6 phase</td>
<td>atoms/cm³</td>
</tr>
<tr>
<td>C</td>
<td>H/δ ratio in ZrHx</td>
<td>-</td>
</tr>
<tr>
<td>T_Q</td>
<td>Heat of transport of hydrogen in δ-ZrH1.6 phase</td>
<td>K</td>
</tr>
<tr>
<td>β</td>
<td>coefficient of expansion of hydrogen</td>
<td>-</td>
</tr>
<tr>
<td>α</td>
<td>temperature dependent coefficient of thermal expansion</td>
<td>K⁻¹</td>
</tr>
<tr>
<td>ε_r, ε_θ, ε_z</td>
<td>radial, azimuthal, and axial strains</td>
<td>cm/cm</td>
</tr>
<tr>
<td>σ_r, σ_θ, σ_z</td>
<td>radial, azimuthal, and axial stresses</td>
<td>MPa</td>
</tr>
<tr>
<td>E</td>
<td>Elastic (Young’s) modulus</td>
<td>GPa</td>
</tr>
<tr>
<td>ν</td>
<td>Poisson’s ratio</td>
<td>-</td>
</tr>
</tbody>
</table>
MICROSTRUCTURAL EVOLUTION IN CERIUM DIOXIDE IRRADIATED WITH HEAVY IONS AT HIGH TEMPERATURE

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ABSTRACT

The nuclear fuel is subjected to high irradiation dose. Then the fuel, uranium dioxide (UO₂) with a cubic fluorite structure, leads to high burn-up structure over the dose. The structure is characterized as formation of subdivided grains and coarsened bubbles, which is attributable to deterioration of the nuclear fuels performance. Therefore, it is necessary to suppress its formation. Cerium dioxide (CeO₂) is expected as the simulation of UO₂ under energetic particle irradiation because both possess the same structure. CeO₂ is a well-studied oxide because of its technological applications, such as a major component in the catalysts, stable capacitors and so on. In order to clarify behavior of lattice defects in the structure, changes of crystal structure and surface morphology in CeO₂ were investigated. Polycrystalline samples were irradiated with 300keV-Xe, 12M eV-O and other ions at temperatures from 370 to 1000 K. Microstructural evolution and composition change was analyzed by scanning electron microscopy (SEM), Raman spectroscopy and X-ray diffractometry (XRD). Grain boundary etching, embossed shapes formation and bubble formation were observed at high temperature, however blistering was not detected. The tendency of surface features remarkably changed at 1000 K because of faster vacancies diffusion. Raman spectra and diffraction data indicated that oxygen vacancies accumulate at the early stage and reach the balance of their formation and recovery, and the formation is correlated with electronic excitation.

Key Words: cerium dioxide, Ion irradiation, Raman spectroscopy, X-ray diffraction, scanning electron microscopy
1. INTRODUCTION

The development of high burnup fuels is required to reduce the total amount of spent nuclear fuels and nuclear energy costs. Characteristics of high burn-up structure or “Rim structure [1-3]” are coarsened bubbles and subdivided grain, all of which result in FP releases, fuel temperature increase, swelling and so on. These phenomena are attributable to pellet-clad interaction and deteriorate the properties of nuclear fuels. Therefore, it is necessary to restrain the high burnup structure formation. There have been models for high burnup structure formation [4-6]. However, the formation mechanism still remains a matter of discussion because fuel pellets were usually investigated after irradiations. In this study, cerium dioxide (CeO₂) was selected so as to simulate the behaviors of uranium dioxide and as a first report the effect of ion-irradiation was investigated. The purpose of the study is that surface morphology change and composition change of CeO₂ are clarified.

2. EXPERIMENTAL METHODS

High purity (99.99 %) CeO₂ powders was pressed and sintered at 1670 K for 8 hours in air. The bulk materials were cut into disks of 9mm × 1mm followed by optical polish to achieve specimen for ion irradiations and the density was 95% of the theoretical density. Ion irradiation has performed at TIARA (Takasaki Ion Accelerators for Advanced Radiation Application) in JAEA. The samples were irradiated with single 300 keV X e⁺, 18 M eV I⁵⁺, 350 keV O⁺ and 12 M eV O³⁺ ion so as to simulate fission products and self ion. Sample temperatures under irradiation were measured by infrared radiation thermometer and were feed back to minimize beam heating effect. Irradiation conditions of CeO₂ specimens shows in Table 1. Irradiated samples were investigated with scanning electron microscopy (SEM), energy dispersive x-ray spectroscopy (EDS), Raman spectroscopy and x-ray diffractometry (XRD).

<table>
<thead>
<tr>
<th>Ion</th>
<th>300 keV X e⁺</th>
<th>18 M eV I⁵⁺</th>
<th>350 keV O⁺</th>
<th>12 M eV O³⁺</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flux [i/cm²]</td>
<td>2.9 × 10¹³</td>
<td>1.3 × 10¹²</td>
<td>6.4 × 10¹²</td>
<td>1.2 × 7.3 × 10¹²</td>
</tr>
<tr>
<td>Fluence [i/cm²]</td>
<td>2.1 × 10¹⁷</td>
<td>0.9 × 2.9 × 10¹⁶</td>
<td>3.1 × 7.8 × 10¹⁶</td>
<td>0.3 × 1.8 × 10¹⁷</td>
</tr>
<tr>
<td>Irradiation temperature[K]</td>
<td>388, 633, 893, 1003</td>
<td>773</td>
<td>478, 633</td>
<td>773, 1073</td>
</tr>
</tbody>
</table>

3. RESULTS AND DISCUSSION

The surface for CeO₂ disks irradiated with 300 keV X e⁺ was observed by SEM.

3.1. Surface morphology features

Fig. 1 shows SEM images of the surface for CeO₂ disks irradiated to 2.5 × 10¹⁷ i/cm². Surface morphology features under ion irradiations are described in the following. Priority etching of grain boundary by sputtering was observed in Fig. 1(b)-(d). And rounded dents were observed in...
Fig. 1. This results from sputtering and surface diffusion. Transgranular embossed shapes was observed in Fig. 1(b)-(d). The shapes result from self-organization by sputtering in competition with surface diffusion.

3.1.1. Irradiation temperature dependence of surface morphology

Fig. 1 shows SEM images of the surface for CeO$_2$ disks irradiated at $2.5 \times 10^{17}$ ions/cm$^2$ as a function of irradiation temperature. This figure show that grain boundaries were etched with increasing temperature, but the effect was recovered at 1003 K. And embossed shapes were remarkable with increasing temperature, but the variation between 893 K and 1003 K was unclear. We suggested that the characteristic variations at 1003 K indicate lattice defects migration evident.

3.1.2. Coarsened Xe bubbles formation at high temperature

Fig. 2(a) shows a SEM images of the surface irradiated to a fluence of $2.5 \times 10^{17}$ ions/cm$^2$ at 1003 K. Dot contrasts collecting around pits were observed. Fig. 2 showed that the contrasts corresponded with Xe element distribution by EDS. These results suggests that Xe bubbles are located immediately below surface of pits, however blistering which is one of main ion-irradiated surface effect was not observed.
3.1.3. Flaking at low temperature

Fig. 3 shows a SEM image of the surface irradiated to fluence of $5.5 \times 10^{17}$ ions/cm$^2$ at 388 K. Flaking was observed, but blistering was not observed. This result suggests that bubbles below surface don’t lead to blistering and generate flaking in CeO$_2$.

3.2. Structure analysis

3.2.1. Raman spectra

The main peak of the Raman spectrum in CeO$_2$ is F$_{2g}$ triple degenerate vibrational mode and corresponds with vibration of oxygen sublattice. Measurement region by Raman spectroscopy correspond with ion-implanted region ($300$ keV Xe$^+$, $350$ keV O$^+$) and directly below surface ($18$ MeV I$_{5+}$, $12$ MeV O$_3^+$) compared with vacancy and ion range distribution calculated by the SRIM-2006 code as shown in Fig. 4.

Fig. 5 shows F$_{2g}$ mode shift in CeO$_2$ irradiated with $300$ keV Xe$^+$, $350$ keV O$^+$, $18$ MeV I$_{5+}$, $12$ MeV O$_3^+$, F$_{2g}$ mode shift in irradiated CeO$_2$ varied toward lower frequency in initial irradiation period in Fig. 5. In high fluence region F$_{2g}$ mode shift in CeO$_2$ irradiated with $350$ keV O$^+$ varied toward higher frequency in Fig. 5(a). The one in CeO$_2$ irradiated with high energy ion at $773$ K varied toward lower frequency and the one at $1073$ K varied toward higher frequency in Fig. 5(b). Lower frequency variations are reported to result from effect of oxygen vacancies [7]. A higher sputtering rate by $300$ keV Xe$^+$ causes for those results and its variation corresponds with surface morphology by SEM observation. The variation toward higher frequency in CeO$_2$ irradiated with $350$ keV O$^+$ indicates that interstitial atoms accumulate. Continuous variation toward lower frequency at $773$ K indicates that vacancies accumulate. On the other hand variation toward higher frequency at $1073$ K indicates that vacancies annihilate.
3.2.2. X-ray diffraction

Fig. 6 show that lattice constant for CeO\textsubscript{2} irradiated with 18 MeV I\textsuperscript{5+}, 12 MeV O\textsuperscript{3+} at a function of fluence. Lattice constant increased in initial irradiation with increasing fluence. And then lattice constant showed a tendency to decrease. Lattice constant for CeO\textsubscript{2} irradiated with 12 MeV O\textsuperscript{3+} was asymptotic with increasing fluence.

![Figure 6. Lattice constant for CeO\textsubscript{2} irradiated with 18 MeV I\textsuperscript{5+}, 12 MeV O\textsuperscript{3+} at a function of fluence.](image)

**Figure 4.** Depth profiles of vacancy (open symbols) and ion range (solid symbols) distribution of CeO\textsubscript{2} irradiated with (a) 18 MeV I\textsuperscript{5+} (triangle), 12 MeV O\textsuperscript{3+} (square), (b) 350 keV O\textsuperscript{+}

**Figure 5.** \(F_{2g}\) mode shift in CeO\textsubscript{2} irradiated with (a) 300 keV Xe\textsuperscript{+}, 350 keV O\textsuperscript{+}; (b) 18 MeV I\textsuperscript{5+}, 12 MeV O\textsuperscript{3+}

3.2.2. X-ray diffraction

Fig. 6 show that lattice constant for CeO\textsubscript{2} irradiated with 18 MeV I\textsuperscript{5+}, 12 MeV O\textsuperscript{3+} at a function of fluence. Lattice constant increased in initial irradiation with increasing fluence. And then lattice constant showed a tendency to decrease. Lattice constant for CeO\textsubscript{2} irradiated with 12 MeV O\textsuperscript{3+} was asymptotic with increasing fluence.

![Figure 6. Lattice constant for CeO\textsubscript{2} irradiated with 18 MeV I\textsuperscript{5+}, 12 MeV O\textsuperscript{3+} at a function of fluence.](image)
Raman spectra and XRD results indicate following phenomena. In initial irradiation vacancies and interstitial atoms accumulate. While vacancies continuously accumulate near surface of CeO₂ irradiated with high energy ion at 773 K, vacancies annihilate at 1073 K. Interstitial atom accumulate near surface of CeO₂ irradiated with 350 keV O⁺ at 773 K. Then, defect productions and annihilations are asymptotic.

3. CONCLUSIONS

Disks of CeO₂ irradiated at irradiation temperatures between 388 K and 1073 K, with 300 keV Xe⁺, 18 M eV I⁵⁺, 350 keV O⁺ and 12 M eV O³⁺. The changes in surface morphology and lattice defects in these pellets were studied by SEM/EDS observation, Raman spectroscopy and XRD. The results were summarized as follows:
1. The surface of irradiated CeO₂ showed grain boundary etching, rounded dents and transgranular embossed shapes.
2. Xe bubbles, the size of 100 nm, below surface of pits were observed, and were formed as surface move by etching (893 K, 1003 K). Flaking was observed, and was formed by Xe accumulation (388K).
3. The variations of SEM, the Raman peak and lattice parameter at about 1000 K give suggestions that the oxygen vacancies more diffuse over the temperature.
4. The variations of the Raman peak and lattice parameter in irradiated CeO₂ indicate that interstitial atoms accumulate (350 keV O⁺) and production of defects is balanced out by annihilation (12M eV O³⁺).

ACKNOWLEDGMENTS

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REFERENCES

SCREENING OF FULLERENE C$_{60}$ CRYSTALLIZATION USING MICROFLUIDIC DEVICES

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ABSTRACT

We have carried out screening of C$_{60}$ crystallization using microfluidic devices. By changing temperature, residence time, and solvent, unique structural and optical properties of C$_{60}$ crystals were found in microfluidic environment.

Key Words: Fullerene, Microfluidics, Liquid-liquid interface

1. INTRODUCTION

Since their initial discovery in 1985 [1], Fullerenes C$_{60}$ have attracted significant attention for their unique physical and chemical properties, and have resulted in the creation of a new research field. Fullerene C$_{60}$ is also excellent candidate of new medicine. Since the metastable crystals showed different solubility and bioavailability from those of stable crystals, screening of metastable or stable phases of crystals is critical for development of new medicine. In terms of protein crystals, efficient screening techniques using microfluidic chips had already been presented [2]. Here, we have developed a screening method of C$_{60}$ crystallization using microfluidic devices.
2. EXPERIMENTAL

2.1. Raw Material.
The raw material used was a commercial product of 99.5% $C_60$ powder formed by the electric-arc technique (Lot. No.020715 made by the Honjo Chemical Co., Tokyo, Japan).

2.2. Screening of Structural Properties of $C_60$ Crystals in the Microchannel Reactor.
The test solutions were toluene with $C_60$ and alcohol, including isopropanol, ethanol, and methanol. In all the experiments, the concentration of $C_60$ dissolved in toluene was constant at 0.58 mg/mL (0.13 mol%), which was a quarter of the saturation concentration at 25°C. Two solutions were kept at a low temperature of 0°C by a temperature controller until just before introduction into the channels. Toluene and alcohol were introduced into a Y-shaped microchannel with 100 μm width, 40 μm depth and 40 mm length (ICC-SY05, Institute of Microchemical Technology Co., Kawasaki, Japan) independently with a flow rate of 1.0 μL/min (Fig. 1A). The microchannel was fabricated on the glass substrate by wet etching technique. The microchannel was immersed in a water bath and a water immersion objective lens (Olympus LUMPlanFl M = 100x, NA = 1.0, Olympus Co., Tokyo, Japan) was used for in situ observations (Fig. 1B). The room temperature was kept at 20°C using an air conditioner. The starting water temperature was 5°C as set by the temperature controller until just before introduction of the test solutions. Subsequently, the water bath was left out at room temperature. Therefore, the temperature of the water bath increased and approached 20°C gradually during the experiments. The water bath temperature was constantly checked by a thermometer.

In the microchannel, two solutions formed a liquid/liquid interface and multi-stream laminar flow [3]. With flow, the two solutions did not mix because of extremely low Reynolds number $Re = 0.23$. After 30 seconds, the formation of the liquid/liquid interface was confirmed (Fig. 2A-a) and the fluid flow was stopped in order to mix the two solutions. If any crystallization was confirmed, we noted the mixing time and the water bath temperature. To capture the crystals onto a microgrid for SEM and TEM observations, the test solutions were soon introduced again at 1.0 μL/min for 30 seconds. The mixing time was then reset. After 30 seconds, the fluid flow was stopped again and fresh test solutions started to mix. We repeated this process: introduce, stop, mix, observe, check mixing time and temperature, wash away, stop.

**Figure 1. Schematic of experimental setup.**
2.3. Screening of Optical Properties of C$_{60}$ Crystals in the Microchannel Reactor.

Apart from the structural screening of crystallization described above, their optical properties were also investigated. The sample C$_{60}$ crystals were also synthesized by a microfluidic liquid-liquid interfacial precipitation method. The test solutions were C$_{60}$-toluene solution and alcohol. The formation of C$_{60}$-alcohol solvates were suggested in microchannel reactors. In order to investigate the effect of alcohol molecules, two kinds of alcohol including isopropanol (IPA) and ethanol (ETH) were used. The synthesis apparatus was same as Figure 1. The room temperature and the water bath temperature was maintained at 17°C using a thermocontroller. A C$_{60}$-toluene solution (good solvent) and alcohol (poor solvent) isopropanol (IPA) or ethanol (ETH) were introduced into the inlet channels by a pressure-driven microsyringe pump independently with a constant flow rate. In order to examine the effect of the flow rate on C$_{60}$ crystallization, nucleation of crystals was carried out under the different flow rate conditions of 1.0, 10, or 100 μL/min, respectively.

The intermolecular bonding in the C$_{60}$ rods was investigated by means of Raman and PL spectroscopy. The Raman spectra and the PL spectra of the samples were measured by a spectrometer (Renishaw in Ramascope, U.K.) using a CCD detector. The probing laser was an Argon ion laser (514 nm excitation). The irradiation power for the Raman measurement was 0.1 mW. The irradiation power for the PL measurements was changed from 0.02 to 12.5 mW. We carefully checked that no changes occurred in the samples by illumination of the excitation-probing laser.

3. RESULTS AND DISCUSSION

3.1. Structural Properties of C$_{60}$ Crystals

In the first experiment, we introduced toluene with C$_{60}$ and IPA, which is the usual combination for C$_{60}$ nanowhiskers (NWs) synthesis [4], into the channels. Within 5 minutes after the fluid flow was stopped, at the bath temperature of 6°C, C$_{60}$ crystals were formed in Fig. 2A-b. The lengths of the crystals were approximately 30 μm. The center axis areas of the crystals were semi-transparent in left inset Fig. 2b. In addition, several crystals had crosswise cut edges in right inset Fig.2A-b. Scanning electron microscopy (SEM) revealed the reason for the formation of the semi-transparent NWs in Fig. 2B-a. Several “macaroni”-like crystals with 7 μm diameter were observed. Their cross-sections were hexagonal, corresponding to the normal C$_{60}$ NWs [4]. Another type of tube which was cut crosswise at the edge was also observed in Fig. 2B-b. The edge outline was not parallel to the center axis, indicating that the cross-sections of these “penne”-like crystals were also hexagonal. The characteristics of the cross-sections of these crystals were emphasized in another magnified SEM image in Fig. 2B-c. Within the void area, multi-layers of the crystal were observed. The cross-section of the void was not circular but was hexagonal. Some spherical crystalsshown in Fig. 2B-d were also observed on the same microgrid. Within 15 minutes after the flow was stopped, at a bath temperature of 9°C, tree-like NWs appeared in Fig. 2A-c. A number of branches were observed to sprout from a main trunk. The main trunks and the branches were approximately 50 μm and 10 μm in length, respectively. The main trunks originated at the upper channel wall, indicating that nucleation of the crystals occurred at the wall on the toluene side. Thirty-two minutes after the flow was stopped, at a bath temperature of 16°C, short rhombus-shaped columns with 10 μm length were formed in Fig. 2B-e. Their aspect ratio was smaller than that of normal NWs. Forty minutes after the flow was stopped, at a bath temperature of 18°C, the typical C$_{60}$ NWs with high aspect ratios appeared in Fig.
2B-f. Their lengths and diameters were more than 20 μm and less than 5 μm, respectively. These corresponded well with the properties of the well-known C60 NWs synthesized via the toluene-IPA combination [4].

In the flask experiments, the usable alcohol solution (poor solvent) was limited to IPA for the synthesis of C60 NWs crystals. To change the solvated state and the supersaturation of C60 from the conditions of the previous studies, in the second experiment, we attempted to use ethanol instead of IPA. One minute after the flow was stopped, at a bath temperature of 5°C, crystals with hollow ended appeared on the toluene side in Fig. 2A-d. The size was about 4 μm in diameter and 30 μm in length. A transmission electron microscope (TEM) micrograph showed the outline of the open side void in Fig. 2C-a. Five minutes after the flow was stopped, at a bath temperature of 7°C, many particles of a few micrometers size were formed on the toluene side in Fig. 2A-e. Several sprouts from the particles were also observed (right arrow in Fig. 2A-e). The sprouts grew in one direction. On the other hand, a Y-shaped branch crystal was also observed (left arrow). The Y-shaped crystal had two (Fig. 2C-b) or three (Fig. 2C-c) of the small separation branches. Seven minutes after the flow was stopped, at a bath temperature of 8°C, several crystals with sharp edges appeared in Fig. 2A-f. The crystals radiated in all directions and became multiple pods. The crystals had 6-8 columns with two- (Fig. 2C-d) or three-dimensional structures (Fig. 2C-e). Some crystals had acute edges in Fig. 2C-f. The size of each crystal was about 20 μm, and interestingly, each other column was positioned at a regular angle of 60°.

Summarizing the bulk structures described above, we noticed a certain theme. The bulk structures of the C60 crystals synthesized in the microfluidic device were very similar to those of snow crystals. Snow crystals have several types of structures including plate, prism, column, end-hollow column, dendrite, and needle. Each structure type has been categorized by temperature and supersaturation of water [5].

Figure 2. (A) Optical micrographs and SEM images of C60 crystals synthesized in (B) toluene-IPA and (C) toluene-ethanol systems
3.2. Optoelectronic Properties of C$_{60}$ Crystals

The intermolecular bonding of the C$_{60}$ crystals synthesized in the microchannel reactor was investigated by means of PL spectroscopy. Fig. 3a shows comparison of the C$_{60}$ rods with the bulk C$_{60}$ crystal synthesized by the toluene-isopropanol system at the flow rate of 1.0 $\mu$L/min. One emission band around 1.70 eV, known as the zero-phonon line (ZPL) corresponding to direct exciton-polaron recombination, was observed. The PL intensity of the rods was one order of magnitude higher than that of the bulk crystal, indicating that Herzberg-Teller coupling between the 3F1g and 1Ag states was stronger in the C$_{60}$ rods. This may be attributed to a decrease in the symmetry of the C$_{60}$ lattice structure by the doping of the guest solvent molecules [6].

To investigate the doping effect, we measured the PL spectrum as a function of the excitation laser power. Fig. 3b shows a typical example of the C$_{60}$ rods in Tol-ETH-10 -10 $\mu$L/min. The intensity of 1.73 eV peak increase in proportion to the laser intensity. There was no peak at 1.50 eV, which is assignment to photopolymer of C$_{60}$, indicating that any laser-induced polymerization did not occur during the PL measurement. The integrated PL intensity was plotted on a log-log scale as a function of the excitation laser power $I_{EX}$ (Fig. 3c). The integrated PL intensity $I_{PL}$ was analyzed in terms of the equation: $I_{PL} \propto I_{EX}^\alpha$ where $\alpha$ is the power index that provides a good measure for the characterization of the PL emission mechanism. There was no significant difference in PL intensity-power dependence between the different flow rates. Conversely, an apparent difference between toluene-isopropanol systems and toluene-ethanol systems was observed. In the toluene-isopropanol systems, $\alpha$ (gradient) was ~0.3-0.5. On the other hand, in the toluene-ethanol systems, $\alpha$ (gradient) was ~0.7-0.9. This difference suggested that the guest alcohol molecules were trapped in the lattice structure of the C$_{60}$ rods, and they affected the excited electronic states. In general, not alcohol solvent (poor solvent) molecules but toluene (good solvent) molecules form solvates with Fullerene C$_{60}$ [7]. Conversely, the data obtained here suggested that alcohol solvent molecules were significant for optoelectronic properties of C$_{60}$ crystals in microfluidic environment.

In addition, we noticed transition of the peak positions. Fig. 3d shows the peak positions of the C$_{60}$ rods around 1.70 eV (ZPL). The peak positions of the C$_{60}$ rods synthesized in the microfluidic device were blue-shifted compared with those in raw C$_{60}$. A clear difference in the peak positions between the toluene-isopropanol systems and the toluene-ethanol systems was observed. In toluene-ethanol systems, the peak positions were more blue-shifted compared to those in toluene-isopropanol systems. Since HOMO-LUMO transitions are sensitive to the proximity of adjacent molecules, the enhanced luminescence (Fig.3a) and the blueshift may be the result of distortion of the fcc structure induced by the doping of the guest alcohol (poor-solvent) molecules. This is a specific phenomenon in the microfluidic environment.

3. CONCLUSIONS

By exploiting microfluidic environment, various metastable phases of C$_{60}$ crystals were confirmed. If further new phases of C$_{60}$ crystals were found after the C$_{60}$ drug medicine were placed on the market, its social impact would be serious. The present method developed here and findings suggest strong need of urgent screening of metastable C$_{60}$ crystals for development of Fullerene drug and medicine. In addition, we have reported poor solvent molecule-dependent luminescence properties of C$_{60}$ rods formed in a microfluidic environment. The rods look promising as novel materials for micrometer-sized photodetectors, thermal storage devices, solvent molecule sensors, and flow sensors. Unlike for bulk-scale synthesis, this finding suggests
the significance of the poor solvent molecules and flow rate on the optoelectronic properties for microfluidic synthesis of micro- and nanocrystals.

ACKNOWLEDGMENTS

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REFERENCES

Session 4: Panel Discussions
“Society, International Education, and Nuclear Technologies”

Moderated by
Dr. T. Isaacs
LLNL

Panelists:
Jasmina Vujic, Professor, UCBNE
Joonhong Ahn, Professor, UCBNE
Jor-Shan Choi, Professor, UTNEM, LLNL
Taira Okita, Associate Professor, Department of Quantum Engineering and Systems Science, UT
Takaharu Fukuzaki, Professor UTNEM
Yoshiaki Oka, Professor, UTNEM

3:00 PM – 4:50 PM
January 8, 2008
Education and Research of Nuclear Engineering and GoNERI program of University of Tokyo

Yoshiaki Oka
Professor, Department of Nuclear Engineering and Management, School of Engineering, The University of Tokyo
oka@nuclear.jp

Panel discussion “Society, International Education and Nuclear Technology” UCB, January 8, 2009

Contents

1. Foundation of Two Nuclear Departments
2. Department of Nuclear Engineering and Management
3. Nuclear Professional School
4. Global COE Program “Nuclear Education and Research Initiative” (GoNERI)
1. Foundation of Two Nuclear Departments

The University of Tokyo

Graduate School of Engineering

Department of Quantum Engineering and Systems Science

Department of Nuclear Engineering and Management

Nuclear Professional School

Started in April of 2005 as the 20th and 21st departments of the Graduate School of Engineering

Operated and managed together

Cooperated in some studies

2. Department of Nuclear Engineering and Management

- Regular graduate school with master course and doctoral course
- Unique features to cover the social science aspects as well as the nuclear science and engineering

Consisting of three main research areas

Nuclear-Socio Engineering

Nuclear Energy

Radiation Application
Capacity and foreign student program

**Capacity:**
- 23 students per year for the master course
- 13 students per year for the doctoral course

Foreign students will be accepted outside the capacity.

**Special graduate program for foreign students with scholarship since 1989:**
- in collaboration with the Department of Quantum Engineering and Systems Science, and
- fourteen lectures in English.

The number of foreign students in these 17 years:
125 (including 91 of the special program)

Research Facilities

- the fast neutron source reactor,
- femto-second linear accelerator and femto-second terra-watt laser system,
- a tandem Van de Graaf accelerator and a Tandetron for heavy irradiation research, and
- a tandem Van de Graaf accelerator for accelerator mass spectrometry etc.
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### 3. Nuclear Professional School

**Purpose**

- To meet the needs of highly educated professionals such as chief licensed reactor engineers of nuclear power plants
- To establish comprehensive schooling of nuclear engineering

**Administration**

- Located in Tokai-mura
- A type of professional school like a law school
- Jointly operated with JAEA (Japan Atomic Energy Agency) and the University of Tokyo
- Taking the responsibility and converting Nuclear Engineering Research Laboratory (NERL) into the Nuclear Professional School
Education in Nuclear Professional School

Characteristics

- One-year graduate school providing extensive schooling, but requiring no thesis study. Capacity: 15 Japanese students.
- Giving master’s degree of nuclear professional.
- Lectures are sent to Tokyo (Dep. NEM) via internet TV.

Contents

- Lectures, Exercises and Experiments of full 5 days of summer and winter semesters.
- Exercises and Experiments and Internship between the semesters.

Characteristics

- Contents at the international training center of JAEA.
- Exercises at the plant simulator of JAPC (Japan Atomic Power Company).
- Exercises at the facilities of the University of Tokyo and JAEA such as research reactors, critical assemblies and thermal hydraulics loops.

Textbooks

Textbooks

- Early textbooks:
  - out of print,
  - not well-covered progress and practice of the nuclear power utilization in these 30 years.

- Nineteen textbooks:
  - under preparation,
  - including nuclear socio-engineering,
  - for human resource development in Japan.
4. Global COE program of MEXT (J SPS)
Nuclear Education and Research Initiative

Systematic Education and Research including Nuclear Energy Sociology

First in the World

Nuclear Energy Sociology
What is Technology for Society?
In collaboration with people outside Univ.

Nuclear Energy
Technology Innovation
Through comprehensive and interdisciplinary approach

Radiation Application
Therapy, diagnosis, biology, etc.
Spread in interdisciplinary fields: medicine, agriculture and so on

“We prepare next generation researchers to grasp the perspectives of complicated and divergent fields of nuclear energy.” - Dr. Yoshiaki OKA, Prof. UT, Program Leader -

4.1. Nuclear Energy Sociology

Nuclear Law/Legislation
- To pursue the issue of efficient yet safe nuclear energy
- To deal with the relation between technology and law

Nuclear Non-Proliferation
- To coexist with the peaceful use of nuclear energy
- To identify the technological and systematic problems

Public Communication
- To inform the public along with ways to improve general science and technological understanding
- To facilitate the education through civil lectures

Method: Practical Education
- Law/Legislation: Cooperation with Prof. Shiroyama of the Graduate School of Law and Public Policy
- Non-Proliferation: Collaboration with visiting professors, Internship to IAEA
- Communication: Collaboration with Visiting Professor and Public Relation Specialist

To produce Ph.D students with the following capabilities:
- To study the regulation system, to identify problems, and to find solutions
- To serve in policy making in the world with the necessary expertise
- To communicate with the public on social aspects of nuclear energy
4.2. Nuclear Energy

To develop University-of-Tokyo Originated and Japan Leading technologies for worldwide contribution

Three Key Subjects

- Future Nuclear Energy System
- Disposal of Radioactive Waste
- Safe and Stable Operation

Roadmap of Nuclear Energy R&D

Now 2020 2040 2060

4.2.(1) Future Nuclear Energy System

Example of studies

Japan, 10 EU organizations, Canada, Korea, China, etc.
UT leads R&D by 1.6 billion yen competitive fund

To provide those staff who can understand multiple disciplines and its inter-field connection by the design study

UT-Origin. World-Leading MPS Method:
The first grid-less method of calculating incompressible fluid/solid for split, splash and large deformation
Awarded by many prizes: Japan Society for the Promotion of Science (2006), etc.
Utilized in many fields: ship and ocean, civil engineering, bionics, video, etc. as commercially in some cases

To understand and resolve Nuclear Complicated Problem, such as Nuclear Accident, by inter-field approach
4.2.(2) Radioactive Waste / Fuel Cycle

Collaboration with oversea research institutes through international hub office

Deepening Cooperation of earth science and Nuclear-socio engineering for problem resolution to contribute the world

4.2.(3) Nuclear Power Plant Maintenance

Utilization of operational experience from many power plants

Safe/ Stable operation

To lead safe/stable nuclear utilization by combining research fields
4.3. Radiation Application

**R&D Medical Physics**
Application of the capability and techniques into space-time control of radiation therapy, diagnosis and inspection by developing table-top accelerator for cancer treatment and micro-PET.

**Radiation Chemistry**
Initial reaction stage R&D by highest performance pulse radiolysis device, and its application to radiobiology.

**Radiation Safety/Biology/Environment**
Radiation safety, radiobiology and earth environment analysis by accelerator mass spectrometry, etc.

**To lead world by Advanced R&D on Radiation Application**

4.4. Doctor-Course Student Development Program

- **Wide view education program**
  Student evaluation, Research associate, Oversea discipline, Inter-department lecture, Internship, attendance to several outside seminars, and faculty development.

- **International education program**

- **Young International workshop**
  International workshop by young researchers, Summer school, Internship at hub institutes in Japan/world, Personal connection network.

- **Internship to venture business**
  Experience on commercialization process of research result, Communication with entrepreneurs.

- **Practical study on Nuclear-Scio Engineering**
  Civil lecture, Joint guidance/research with visiting professors, Internship, Interdisciplinary collaboration in the university: Graduate School of Public Policy, etc.
What is definition of risk? (definition of safety)

1. By Reactor Engineer / Scientist
2. By Socio-psychologist (Public)
3. By Manager, CEO, Investor

Definition of Risk

Reactor engineer; Probabilistic risk
Risk=probability (accident frequency) X consequence (public deaths/accident)

Risk communication specialist (general public);
Risk=Danger + Big anger

Manager, CEO, Investor;
Risk=Uncertainty affecting company profit
=Expected profit - Real profit
There is no relation between probabilistic risk and risk perception of general public. “Risk” defined by communication specialists is the risk perceived by the public, not probabilistic risk.

Probabilistic risk does not include social risk such as damage by rumor and psychological stress etc.

Late prof. Yasumasa Tanaka

“Big anger” of general public is affected by;

- How big the perceived danger is.
- Lack of knowledge of the risk
- Distrust of the risk management system and its organization
- Degree of media attention
cause of anger

- fear
- threat to himself/herself and the family
- disappointment
- helplessness
- feel of neglected/slighted
- feel of disregarded/ignored

risk / uncertainty of us nuclear power business

- waste disposal (yucca)
- disaster (price-anderson)
- regulatory (nrc / doe)
- technology / design
- development / plant siting
- transmission availability
- construction
- commissioning
- operating
- fuel price / supply
- demand
- dispatch
- lawsuit
Purpose of education and research of nuclear-socio science at UT and GoNERI

• Reduce the risk / uncertainties of nuclear utilization for achieving level playing field in the competitive / de-regulated economic environment.
• Make technical people to understand the difficulty / complexity of public communication and social science aspects of nuclear utilization.
# List of participants

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