Office of Nuclear Energy, Science and Technology

Advanced Reactor, Fuel Cycle, and Energy Products Workshop for Universities

WORKSHOP PROCEEDINGS

June 16-17, 2005

DoubleTree Hotel and Executive Meeting Center
1750 Rockville Pike • Rockville, MD 20852

Table of Contents

Announcement of the FY 2005 Advanced Reactor, Fuel Cycle, and Energy Products Workshop for Universities

The Department of Energy (DOE)'s Office of Nuclear Energy, Science and Technology will hold a workshop for universities on June 16 and 17, 2005, at the DoubleTree Hotel in Rockville, MD. The workshop will provide U.S. universities the opportunity to become familiar with the research and development (R&D) requirements of the various programs of the Office of Nuclear Energy, Science and Technology.

Program plans and schedules including R&D requirements for the Generation IV Nuclear Energy Systems Initiative (Generation IV), the Advanced Fuel Cycle Initiative (AFCI), and the Nuclear Hydrogen Initiative (NHI) programs will be presented by DOE and laboratory program leads. Ample opportunity will be provided for questions and constructive dialogue on these research needs. The workshop will feature breakout sessions in fourteen research areas of the programs. All breakout sessions will be repeated the second day to allow individuals to attend more topics. The preliminary agenda for the workshop is attached.

Following the workshop, a solicitation is expected to be issued in June 2005. The solicitation, which will be open to all U.S. universities, will provide an opportunity for universities to participate in these research initiatives. Limited collaboration with national laboratory and industry partners is allowed. The university research projects are expected to be for one to three years. Applications in response to the solicitation will be due in August 2005 with awards selection scheduled for November 2005. FY 2006 funding for this solicitation will be from the budgets of the Generation IV, AFCI, and NHI programs and could total as much as $4 million.

Workshop Attendance: A pre-registration site has been set up for the workshop at the following web address: http://www.energetics.com/univworkshopjune05.html. Attendees should pre-register for the workshop by **May 15, 2005** to assist DOE in planning for seating and other arrangements. In addition, pre-registration will facilitate the gathering of contact information on the attendees for dissemination of future program-related information. If you have any questions regarding the pre-registration process, please contact MaryLee Blackwood or Katie Pluebell at (410) 290-0370.

Information: For more information regarding the workshop and R&D programs, please visit http://neri.inl.gov. For any additional questions, please contact Suibel Schuppner (301-903-1652) or Andrew Griffith (301-903-7120), or send an e-mail to NERI@nuclear.energy.gov.

U.S. Department of Energy
Office of Nuclear Energy, Science and Technology

Advanced Reactor, Fuel Cycle, and Energy Products Workshop for Universities

June 16-17, 2005

DoubleTree Hotel and Executive Meeting Center
1750 Rockville Pike
Rockville, MD 20852
Tel: 1-800-222-8733 / 301-468-1100
Fax: 301-468-0163

Agenda

Thursday, June 16, 2005

7:00AM-8:00AM	**Registration/Continental Breakfast**
8:00AM-12:00AM	**Plenary** *(Plaza Ballroom)*
8:00AM-8:15AM	**Welcome Remarks** • Shane Johnson – Deputy Director for Technology, NE
8:15AM-8:45AM	**Overview/Purpose** • Andrew Griffith – NERI Program Manager • Buzz Savage – NE Program Director for Advanced Fuel Cycle Initiative • Rob Versluis – NE Program Director for Generation IV Initiative • David Henderson – NE Program Manager for Nuclear Hydrogen Initiative
8:45AM-9:45AM	**Ten Minute Presentations** by National Technical Directors and System Integration Managers • Systems Analysis - *Kathy McCarthy* • Design and Evaluation Methods - *Hussein Khalil* • Fuels - *Kemal Passamehmetoglu* • Materials - *Bill Corwin* • Transmutation - *Mike Cappiello*
9:45AM-10:15AM	**Break**

10:15AM-12:00AM **Ten Minute Presentations** by National Technical Directors and System Integration Managers

- Energy Conversion - *Paul Pickard*
- Separations - *Jim Laidler*
- Very-High - Temperature Reactor- *Phil MacDonald*
- Supercritical-Water-Cooled Reactor - *Mike Modro*
- Lead-Alloy Liquid-Metal-Cooled Fast Reactor - *Bill Halsey*
- Gas-Cooled Fast Reactor - *Kevan Weaver*
- Thermochemical Cycle - *John Kolts*
- High-Temperature Electrolysis - *Steve Herring*
- Reactor-Hydrogen Production Process Interface - *Steve Sherman*

12:00PM-1:00PM **Lunch** *(Atrium Restaurant)*

1:00PM-3:00PM **Break-out Sessions by Topic Area** – Meeting Center Breakout Rooms

- Materials *(Adams Room)*
- Systems Analysis *(Wilson Room)*
- Fuels *(Jefferson Room)*
- Reactor-Hydrogen Production Process Interface *(Truman Room)*
- Thermochemical Cycle *(Monroe Room)*
- Separations *(Lincoln Room)*
- Supercritical-Water-Cooled Reactor *(Jackson Room)*

3:00PM-3:30PM **Break**

3:30PM-5:30PM **Break-out Sessions by Topic Area** – Meeting Center Breakout Rooms

- Transmutation *(Wilson Room)*
- Design and Evaluation Methods *(Truman Room)*
- Very-High-Temperature Reactor *(Jefferson Room)*
- Gas-Cooled Fast Reactor *(Monroe Room)*
- High-Temperature Electrolysis *(Adams Room)*
- Lead-Alloy Liquid-Metal-Cooled Fast Reactor *(Lincoln Room)*
- Energy Conversion *(Jackson Room)*

Friday, June 17, 2005

7:00AM-8:00AM **Continental Breakfast**

8:00AM-10:00AM **Break-out Sessions by Topic Area** – Meeting Center Breakout Rooms

- Materials *(Adams Room)*
- Systems Analysis *(Wilson Room)*
- Fuels *(Jefferson Room)*
- Reactor-Hydrogen Production Process Interface *(Truman Room)*
- Thermochemical Cycle *(Monroe Room)*
- Separations *(Lincoln Room)*
- Supercritical-Water-Cooled Reactor *(Jackson Room)*

10:00AM-10:30AM **Break**

10:30AM-12:30PM **Break-out Sessions by Topic Area** – Meeting Center Breakout Rooms

- Transmutation *(Wilson Room)*
- Design and Evaluation Methods *(Truman Room)*
- Next Generation Nuclear Plant *(Jefferson Room)*
- Gas-Cooled Fast Reactor *(Monroe Room)*
- High Temperature Electrolysis *(Adams Room)*
- Lead-Alloy Liquid-Metal-Cooled Fast Reactor *(Lincoln Room)*
- Energy Conversion *(Jackson Room)*

Notes: 1) The break-out topics covered in the June 16 afternoon sessions are repeated in the June 17 morning sessions to allow individuals to attend more topics.
2) Assigned breakout rooms are noted in italics following the topic area description.

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Plenary Presentations

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

R. Shane Johnson

Acting Director

Office of Nuclear Energy, Science and Technology

Opening Remarks

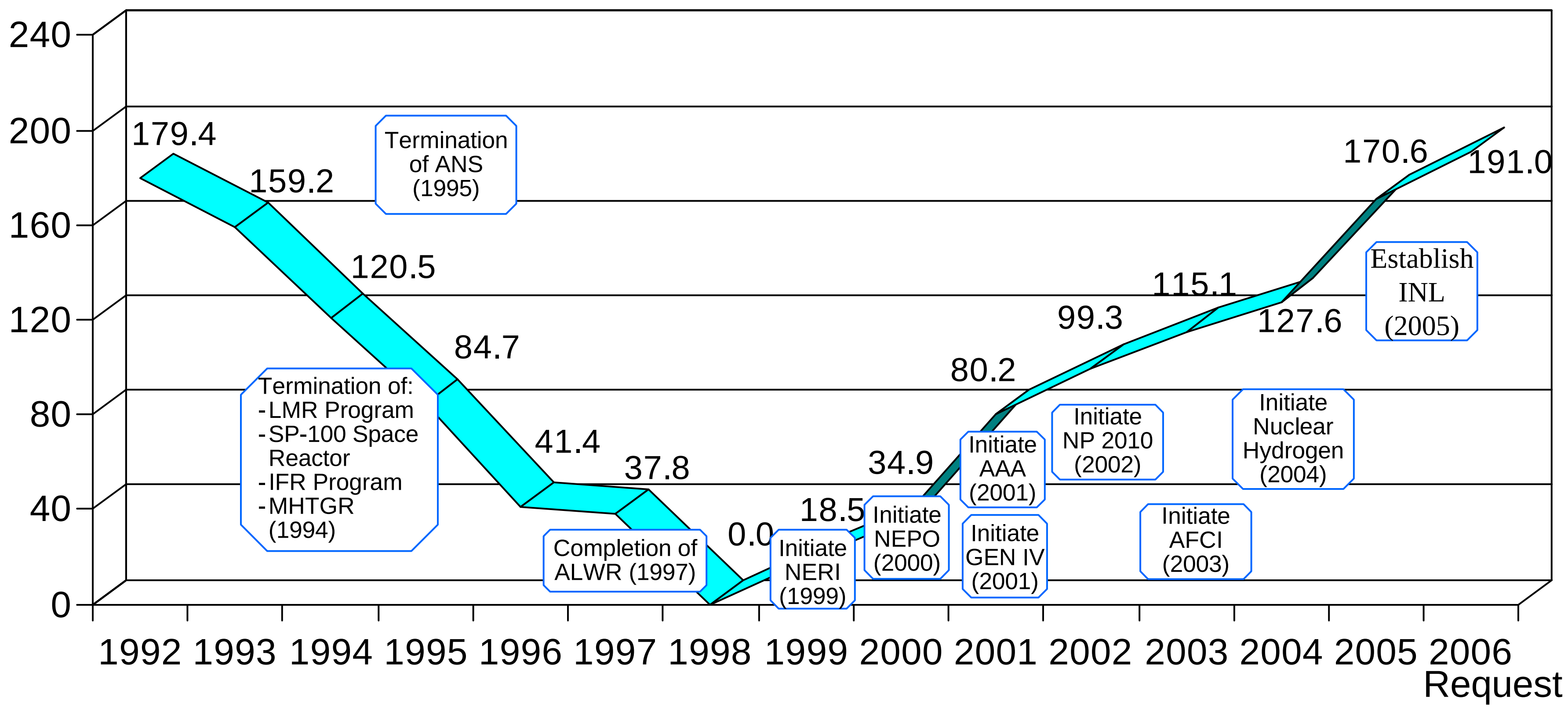
Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Nuclear Energy Research and Development

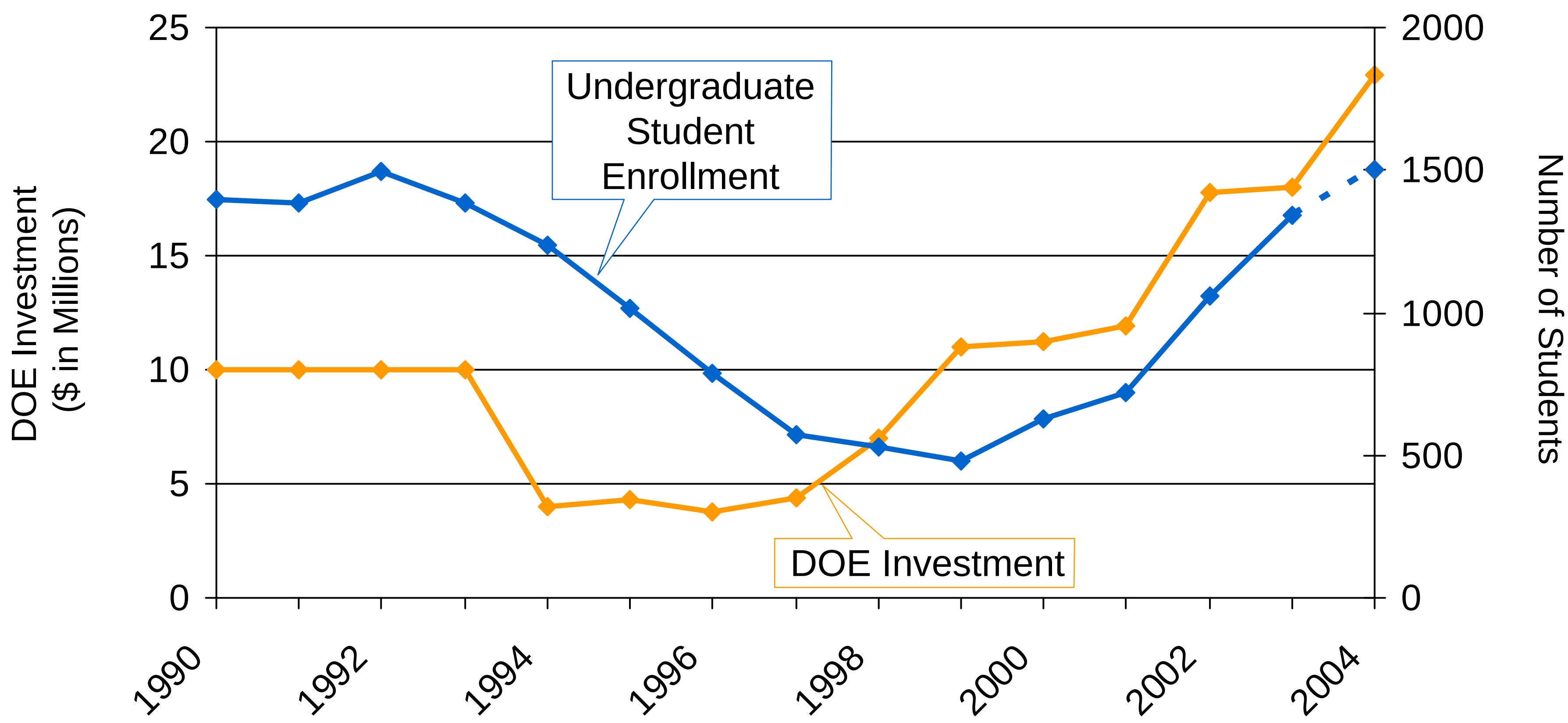
- Our goal is to sustain and broaden the benefits of clean, safe, and secure nuclear energy.
- We accomplish this goal by investing with industry, the national laboratories, and the university community to overcome institutional and technological barriers to reliance on nuclear energy.
- We engage the international community to coordinate our efforts and multiply their effect.

Resurgence in Nuclear Energy Research and Development

Graph of Federal Nuclear R&D Budgets



Resurgence in Nuclear Engineering University Enrollments

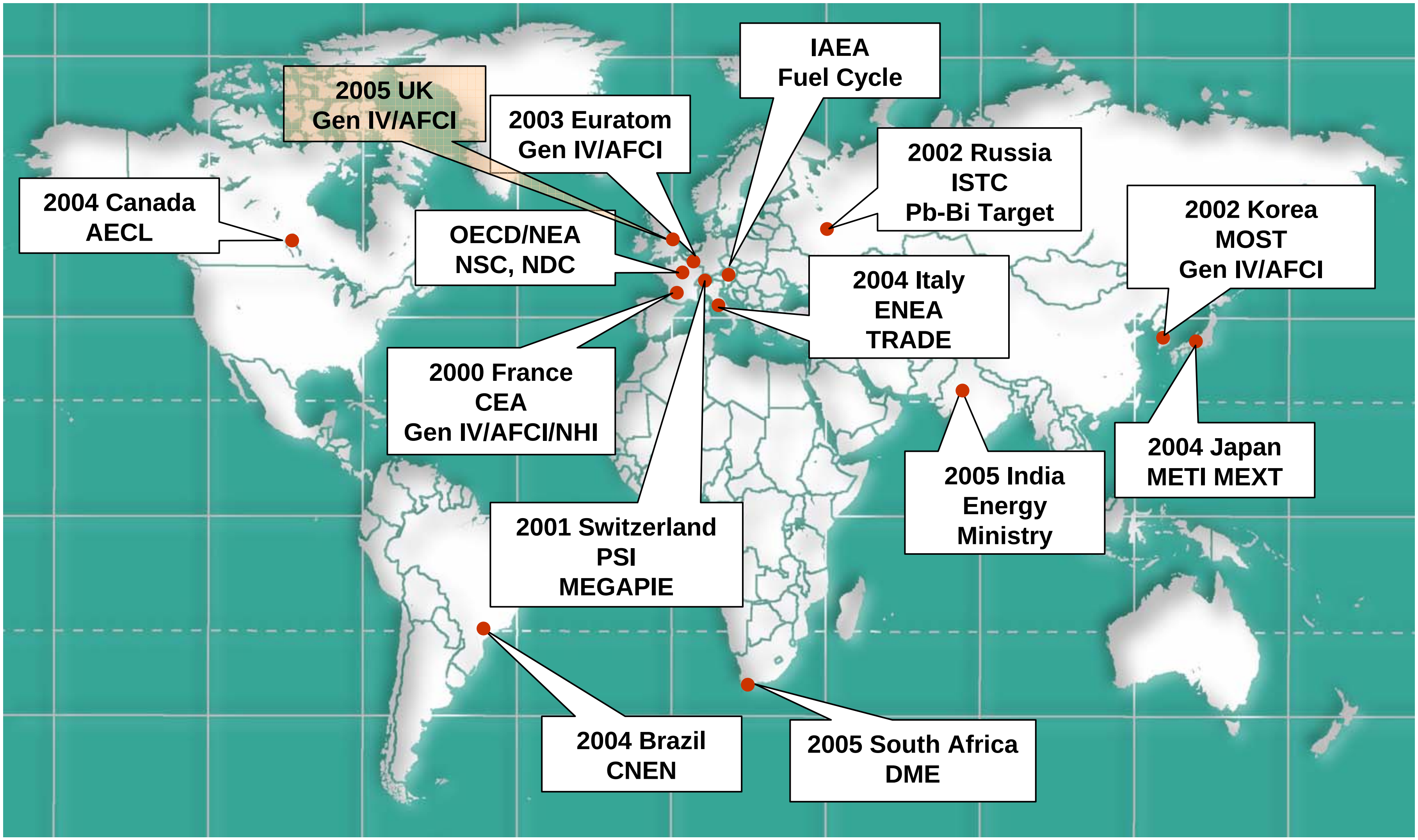


FY 2006 Nuclear Energy Research and Development Budget

(dollars in thousands)

	FY 2005 Current Appropriation	FY 2006 Request	FY 2006 House Mark	FY 2006 Senate Mark
Nuclear Energy Plant Optimization	2,480	0	0	N/A
Nuclear Energy Research Initiative	2,481	0	0	N/A
Nuclear Power 2010	49,605	56,000	46,000	N/A
Generation IV Nuclear Energy Systems Initiative	39,683	45,000	45,000	N/A
Nuclear Hydrogen Initiative	8,929	20,000	20,000	N/A
Advanced Fuel Cycle Initiative	67,462	70,000	75,500	N/A
Total, Research and Development	**170,640**	**191,000**	**186,500**	**N/A**

International Collaborations



ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Andrew Griffith

Program Manager

Office of Advanced Nuclear Research

Office of Nuclear Energy, Science and Technology

FY 2005 Workshop Overview

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Workshop Purpose

- **Familiarize** U.S. Universities with the DOE Office of Nuclear Energy, Science and Technology priorities for 2006-2008:
  - **GEN IV (Generation IV Nuclear Energy System Initiative)**
  - **AFCI (Advanced Fuel Cycle Initiative)**
  - **NHI (Nuclear Hydrogen Initiative)**
- Provide an opportunity for the U.S. Universities to become **directly involved in an integrated teaming relationship** with the DOE and its national laboratories

AGENDA SUMMARY

Thursday

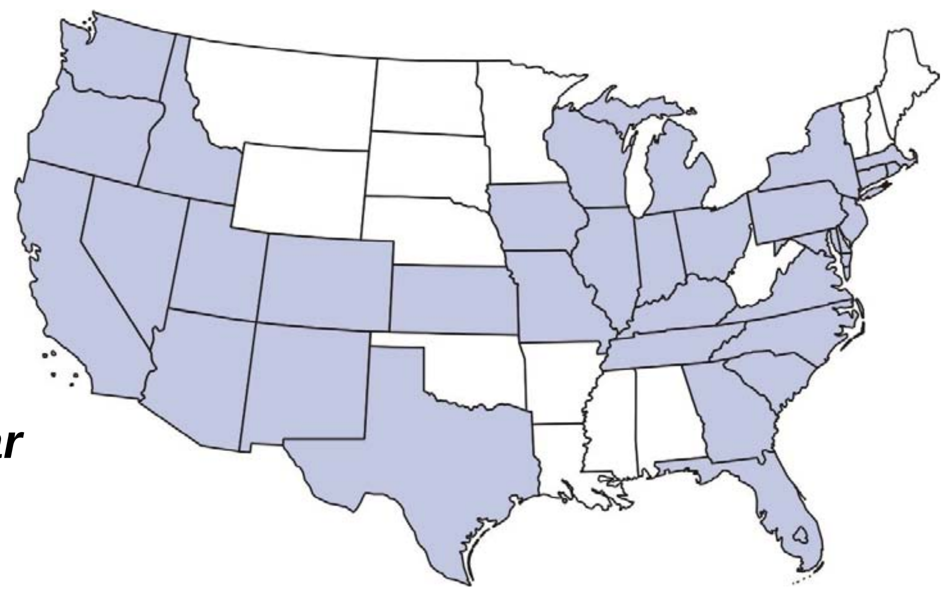
8:00 – 8:45	Overview
8:45 – 9:45	Introduction of 5 Topics
10:15 – 12:00	Introduction of 9 Topics
12:00 – 1:00	LUNCH
1:00 – 3:00	Breakout Session # 1 (7 topics)
3:30 – 5:30	Breakout Session # 2 (7 topics)

Friday

8:00 – 10:00	Breakout Session # 3 (7 topics)
10:30 – 12:30	Breakout Session # 4 (7 topics)

NERI Program Accomplishments
University Projects

- Sponsor R&D to address the principal technical barriers to the future use of nuclear energy
- Help preserve nuclear science and engineering infrastructure within the Nation's universities, laboratories, and industry
- NERI's new focus: meaningful involvement of U.S. universities in NE's principal research programs:
  - ***Generation IV Initiative***
  - ***Advanced Fuel Cycle Initiative***
  - ***Nuclear Hydrogen Initiative***
- FY 2005 Program
  - ***awarded 35 new projects***
  - ***involving 26 U.S. universities***
  - ***$21M in funding over the three-year project duration period***



EXPECTED FUNDING FOR FY 2006 NERI PROGRAM

Advanced Fuel Cycle Initiative (AFCI)	$ 4 million
Generation IV Initiative	$ 4 million
Nuclear Hydrogen Initiative	$ 2 million
Total	$10 million

- ~ $4 million is planned for use toward new FY 2006 NERI Program awards
- ~ $6 million is for “mortgage” work started in FY 2005 and program administration

Solicitation Schedule

♦ Issue Solicitation	June 2005
♦ Applications Due	August 2005
♦ Results Announced	November 2005
♦ Complete Awards	January 2006

SOLICITATION ANNOUNCEMENT

- **Department of Energy's e-Center for Business and Financial Assistance web site**
  - http:e-center.doe.gov
- **Federal Business Opportunities web site**
  - http://www.fedbizopps.gov
- **Office of Nuclear Energy (NE) web site**
  - http://www.nuclear.gov
- **E-mail Distribution to Workshop Participants**

APPLICATION REVIEW AND SELECTION PROCESS

- **Peer Review – Quality of Applications**
- **Program Relevance Review – National Laboratory Program Managers**
- **Management Review by DOE**

Workshop Logistics

- Registration
- Available Documents
- Workshop Proceedings
- Breakout Room Assignments
- Lunch and Breaks
- Restrooms
- Energetics Workshop Support Staff

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Buzz Savage

AFCI Program Director

Office of Nuclear Energy, Science and Technology

Advanced Fuel Cycle Initiative

Workshop for Universities
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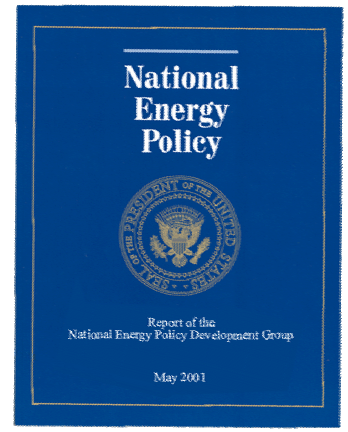
The National Energy Policy and Nuclear Power

"The NEPD Group recommends that the President support the expansion of nuclear energy in the United States as a major component of our national energy policy."

Report of the National Energy Policy Development Group, May 2001



Calvert Cliffs Nuclear Power Plant

Recommendations:

- Support expansion of nuclear energy in the United States
- Develop advanced nuclear fuel cycles and next generation technologies
- Develop advanced reprocessing and fuel treatment technologies

Advanced Fuel Cycle Initiative

♦ Mission

- ***Develop proliferation-resistant spent nuclear fuel treatment, fuel and transmutation technologies to enable the transition from the once-through fuel cycle to a stable, long-term, environmentally, economically, and politically acceptable advanced closed fuel cycle.***

♦ Strategic Goals

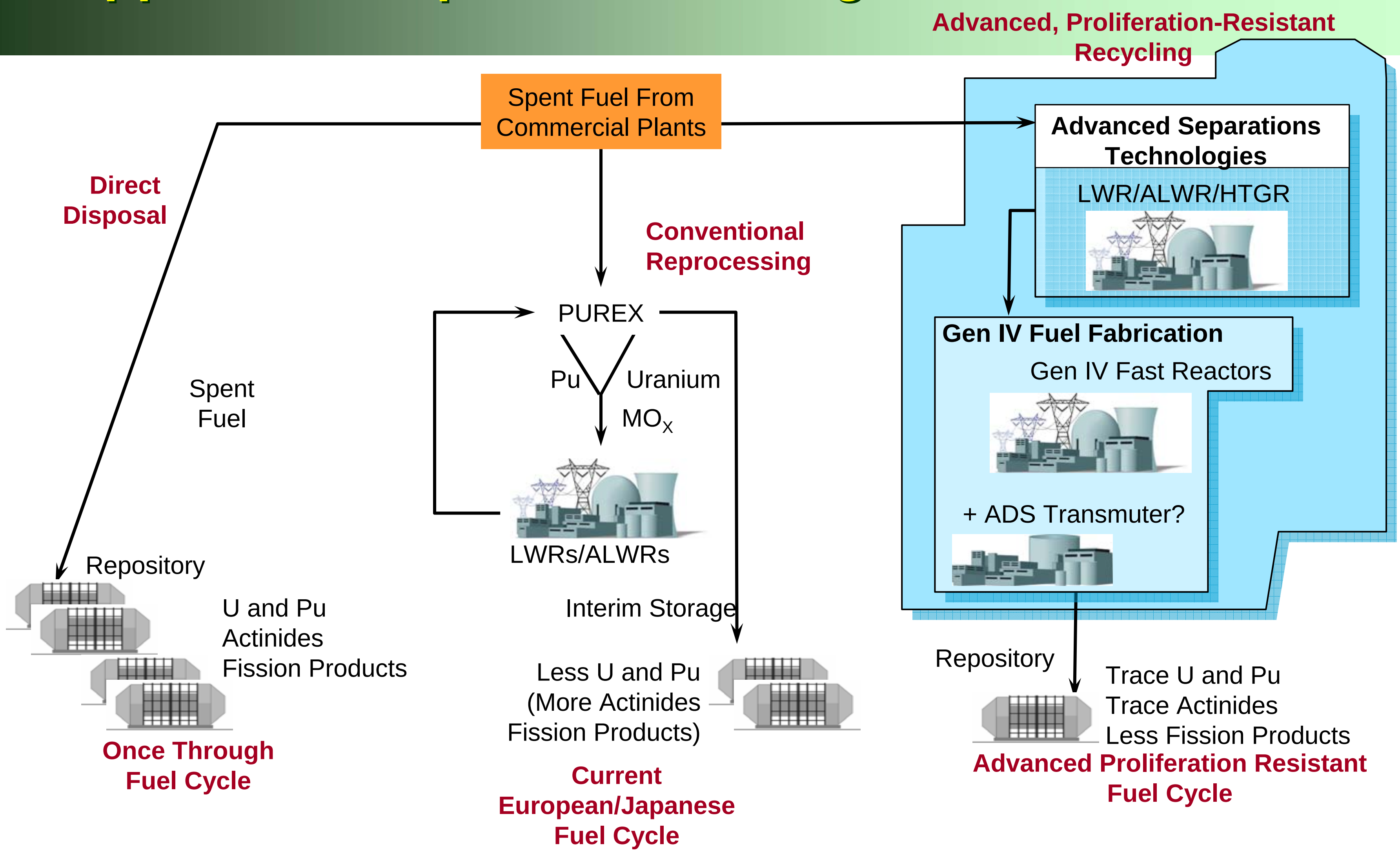
- ***Develop and make available for industry the separations technology needed to deploy by 2025 a commercial-scale spent fuel treatment facility capable of separating transuranics in a proliferation-resistant manner for their recycle and destruction via transmutation reactors***

- ***Develop and make available the separations and fuels technology needed for commercial deployment by 2040 of fast spectrum reactors operating either exclusively as transuranics transmuters or as combined fuel breeders and transmuters***

AFCI Program Objectives

- **Reduce the long-term environmental burden of nuclear energy through more efficient <u>disposal of waste</u> materials**
  - *Remove transuranics (TRU) from waste*
  - *More efficiently utilize permanent disposal space*
  - *Significantly reduce released dose and radiotoxicity*
- **Enhance overall nuclear fuel cycle <u>proliferation resistance</u> via improved technologies for spent fuel management**
  - *Avoid disposal of weapons-usable materials*
  - *Improve inherent barriers and safeguards*
- **Enhance energy security by extracting energy recoverable in spent fuel, avoiding <u>uranium resource</u> limitations**
  - *Extend nuclear fuel supply*
- **Continue competitive fuel cycle <u>economics</u> and excellent <u>safety</u> performance of the entire nuclear fuel cycle system**

AFCI Approach to Spent Fuel Management



AFCI Research Areas

- Advanced aqueous and pyroprocessing spent fuel treatment technologies
- Advanced transmutation and reference fuels for thermal and fast Generation IV reactor systems
- Transmutation Engineering
  - *Physics*
  - *Materials – structural and coolants*
  - *Accelerator-Driven Systems (ADS)*
- Systems Analysis and Modeling

Advanced Fuel Cycle Initiative Budget ($M)

Research Area:	FY 2004 Approp.	FY 2005 Approp.	FY 2006 Request
Separations	$29.1	$23.5	$28.0
Fuels	$13.3	$10.6	$15.6
Transmutation Engineering	$5.4	$11.9	$10.0
Systems Analysis	$4.3	$4.5	$5.0
Education	$7.9	$12.0	$9.0
Other (SBIR, DOE, Tech. Integration, Project Controls)	$8.0	$5.5	$2.4
AFCI Total	$68.0	$68.0	$70.0

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Rob Versluis

Program Director (Acting)
Generation IV Nuclear Energy Systems Initiative
Office of Nuclear Energy, Science and Technology

Generation IV Initiative Overview

Workshop for Universities
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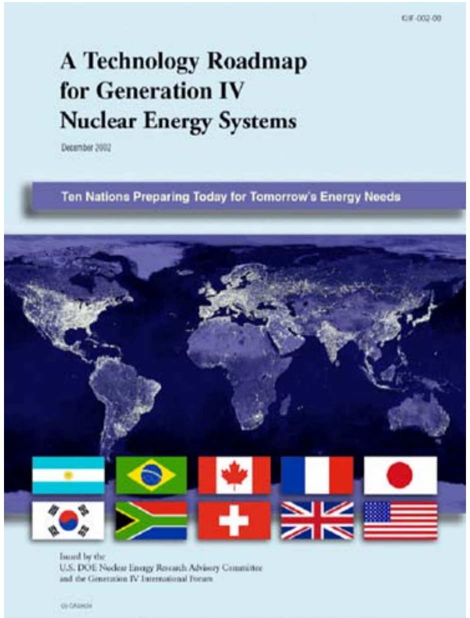
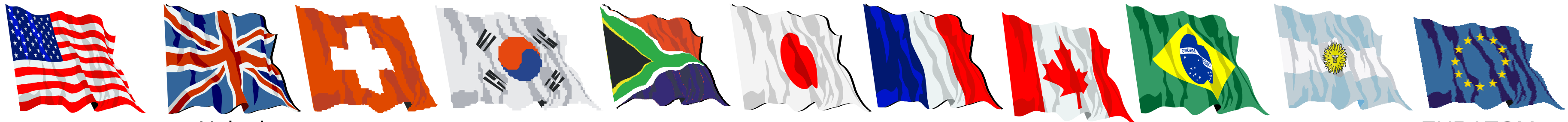
Generation IV International Forum

A Technology Roadmap for Generation IV Nuclear Energy Systems

- In 2000, the Generation IV International Forum (GIF), under NE leadership, started a 2-year Generation IV Roadmap effort:
  - **More than 100 experts from 12 nations**
  - **Six candidate Generation IV systems selected by GIF for further development**
  - **R&D needs for systems, and crosscutting technology identified**
  - **Basis for joint design development and system demonstration**
- In 2002, U.S. initiated bilateral collaborations with many GIF counties under I-NERI arrangements
- In 2005, Canada, France, Japan, U.K., U.S.A signed GIF Framework Agreement, providing basis for multilateral collaboration





U.S.A. | United Kingdom | Switzerland | South Korea | South Africa | Japan | France | Canada | Brazil | Argentina | EURATOM

Generation IV International Forum

National Program Interests

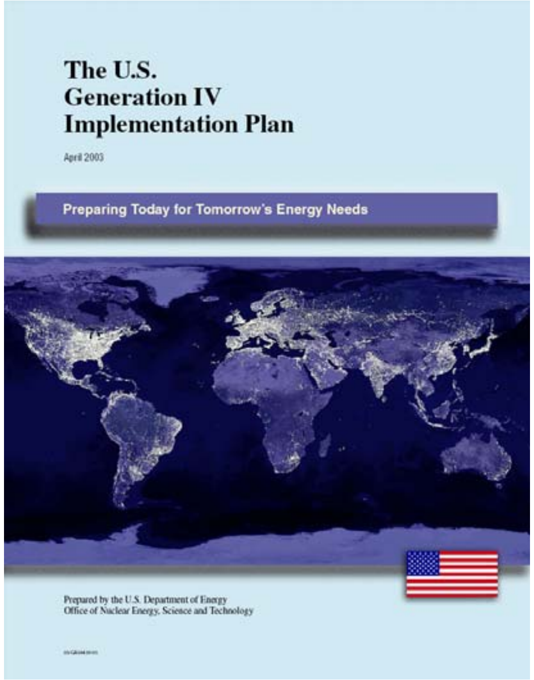
	Argentina	Brazil	Canada	Euratom	France	Japan	Korea	RSA	Switzer-land	UK	USA
VHTR			Participant	Participant	Participant	Participant	Participant	Participant	Participant	Participant	Lead
GFR				Participant	Lead	Participant	Participant	Participant	Participant	Participant	Participant
SFR				Participant	Participant	Lead	Participant			Participant	Participant
SCWR			Lead	Participant		Participant	Participant		Participant		Participant
LFR				Participant		Participant	Participant				Lead
MSR				Participant	Lead						Participant

GFR -- Gas-cooled fast reactor
LFR -- Lead-cooled fast reactor
MSR -- Molten salt reactor
SFR -- Sodium-cooled fast reactor
SCWR -- Supercritical water-cooled reactor
VHTR -- Very high temperature reactor

Lead
Participant

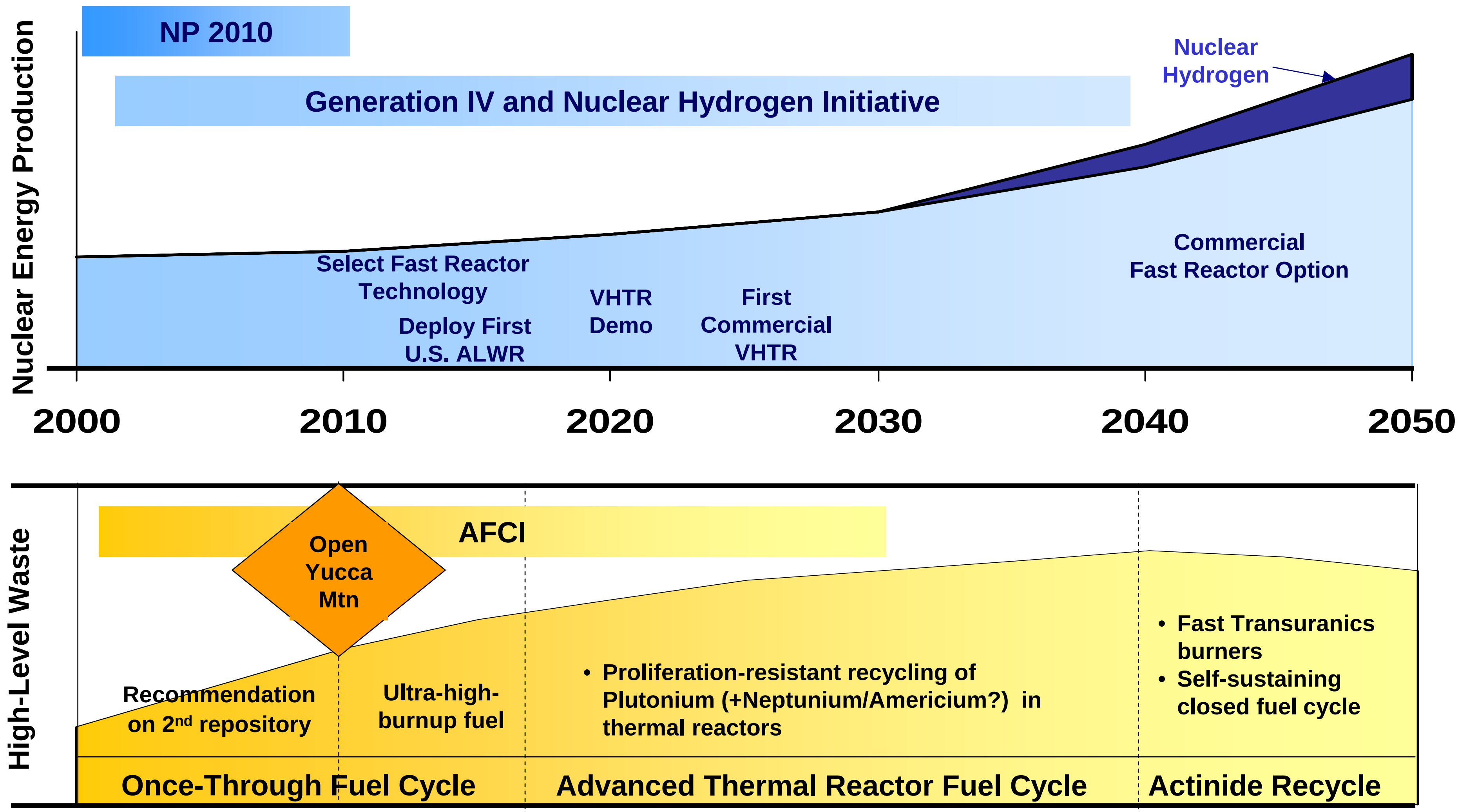
Generation IV Initiative
U.S. Generation IV Program R&D Agenda

- **The Generation IV Technology Roadmap was completed in 2003**
- **The U.S. implementation of the Roadmap recommendations was influenced by:**
  - President's Hydrogen Fuel Initiative announced in 2003
  - Industry recommendations on pursuit of higher efficiency electricity production
  - Market preference for small size, modularity, ease of entrance onto the grid – "gas turbine" model
  - President's National Energy Plan urging exploration of advanced fuel cycles
    - Future full recycle of spent nuclear fuel in Generation IV reactors
    - Options for management and disposition of existing LWR spent nuclear fuel



Generation IV Initiative

Timeline for U.S. Generation IV, AFCI, NHI Objectives



Generation IV Initiative
VHTR Research and Development

- Design concept trade studies for prismatic and pebble bed design with helium and liquid salt coolant leading to design concept selection
- Thermal-hydraulic studies using coolant flow loops and Matched Index of Refraction test facility
- Preliminary component studies, especially intermediate heat exchanger and energy conversion
- Development and qualification of UCO particle fuel
- Selection, development and qualification of structural materials
  - Graphites and composite materials for in-core use
  - Metallic materials for out-of-core use
- ASME codes and standards for VHTR design
- Development and validation of design and safety analytical methods

Generation IV Initiative
Fast Reactor and SCWR Research and Development

- **Development of point design concepts for GFR, LFR, and SCWR, sufficient to establish fuel, core design, materials, components, and secondary-side requirements**
- **Demonstration of fuel, materials, safety concept viability**
- **Development of evaluation methods to assist in down-selection**
- **Most fuel and fuel cycle R&D in AFCI program**

Generation IV Initiative

R&D Program Elements and Funding

WBS	Program Element	FY05	FY06 Request
01	Technical Integration	797	1,423
02	VHTR	25,090	25,000
03	SCWR	850	800
04	GFR	1,050	1,300
05	LFR	1,000	1,300
06	SFR	40	40
07	MSR	40	40
08	Design & Evaluation Methods	800	1,100
09	Materials	1,700	2,000
10	Energy Conversion	600	1,190
11	Systems Analysis	100	100
12	I-NERI	2,839	4,020
13	NERI	2,500	4,000
14	GIF Support	1,431	1,500

THE END

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

A. David Henderson

Office of Advanced Nuclear Research

Office of Nuclear Energy, Science and Technology

Nuclear Hydrogen Initiative Overview

Workshop for Universities
DoubleTree Hotel, Rockville, MD
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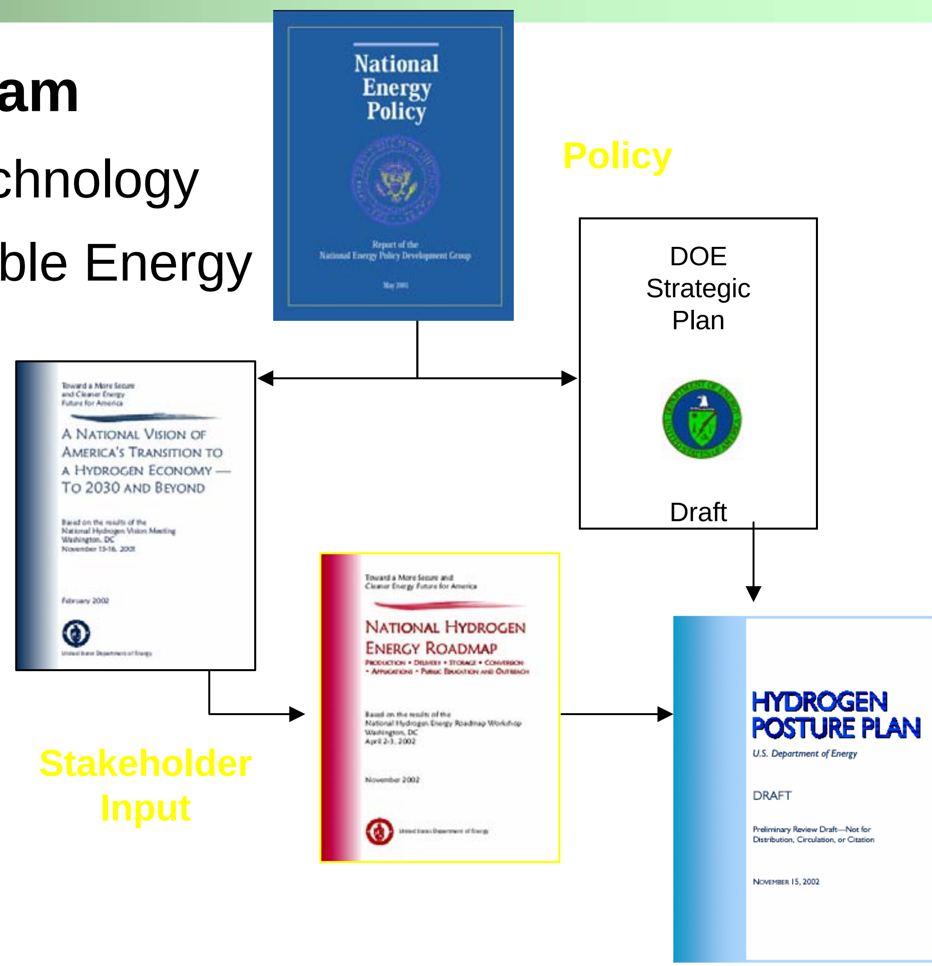
DOE Hydrogen Program

- ♦ **Integrated, inter-office program**
  - Nuclear Energy, Science & Technology
  - Energy Efficiency and Renewable Energy
  - Fossil Energy
  - Science
- ♦ **Hydrogen Posture Plan**
  - Expands upon the National Hydrogen Energy Roadmap to define DOE's role in hydrogen R&D
  - Outlines inter-relationship of offices to accomplish DOE Hydrogen Program goals
  - Presents milestones for hydrogen programs and their integration



Nuclear Hydrogen Initiative

- **Focus:** Hydrogen production technologies that are compatible with nuclear energy systems and do not produce greenhouse gases
- **Objective:** By 2017, operate the nuclear hydrogen production plant to produce hydrogen at a cost competitive with other alternative transportation fuels.
- **Major Program Milestones**
  - FY 2007: Complete the design and construction of laboratory-scale hydrogen production experiments and commence testing
  - FY 2011: Complete the design and construction of pilot-scale hydrogen production experiments and commence testing
  - FY 2017: Demonstrate commercial-scale hydrogen production using heat from a nuclear energy system

Technology Options

- Steam Methane Reforming
- Electrolysis
- Thermochemical Cycles
- High-Temperature Electrolysis

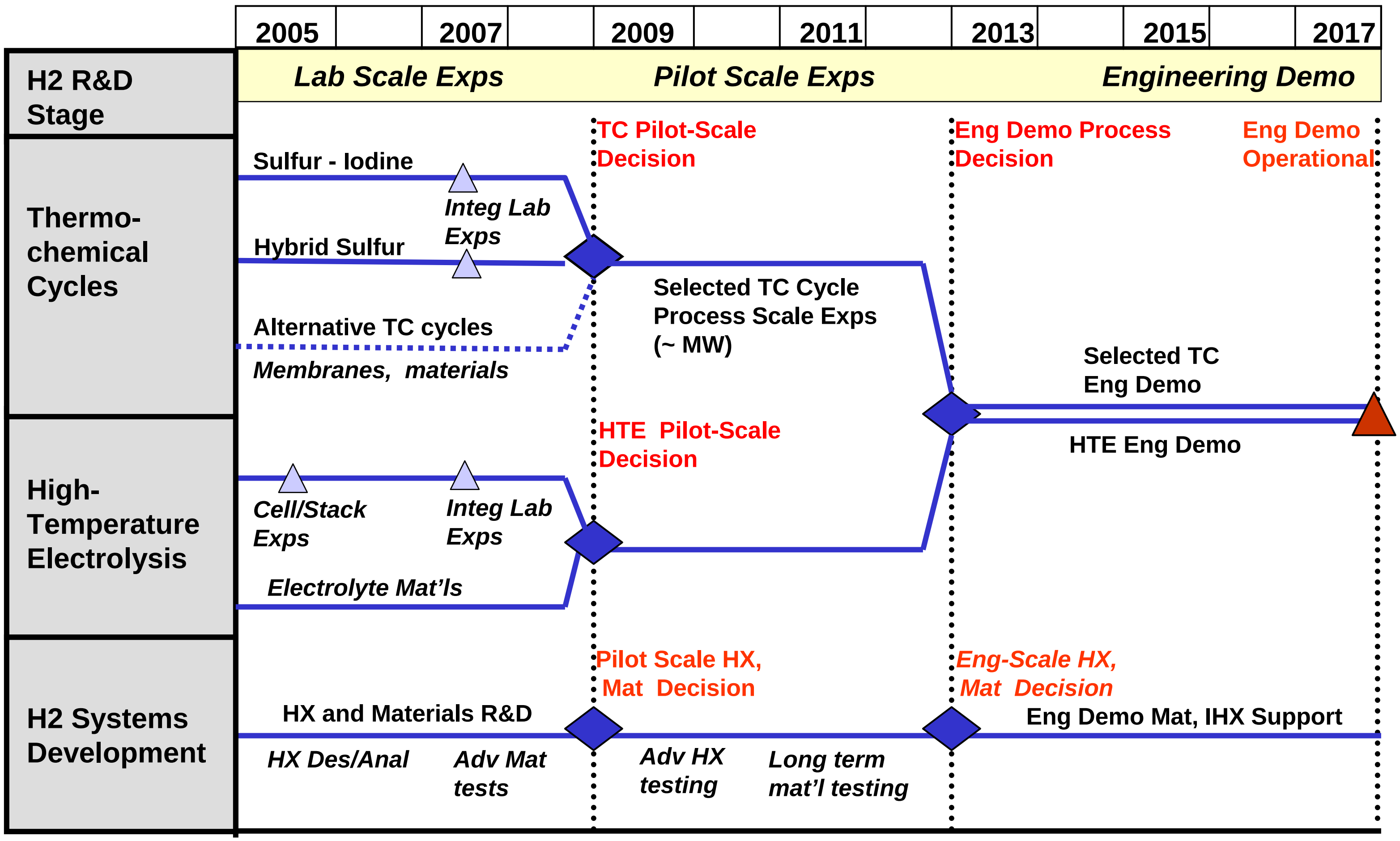
Mature Concepts

- **Steam Methane Reforming**
  - Commercial technology
  - Price subject to natural gas price fluctuations
  - Produces greenhouse gases
- **Electrolysis**
  - Commercial technology
  - Electricity production determines:
    - Efficiency < 25% = product of electricity production (33%) and hydrogen production(< 75%)
    - Emissions - not an issue for nuclear electricity
  - Modular - same attributes as High-Temperature Electrolysis
  - “Distributed energy source”
    - Hydrogen can be produced at site of use instead of central location
    - Reduces transportation requirements/costs

NHI Research Focus

- **Thermochemical Cycles**
  - Best suited process for nuclear
  - Volumetric scaling - cost increases slowly with size
  - High efficiencies - 40-60% theoretical
  - High production volumes
- **High-Temperature Electrolysis**
  - Less extreme materials requirements
  - Similar efficiencies - 40-55% theoretical
  - Modular
    - Scales up by addition of modules
    - Lower investment capital
    - Higher total capital

Research Schedule



Technical Program FY05 Budget

Program Element	FY04 Funding	FY05 Funding	FY06 Request
Technical Integration	380	350	500
Thermochemical	2,400	2,840	9,800
High-Temp Elect.	1,800	1,420	4,700
Support Systems & Systems Interface	605	2,720	3,000
Program Management & Other Costs	1,192	1,599	2,000
Totals	6,377	8,929	20,000

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Dr. Kathryn McCarthy

Advanced Fuel Cycle Initiative Systems Analysis

Idaho National Laboratory

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AFCI Systems Analysis Includes

AFCI Systems Analysis has tasks in:

- **Broad System Studies**
  - AFCI Criteria and Assessment
  - Scenario Evaluations and Trade Studies
  - Repository Benefits
  - Economic Benefits
- **Transmutation Systems Studies**
  - Transmutation Criteria
  - Transmutation Analyses
  - Nonproliferation and Safeguards

Participating Labs are:

ANL, INEEL, BNL, LANL, LLNL, ORNL, PNNL, SNL and SRS

Systems Analysis Objectives

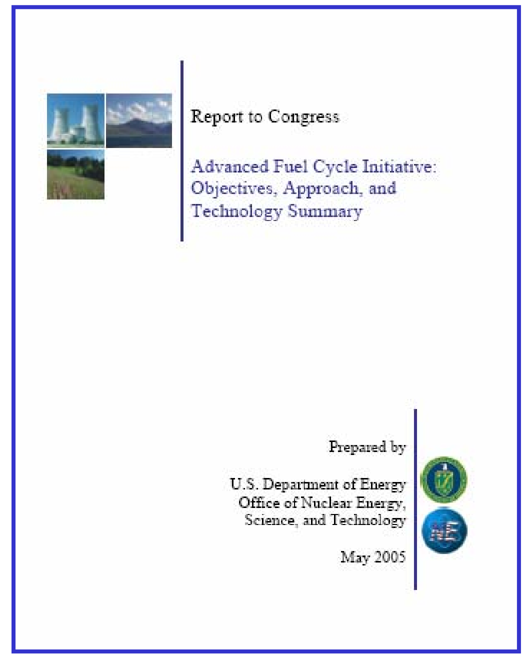
- Develop deployment strategies for the best fuel cycles in the intermediate- and long-term based on environmental, nonproliferation, energy, and economic benefits of advanced fuel cycles, balanced by the understanding of development costs and technology risks
- Assess transmutation approaches and optimize a preferred nuclear fuel cycle for the U.S, including major alternatives and options
- Assess and optimize individual Generation IV systems for the purpose of comparison and technology selection
- Assess performance for specific technology options and facility alternatives that support the program

Major Systems Accomplishments

- Transmutation analyses for a range of fuels and reactor loadings enabling the evaluation of the impact of transmutation approaches on geologic disposal
- Complete fuel cycles analyses for a range of transmutation and separation options to assess spent fuel management strategies
- Gathering of economic data for all fuel cycle process steps to support future defensible life cycle cost analyses
- Dynamic analyses of a range of fuel cycle deployment scenarios to address requests from Congress
- Development of quantitative systems goals and evaluation of their achievability via the above assessments to support congressional requests
- Development of models to support the above activities

Major FY-05 Deliverables

- Prepare the 2005 Report to Congress
- Prepare the 2005 Comparison Report
- Complete first draft of Cost Basis Report
- Prepare report documenting the recommendation on the thermal recycle option
- Prepare white paper on approach to nonproliferation
- Complete first draft of the Technical Options Report (input to Secretarial recommendation)



Report to Congress

Advanced Fuel Cycle Initiative: Objectives, Approach, and Technology Summary

Prepared by

U.S. Department of Energy
Office of Nuclear Energy,
Science, and Technology

May 2005

PLANS FOR FY06-08

- Collaborate with DOE-RW on technical options for future commercial SNF management
- Provide recommendations on fuel types and reactor systems from the standpoint of the overall fuel cycle (continuing)
- Provide analyses as input to recommendation on need for second repository (2007)
- Provide system impact assessments to support reprocessing technology decisions (2007)
- Develop Simulation Institute for Nuclear Energy Modeling and Analysis

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Hussein S. Khalil
Director, Nuclear Engineering Division

Generation IV Design and Evaluation Methods

Argonne National Laboratory

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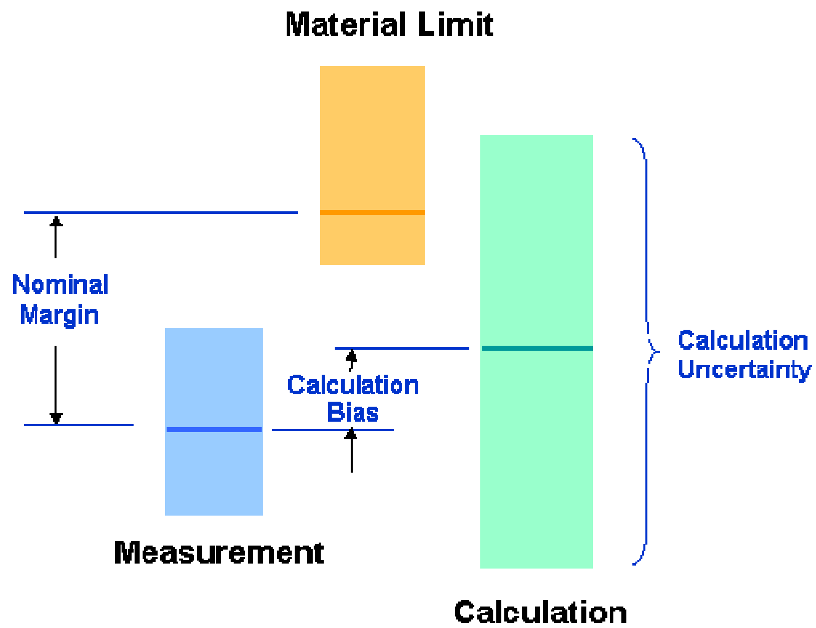
THIS RESEARCH AREA INCLUDES

- **Design and safety analysis capabilities: modeling approaches, computer codes, databases**
  - Specify analytical capabilities needed to design Gen IV systems
  - Assess the adequacy of existing tools
  - Implement and qualify required improvements



- **Methodologies for evaluating system performance against the Gen IV Technology Goals**
  - GIF working groups formed to advance methodologies for
    - Economics
    - Proliferation Resistance & Physical Protection
    - Risk & Safety

FY03-04 ACCOMPLISHMENTS

- **Conducted series of workshops on Gen IV analysis needs and capabilities**
  - Attended by lab, university, and industry representatives
  - Outcome factored into ten-year program plan for Gen IV D&EM
- **Identified, assessed, and prioritized integral physics measurements applicable to validation of VHTR/NGNP physics predictions**
- **Tentatively selected analytical tools for VHTR reactor physics and fuel depletion analysis**
- **Developed first-cut Phenomena Identification and Ranking Table (PIRT) for VHTR design and safety analysis**
- **Participated with CEA in planning of GFR critical experiments that will verify ability to predict GFR neutronic characteristics**

FY03-04 ACCOMPLISHMENTS, cont'd

- **Adapted capital and production cost models for economic evaluations of Gen IV systems**
  - Implemented models in developmental software
  - Initiated applications to ALWR and Gen IV systems

- **Developed and documented initial PR&PP evaluation methodology**
  - Methodology applied at coarse level to an example sodium fast reactor system (ESFR)
  - Workshop for prospective users conducted in November 2004

WORK IN PROGRESS FOR FY05

- **Reactor physics benchmarks**
  - Perform sensitivity studies for high-priority VHTR benchmarks
  - Organize the OECD/NEA International Reactor Physics Benchmark Evaluation Project (IRPhEP)
- **Integral neutronic measurements at CEA facilities**
  - Participate in high-precision measurements of actinide worths in a range of spectra (OSMOSE program)
  - Participate in planning and pre-analysis of GFR integral experiments (ENIGMA program)
- **VHTR T-H and safety analysis tools**
  - Further develop PIRT for a limiting set of transients
  - Perform sensitivity analyses to determine the important parameters and their impact upon key safety criteria
  - Identify measurements relevant to model development and validation

WORK IN PROGRESS FOR FY05, cont'd

- **Economic Evaluation**
  - Perform test calculations for Gen IV systems
  - Refine cost estimation guidelines based on application experience
  - Collect labor and commodity data for European and Asian markets
  - Initiate models for non-electricity energy products and for comparing economics of small and large plant units
- **PR&PP Evaluation**
  - Perform ESFR "demonstration study" as means of further detailing methodology and addressing gaps
  - Initiate software-based Implementation Guide for users
  - Establish interface with AFCI to support application of methodology
- **Risk and Safety Evaluation**
  - First meeting held February 23-24, 2005
  - Defining program of work based on GIF charter

PLANS FOR FY06-08

- **Improve VHTR analysis capabilities and their validation status**
  - Neutronic benchmarks based on measurements in critical facilities and operating reactors
  - Deterministic lattice physics and whole-core analysis tools
  - System code for modeling coupled phenomena in transient and accident scenarios
  - CFD simulation for modeling complex flows (e.g., in outlet plenum)
- **Further define fast reactor and fuel cycle model improvement needs as GFR and LFR designs are specified**
  - Safety confirmation and accident behavior
  - Fuel cycle models for actinide multi-recycle
- **Advance evaluation methodologies through participation in GIF working groups**
  - Support applications aimed at selection of preferred system options

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Kemal O. Pasamehmetoglu

Advanced Fuel Development

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- ADVANCED FUEL DEVELOPMENT FOR LWRs, GEN-IV REACTORS thermal and fast) and DEDICATED TRANSMUTERS (Accelerator Driven Systems or low conversion ratio fast reactor)
- Fabrication, characterization, performance testing, PIE and modeling of TRU bearing Mixed Oxide and Inert Matrix fuels for LWR use
- Fabrication, characterization, performance testing, PIE and modeling of UCO TRISO fuels for VHTR
- Fabrication, characterization, performance testing (in-pile, out-of-pile), PIE and modeling of TRU-bearing nitride, metal, CERCER and CERMET fuels for fast-spectrum systems
  - Including fertile-free and low-fertile versions
- Safety analyses for different advanced fuel types
  - Emphasis on TRU-bearing fuels
  - Transient testing
- Remote fuel fabrication assessment
- Development and selection of advanced clad materials

FY04 MAJOR ACCOMPLISHMENTS

- Completed LWR-1 irradiation at ATR and moved the rodlets to hot cells for PIE
  - WG-MOX, RG-MOX, Np-MOX to ~5% burnup
- Completed the AFC-1 B, C and AE irradiation at ATR and moved the rodlets to hot cells for PIE
  - Fertile-free and low-low fertile (50% Uranium) metal fuel with Pu, Np, Am
  - Fertile-free and low-low fertile (50% Uranium) nitride fuel with Pu, Np, Am
- Completed the low-dose irradiation on GFR fuel matrix materials in ATR
- An International workshop for fuel modeling was held and an International Working Group is established.
- Safety envelope for use of IMF in LWR is assessed and an initial study of Zirconia-Magnesia matrix for LWR IMF is completed.
- TRISO fuel coating process studies are conducted
- Basic characterization capabilities are established for TRISO fuels
- Existing computer models for TRISO fuel is consolidated into an integrated fuel performance code (PARFUME)

WORK IN PROGRESS FOR FY05

- PIE of AFC-1 and LWR-1 rodlets
- Start irradiation of AFC-1G and 1H fuels
  - Fertile-free and low-fertile nitrides to high burnup (~20%)
  - Low-fertile metals to high burnup (~20%)
  - Re-insert AFC-1D (low-fertile metals)
- Fabrication and characterization of the FUTURIX-FTA nitride and metal pellets
- Fabrication of materials samples for FUTURIX-MI irradiation
- Fabrication and characterization of GFR dispersion fuels
- Fabrication and characterization of zirconia - magnesia based IMF
- Fabrication process studies for sphere-pac fuels (emphasis on Am targets in LWRs)
- Oxide fuel modeling studies (thermal properties, oxygen diffusion)
- 3.5 kg of LEUCO kernels fabricated. Characterization started.
- Final design review of AGR-1 test capsule, test train, control and fission product monitoring systems completed
- Coating and compacting process development will be completed and AGR-1 fuel fabrication will start.

PLANS FOR FY06-08

- Complete AFC-1 G & H irradiation in ATR and PIE of AFC-1 B,C,D, AE a
- Complete LWR-2 irradiation in ATR.
- Complete an initial feasibility study for LWR transmutation fuels.
- Complete FUTURIX-FTA and FUTURIX-MI irradiation in Phenix, France
- Prepare samples for JOYO irradiation in Japan
- Assess the initial feasibility for GFR fuel candidates
- Complete fuel-lead-clad compatibility studies for LFR
- Establish a modeling framework for oxide fuels with benchmarks
- Complete the conceptual design of the remote fuel fabrication facility for transmutation fuels
- Publish a handbook on metal, nitride and oxide transmutation fuels for fast spectrum systems
- Complete test capsule and fuel fabrication for AGR-1 tests, start and finish the irradiation
- Complete AGR-3 and 4 capsule and fuel fabrication and start the irradiation

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Bill Corwin
National Technical Director

Gen IV Materials Technology Program

Oak Ridge National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- ♦ **Selection, development, and qualification of structural materials needed to design and build the advanced reactors being developed within the Gen IV Reactor Program**
- ♦ **These activities are part of the Gen IV Reactor Program and are closely coordinated with similar structural materials research for the AFCI and NHI Programs**
- ♦ **Materials needs will be addressed for the NGNP, GFR, SCWR, and LFR reactor systems, as well as for their energy conversion systems**

FY03 and FY04 ACCOMPLISHMENTS

- **Several Assessments of Materials Needs, R&D Plans, and Technology Status Completed**
  - **NGNP, SCWR, LFR, GFR, and NHI Materials**
  - **Crosscutting R&D Plans for Radiation Effects & High-Temperature Materials Experiments and for Development of High-Temperature Structural Design Technology and the Gen IV Materials Handbook**
  - **Modeling and Microstructural Analysis: Needs and Requirements for Generation-IV Fission Reactors**
  - **Impact of High-Performance Computing on Irradiation Modeling**
  - **Generation IV Reactors Integrated Materials Program Plan**

FY03 and FY04 ACCOMPLISHMENTS

Experimental Materials Studies and Codes and Standards Activities Were Initiated

- Irradiation assessment of available nuclear graphites begun for NGNP service
- Initial corrosion testing for SCWR applications was begun and control strategies identified
- Materials were selected and corrosion exposure performed in Pb-Bi for LFR applications
- Joining studies of advanced ODS alloys were begun for GFR applications
- ASTM Standards on Graphite testing and ASME Subcommittee on Graphite for Core Support Applications
- ASME Subgroup on Elevated Temperature Design (NH)

GEN IV MATLS WORK IN PROGRESS FOR FY05

$11,945K, 23 Work Packages, 11 Organizations

- **Crosscutting Materials (ORNL)**
  - **Materials for Radiation Service,High-Temperature Service, and Energy Conversion**
  - **Microstructural Analysis and Modeling**
  - **High-Temperature Design Methodology**
  - **National Materials Program Management**
- **Reactor-Specific Materials Technologies (ORNL, INL, LANL, LLNL, U. of Wisc., U. of Mich., Auburn, MIT)**
- **I-NERIs (ANL, INL, Penn. State)**
  - **Materials for Electrolytic Reduction**
  - **Advanced Corrosion Resistant Zirconium Alloys**
  - **Development and Evaluation of SCWR Materials**

Additional Materials Studies for NHI and AFCI Are in Progress

WORK IN PROGRESS FOR FY05

Crosscutting Materials Work Packages

- **Complete Irradiated Materials Survey and Low-Flux RPV Irradiation Site Selection and Initiate Scoping Irradiations of High-Temperature Alloys in HFIR and Phenix**
- **Initiate Development of *Gen IV Materials Handbook***
- **Update Materials Needs Surveys for Microstructural Modeling and Energy Conversion Systems and Assess Models for ODS Performance and Stability**
- **Develop Preliminary High-Temperature Simplified Design Methods and Initiate Constitutive Equation Development**
- **Upgrade High-Temperature Creep and Creep-Fatigue Testing Facilities**

WORK IN PROGRESS FOR FY05

Reactor-Specific Materials Work Packages

- **NGNP Materials–Selecting & qualifying graphite, high-temp metallic materials & structural composites; improving HTDM; assessing of environmental and thermal aging effects**
- **GFR Materials–ODS materials joining, ion irradiation of ceramics, specimen preparation for high-temperature irradiations in Phenix of ceramics, composites, & refractory alloys**
- **SCWR Materials–Corrosion and SCC testing in SCW**
- **LFR Materials–Corrosion in Pb and Pb-Bi of austenitic & F-M steels, ODS alloys, and BMGs**

MATERIALS CROSSCUTTING PLANS FOR FY06-08

- **Low-dose scoping irradiations of materials and PIE of RPV and reactor internals candidate materials in HFIR**
- **High-dose scoping irradiations of advanced F-M and ODS alloys in Phenix**
- **Establishment of *Gen IV Materials Handbook***
- **Complete initial high-temperature materials scoping studies and codification actions**
- **Modify models for nucleation of extended irradiation-induced defects & their evolution and the behavior of ODS materials; initiate targeted supporting irradiations**
- **Develop initial data and rules for very high-temperature usage of leading Gen IV candidate materials**

REACTOR-SPECIFIC MATLS PLANS FOR FY06-08

- Continue irradiations, high-temperature testing, and environmental and thermal aging effects studies of graphite, metallic materials & structural composites for NGNP
- Develop initial simplified high-temperature design rules and constitutive equations for NGNP materials
- Initiate high-temperature, high-dose irradiations for GFR core support materials
- Continue mechanical properties and corrosion screening experiments for SCWR and LFR

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Mike Cappiello

AFCI: Transmutation

Los Alamos National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Transmutation Science and Engineering

Transmutation Science and Engineering:

- Provides critical research and development to support AFCI transmutation technologies, specifically in the areas of:

  1) transmutation physics,

  2) transmuter materials and coolant technology

  3) accelerator-driven systems (ADS).

- Transmutation is a process by which long-lived radioactive species, particularly actinides, are converted to short-lived nuclides by either neutron fission or capture.

FY04 ACCOMPLISHMENTS

Physics:

- Np-237 differential cross section measurements and analysis
- Am-242g evaluation
- Sensitivity analysis of actinide cross section importance

Materials and Coolants:

- Mechanical Testing of irradiated HT-9 and EP-823.
- Ion irradiations of single crystal Fe with and without helium to validate atomistic models
- Corrosion resistance of Si doped alloys and surface treated materials in Lead-Bismuth
- Modified embedded atom method for irradiation damage model

Accelerator Driven Systems:

- Completion of MUSE accelerator/reactor coupling experiments

WORK IN PROGRESS FOR FY05

Physics:

- Measure Pu-242 fission and capture cross sections
- Evaluate Am-243
- Integrate CINDER into MCNPX

Materials and Coolant Technology:

- Retrieval of MOTA and ACO-3 samples from FFTF
- Fabricate specimens for MATRIX-SMI irradiation in PHENIX
- Molecular dynamics studies of defect formation and migration
- Corrosion testing of amorphous alloys
- Design of Materials Test Station

Accelerator Driven Systems:

- Completion of TRADE 1B experiments and development of RACE project

PLANS FOR FY06-08

FY06:

- Complete data measurements of Pu, Initiate FUTURIX-MI materials irradiation, design engineering scale corrosion test, install MEGAPIE target, develop RACE accelerator, begin area preparation for MTS

FY07:

- Perform data measurements of Am, complete FFTF PIE, perform high temperature lead corrosion test, commission RACE/UT experiment, begin MTS construction

FY08:

- Continue data measurements of actinides, continue MTS construction, analyze MEGAPIE target.

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Paul Pickard

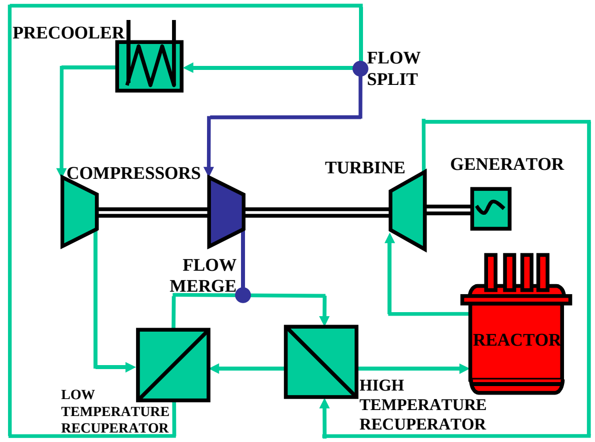
Gen IV Energy Conversion

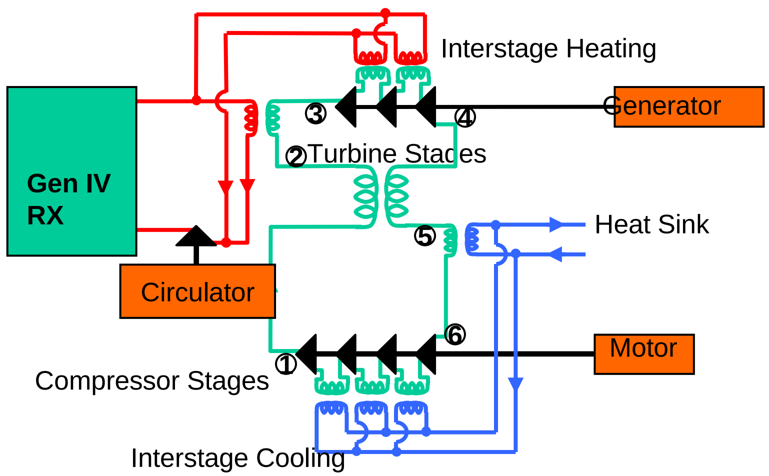
Sandia National Labs

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

GEN IV ENERGY CONVERSION RESEARCH AREA INCLUDES

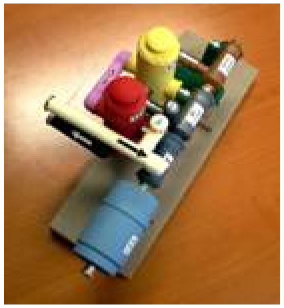
- **Supercritical CO_2 Brayton cycles** for GFR, LFR, SFR, MSR (~550-700 C)
  - *CO_2 turbomachinery design studies*
  - *PCS concept designs*
  - *System controls, scaling studies*
- **High-temperature He Brayton** cycle analysis – VHTR (~1000 C)
  - *Inter-stage heating (IH)/cooling (IC)*
  - *Cost – benefit assessment*
- **Advanced Heat Transport**
  - *Intermediate loop analysis*

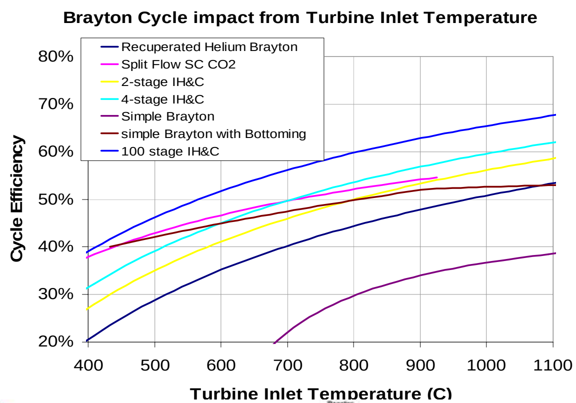


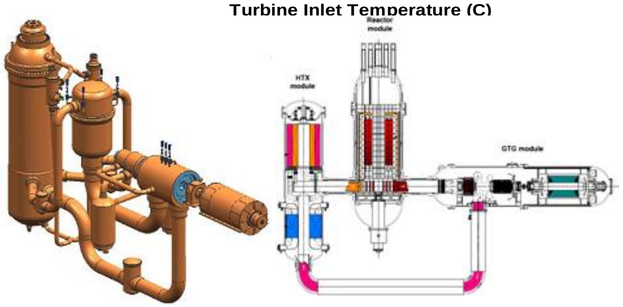


FY04 ACCOMPLISHMENTS

- **Supercritical CO_2 Brayton cycle**
  - Completed initial design studies for turbine and compressors – compact(~1m), efficient (~90+ %)
  - Completed conceptual design for 300 MWe PCS, preliminary cost estimate (~20% less than Rankine)
- **High-Temperature He Brayton cycle**
  - Cost benefit study comparing - combined cycles, IH/IC, S-CO2, alternate working fluids
  - IH/IC effective for He at high temperatures, S-CO_2 at low temperature
- **NGNP Assessment**
  - Completed evaluation of NGNP PCS options (direct/indirect, distributed/integrated, vertical/horizontal)
  - Performance/cost implications of PCS design choices







WORK IN PROGRESS FOR FY05

- **Supercritical CO_2 cycle**
  - Industry assessment of S-CO_2 turbomachinery designs, PCS design approaches
  - Development of dynamic response model for controls studies
  - Scaled experiment design options
- **High-Temperature Brayton cycle options assessment**
  - IH/IC He Brayton design and engineering analysis
  - Preliminary cost - benefit analysis of IH/IC options (reduced sensitivity to component effectiveness)
- **Advanced Heat Transport**
  - Intermediate loop configuration analysis
  - He, liquid salt heat transfer media evaluation

PLANS FOR FY06-08

- Supercritical CO_2 cycle
  - Complete system design, cost assessment for 300 MWe PCS (FY06)
  - Controls analyses – stability, controls approach (FY06)
  - S-CO_2 experiments – small scale validation experiments for key features of S-CO_2 cycle (near critical point compression, controls approach, system stability) (FY06 - FY08)
- High-Temperature He Brayton cycle
  - CBC experiment analyses (model validation using closed Brayton cycle unit, SNL) (FY06)
  - Engineering analysis of IH/IC configurations (FY06)
  - Small-scale experiment design (heat transfer, flow issue) (FY07), fabrication (FY08)
- Advanced Heat Transport
  - Preliminary design for single and multi purpose intermediate loop
  - NHI high temperature systems interface

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

James J. Laidler
National Technical Director

AFCI Separations Technology Development

Argonne National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- Development of aqueous and hybrid aqueous - pyrochemical separations technologies for the processing of spent nuclear fuel from commercial light water reactors
- Development of spent fuel treatment processes for the processing of spent fuel discharged from advanced (Generation IV) nuclear reactors
- Development of improved storage forms for the temporary or permanent storage of low-level and high-level nuclear wastes
- Conceptual design of future spent fuel treatment plants and advanced processing technologies
- Conditioning of spent EBR-II fuel and blankets for disposal

FY04 ACCOMPLISHMENTS

- Conducted a laboratory-scale demonstration of the UREX+2 process with fuel discharged from the Big Rock Point reactor
- Developed an advanced electrorefiner concept for industrial-scale application
- Continued conditioning of EBR-II spent fuel using the electrometallurgical treatment process
- Defined storage concepts for interim storage of UREX+ process products (Am/Cm, Cs/Sr)
- Began testing of advanced process concepts (voloxidation, advanced dissolution methods)

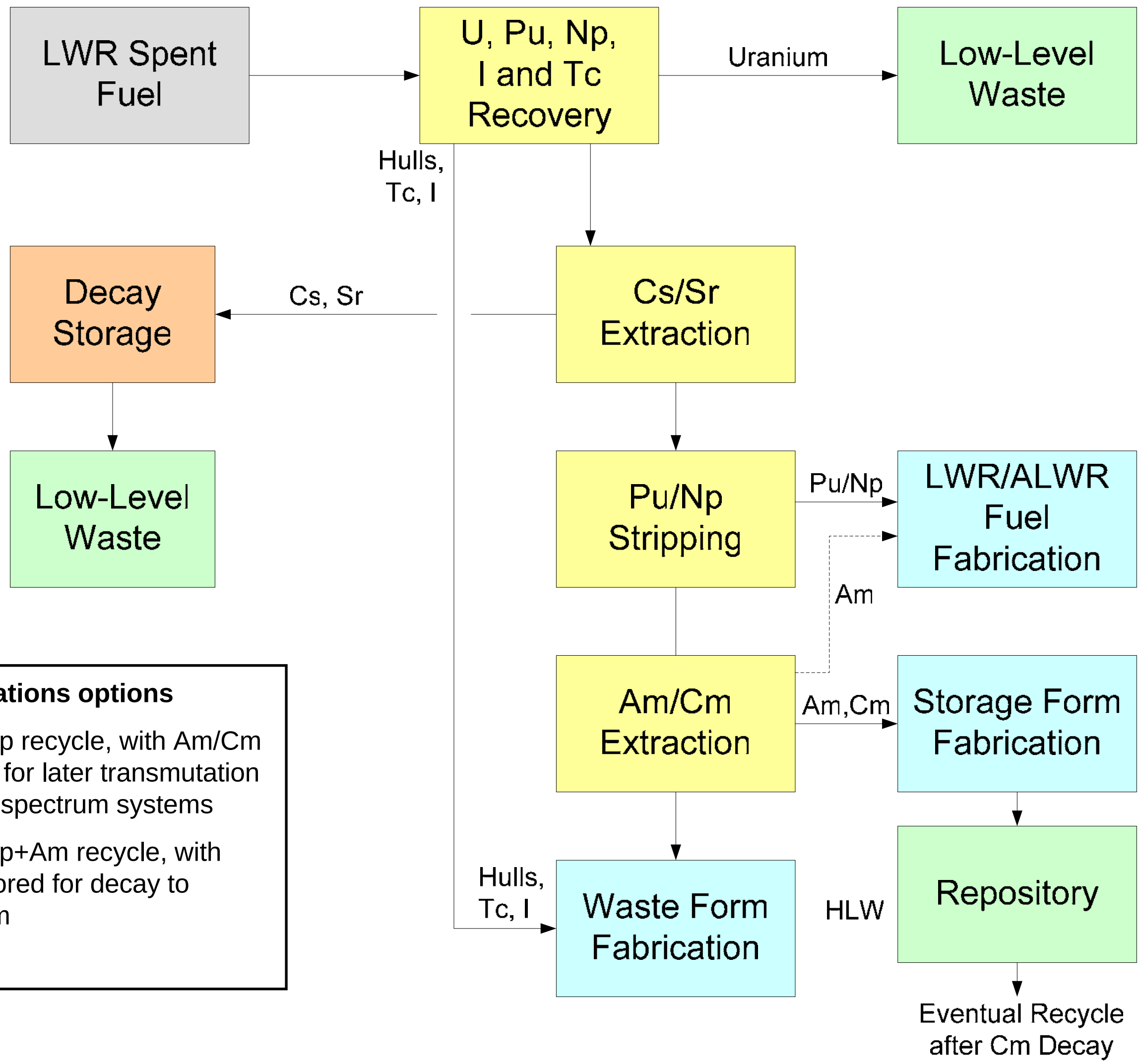
Suite of UREX+ Processes

Process	Prod #1	Prod #2	Prod #3	Prod #4	Prod #5	Prod #6	Prod #7
UREX+1	U	Tc	Cs/Sr	TRU+Ln	FP		
UREX+1a	U	Tc	Cs/Sr	TRU	All FP		
UREX+2	U	Tc	Cs/Sr	Pu+Np	Am+Cm+Ln	FP	
UREX+3	U	Tc	Cs/Sr	Pu+Np	Am+Cm	All FP	
UREX+4	U	Tc	Cs/Sr	Pu+Np	Am	Cm	All FP

Notes: (1) in all cases, iodine is removed as an off-gas from the dissolution process
(2) processes are designed for the generation of no liquid high-level wastes

U: uranium (removed in order to reduce the mass and volume of high-level waste)
Tc: technetium (long-lived fission product, prime contributor to long-term dose at Yucca Mountain)
Cs/Sr: cesium and strontium (primary short-term heat generators; repository impact)
TRU: transuranic elements (Pu: plutonium, Np: neptunium, Am: americium, Cm: curium)
Ln: lanthanide (rare earth) fission products
FP: fission products other than cesium, strontium, technetium, iodine, and the lanthanides

Example: UREX+2 Process, Designed for Thermal Recycle



Separations options

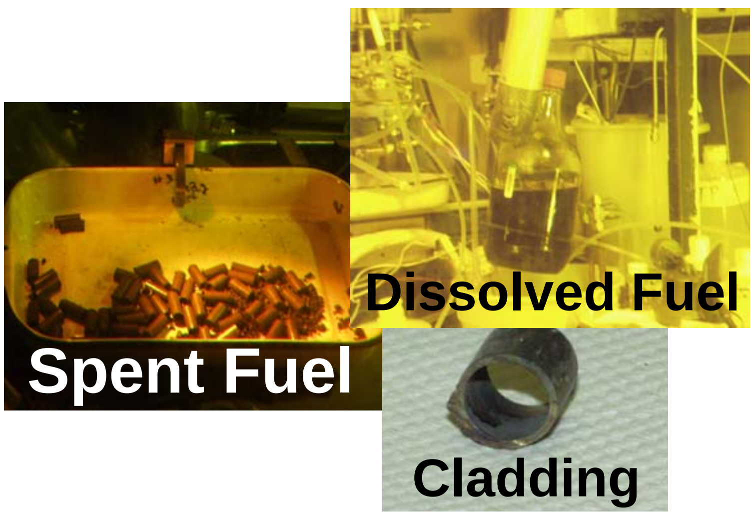
- Pu+Np recycle, with Am/Cm stored for later transmutation in fast spectrum systems
- Pu+Np+Am recycle, with Cm stored for decay to Pu+Am

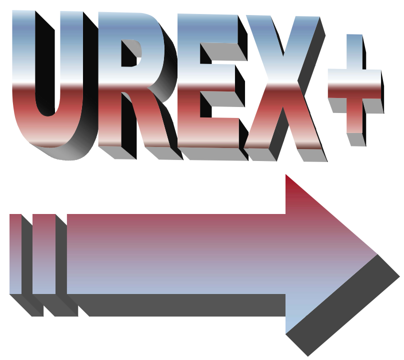
UREX+ Process Equipment

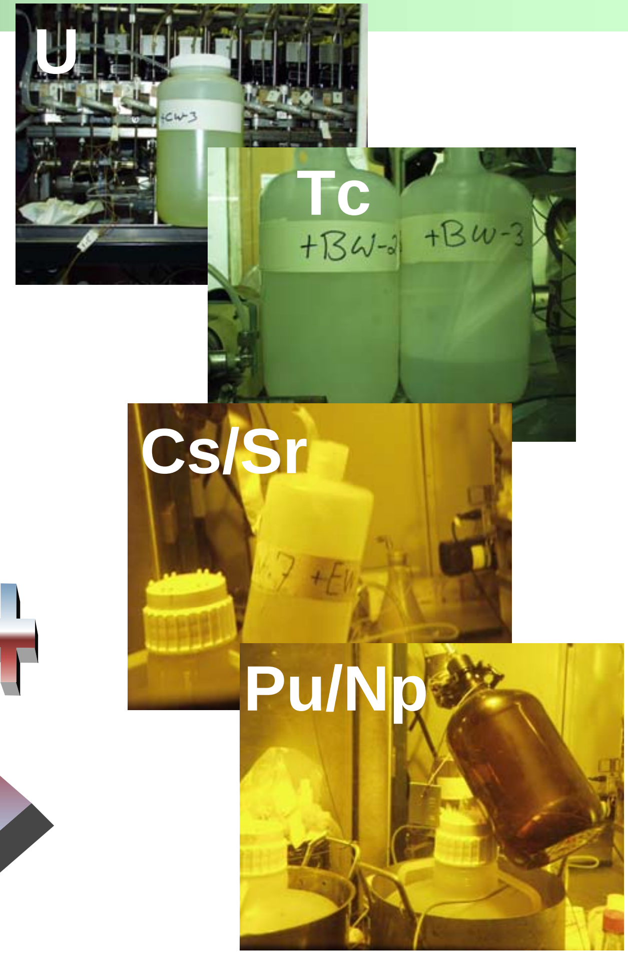
2-cm Centrifugal Contactor before Hot-Cell Placement

UREX+2 Demonstration Results

- **Met process goals**
  - Disposal of uranium as class C LLW
  - Technetium recovery >90%
  - Pu/Np recovery ⩾ 99.5%
  - Disposal of Cs/Sr as LLW
  - Am/Cm recovery ⩾ 99.5%







WORK IN PROGRESS FOR FY05

- **Laboratory-scale hot demonstration of the UREX+1 process (group transuranic extraction flowsheet)**
- **Demonstration of alternative process for Am/Cm separation (in collaboration with the European Union)**
- **Optimization of Cs/Sr extraction process and selection of Cs/Sr storage form**
- **Development of processing concept for treatment of coated-particle fuels**
- **Development of hybrid aqueous/pyrochemical process for LWR spent fuel treatment**
- **Conditioning of EBR-II driver fuel and development of improved methods for blanket processing**
- **Development of advanced head-end operations to minimize process costs**

PLANS FOR FY06-08

- Select reference flowsheet for LWR spent fuel processing
- Demonstrate Am/Cm separations processes and select reference process
- Initiate accelerated EBR-II blanket treatment using advanced processing technology
- Select reference storage/disposal forms for U, Pu/Np, Am/Cm, and Cs/Sr
- Proceed with Advanced Fuel Cycle Laboratory project
- Demonstrate processing of irradiated Gen-IV fuel types at laboratory scale

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Philip E. MacDonald

Very High Temperature Reactor (VHTR)

Idaho National Laboratory

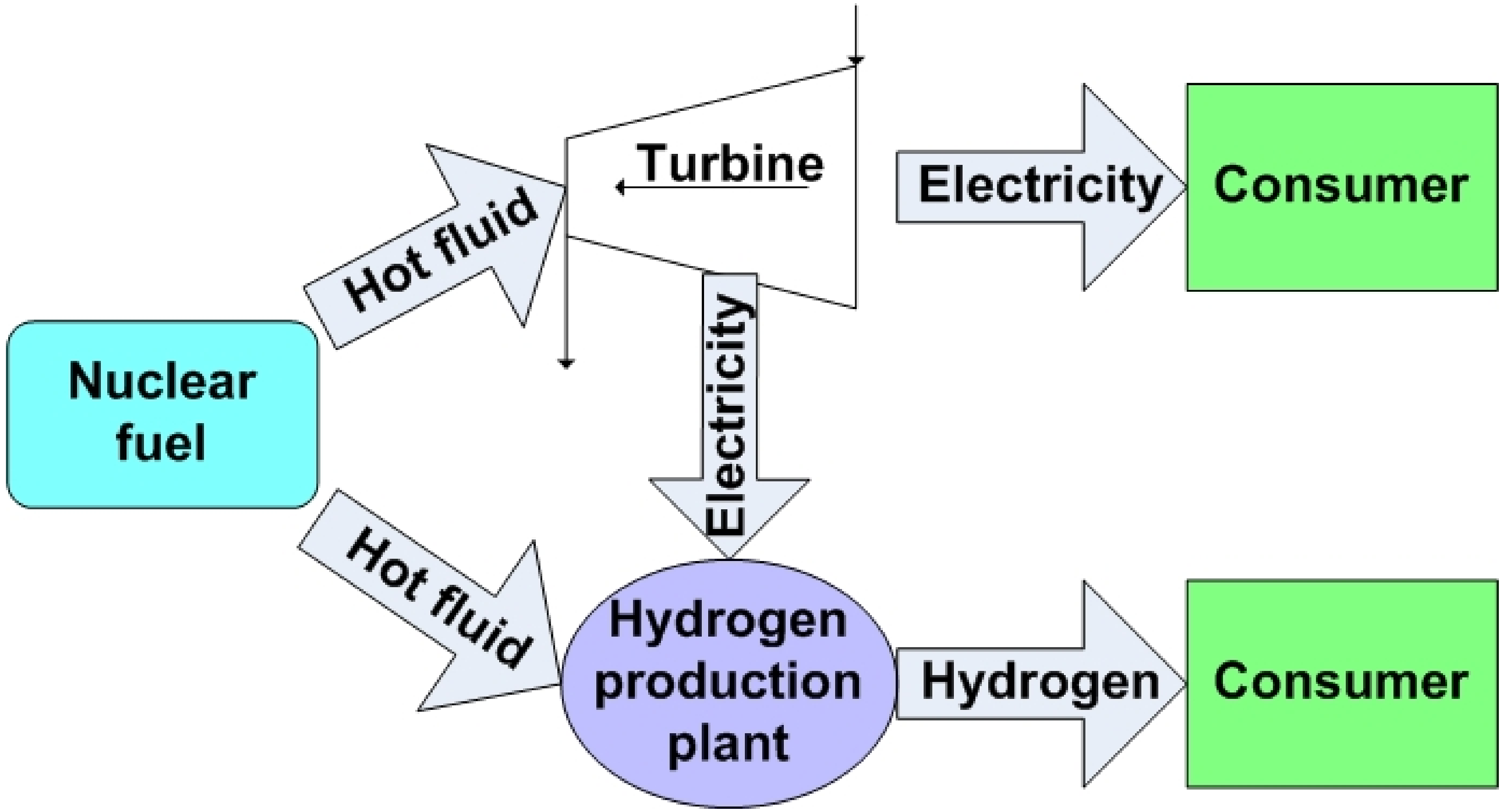
Workshop for Universities
Doubletree Hotel, Rockville, MD
June 16-17, 2005

Very High Temperature Reactor (VHTR) Objectives

- Demonstrate a full-scale prototype VHTR that is commercially licensed by the U.S. Nuclear Regulatory Commission

- Demonstrate safe and economical nuclear production of hydrogen and electricity

VHTR Will Produce Both Electricity and Hydrogen



The High Temperature Gas Reactor is the Current Reference Design

- Utilize inherent characteristics
  - Helium coolant - inert, single phase
  - Refractory coated fuel - high temp capability, low fission product release
  - Graphite moderator - high temp stability, long response times
- Simple modular design:
  - Small unit rating per module
  - Low power density
- Passively safe design:
  - Annular core
  - Large negative temperature coefficient
  - Passive decay heat removal
  - No powered reactor safety systems

Fort St. Vrain Reactor, 1976-1989

This Research Area Includes

- The current VHTR R&D includes work on:
  - Fuels
  - Materials
  - Methods
- There is also hydrogen and BOP electricity production R&D funded separately from the Generation IV program

Materials R&D PROGRESS FOR FY-05

- **Task 1- Nuclear graphite testing and qualification:**
  - Performed site visits of prospective nuclear graphite suppliers and selected the test specimens
  - 16 of 18 HFIR rabbit irradiations of NBG-10 graphite have been completed
  - The ATR graphite compressive creep capsule preliminary design is complete (14 different grades of graphite will be tested)
  - Developed ASTM nuclear graphite fracture toughness standard
  - Nuclear graphite modeling has begun
- **Task 2 -Development of improved high temperature design methodology (HTDM):**
  - Standard chemistry Alloy 617 procured
  - Fusion welds completed and microstructure and properties measured
  - Braze and diffusion welds being made
  - Procurement contract for creep-fatigue environmental test chamber placed
  - ASME Code adequacy with various operating conditions and loading geometries assessed
  - Controlled 617 specification developed
  - Alloy 617 database being developed

Materials R&D PROGRESS FOR FY-05 (cont.)

- **Task 3- Support of ASME code and ASTM standards**
  - ASME draft code currently being prepared for graphite support structures
  - ASTM Nuclear Graphite Materials Specification drafted and submitted for sub-committee ballot
  - ASTM Standards for graphite fracture toughness, XRD, and air oxidation currently in preparation, round robin test program initiated
- **Task 4- Environmental testing and thermal aging :**
  - Low velocity He loop built at INL and 2 loops being refurbished at ORNL
  - Review of helium gas chemistry completed
  - Specimens for creep, tensile, and fatigue testing have been designed
- **Task 5- Reactor pressure vessel materials irradiation facility**
  - Six different potential research reactors were visited and preliminary proposals evaluated
- **Task 6- Composites R&D:**
  - Studies on tube size effects on mechanical properties started
  - Fabrication of tubes and flat plates at Hypertherm and HFIR test capsules underway
  - A high temperature creep testing program for ceramic composites (up to 1700 ^{o}C) has been initiated at the INL

Methods R&D PROGRESS FOR FY-05

- Report documenting CFD validation experimental scaling studies and conceptual designs of lower plenum and heated experiments
- Report summarizing the CFD calculations of the hot channel helium exit temperatures and mixing in the lower plenum
- Report describing development of method for calculating Dancoff factors in doubly heterogeneous media
- Coupling of PEBBED and THERMIX and benchmark calculations
- Implementation of molten salt coolant properties into the RELAP5-3D code to support required liquid salt cooled NGNP safety analyses with flibe, flinak, 92%NaBf4-8%NaF, and 50%NaF-50%ZrF4

Methods R&D PROGRESS FOR FY-05 (cont.)

- Sensitivity and uncertainty analysis has been performed for various core neutronics parameters: k_{eff}, peak power, temperature reactivity effect, and burnup reactivity swing
- CFD validation experiments in the reactor cavity cooling system integral test facility are being developed
- A physics and thermal-hydraulics evaluation of the alternative liquid-salt cooled NGNP is being completed

PLANS FOR FY06-08

VHTR Research Program (to be discussed in the breakout sessions)

- Continued materials R&D including: graphite irradiations and modeling, high temperature metallic alloy testing and design methodology development, ASME and ASTM code support, environmental testing and thermal aging of high temperature metals, RPV materials qualification, and composites development
- Validation of reactor physics and core design analyses tools
- Development and validation of reactor thermal-hydraulic and mechanical design analyses tools
- Safety and risk analyses

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

S. Michael Modro

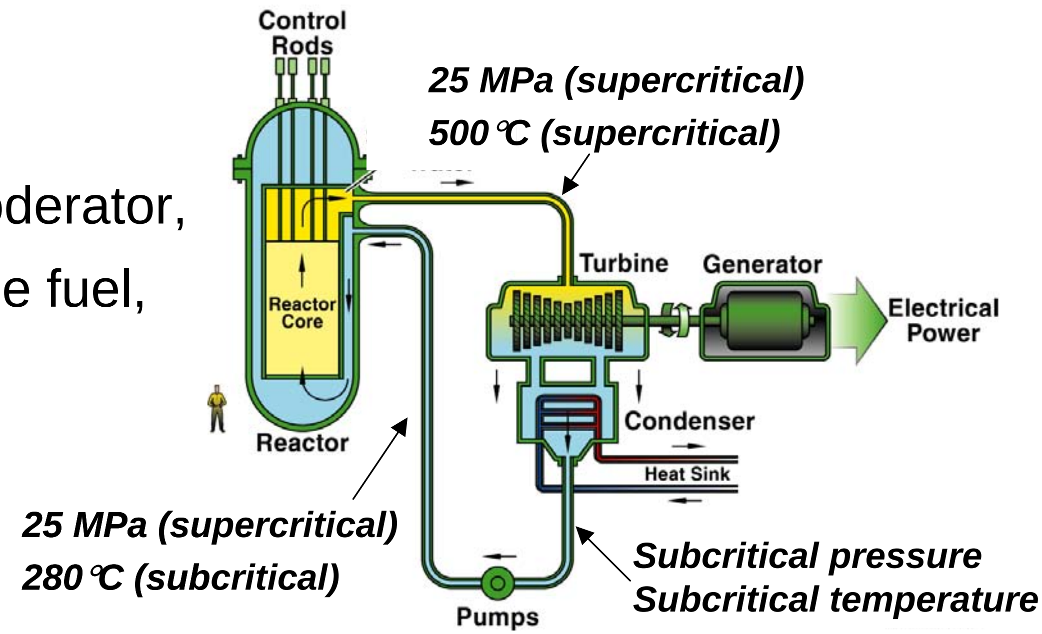
Supercritical Water Cooled Reactor (SCWR)

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Demonstrating technical feasibility of a LWR operating above the critical pressure of water, and producing low-cost electricity.**
- **The U.S. program assumes:**
  - Direct cycle,
  - Thermal spectrum,
  - Light-water coolant and moderator,
  - Low-enriched uranium oxide fuel,
  - Base load operation.



FY04 ACCOMPLISHMENTS

1) **Analyses have shown that the containment designed for SCWR will respond to loss-of-coolant accidents with temperatures and pressures within design limits.**
2) **Solution for safety systems coping with loss-of-feedwater transient and other events with quick core voiding was identified.**
3) **Preliminary subchannel analyses have shown extreme sensitivity of the reference design to hot channel factors leading to unacceptable coolant and cladding temperatures.**
4) **Stability analyses showed that the current SCWR reference design does not satisfy BWR stability criteria at reduced power operation.**
5) **RELAP5 simulations showed that constant pressure start-up procedure yields much more stable conditions that variable pressure start-up.**

FY04 ACCOMPLISHMENTS

6) **500 hours tests were performed for corrosion resistance at 25 ppm and 2 ppb of oxygen on variety of alloy samples.**

7) **A facility for stress corrosion cracking testing was constructed at the University of Michigan.**

8) **Coolant chemistry issues were identified and control strategies proposed.**

- **In summary, the key feasibility issues for the SCWR are the thermal-hydraulic core design and the development of in-core materials.**

WORK IN PROGRESS FOR FY05

5 organizations, $800k

1) **Bundle test section design**
2) **Stability analysis**
3) **Corrosion and SCC testing**
4) **Program management**

PLANS FOR FY05-07

Focus of the program for the next 3 years will be on:

- **Investigation of basic thermal phenomena for the SCWR (e.g., heat transfer, analytical methods, etc.)**
- **Evaluation of dynamic power/flow instabilities**
- **Corrosion and stress-corrosion cracking testing of promising materials for the SCWR core and vessel internals.**

(more details in breakdown sessions)

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Doug Crawford/INL

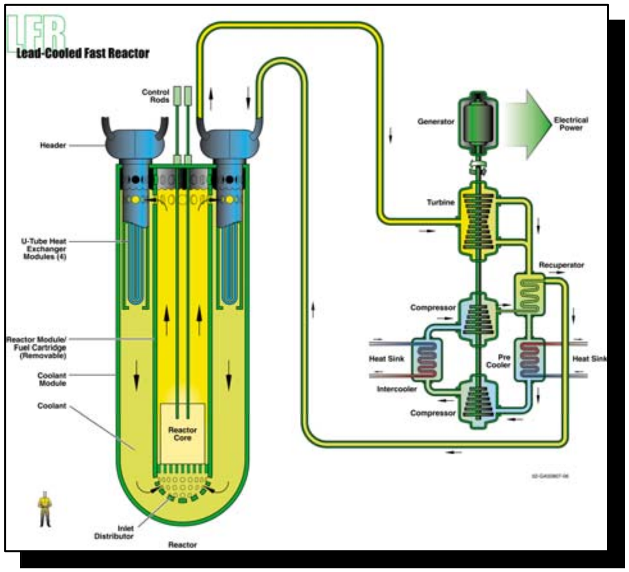
Bill Halsey/LLNL

Lead Cooled Fast Reactor

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Lead-cooled Fast Reactor Research Program

- **R&D objective is a nuclear energy system for deployment in small/remote markets and in developing countries. Low cost, simple initial design enables early LFR technology demonstration and deployment, followed by evolution to larger systems and higher temperatures.**
- **Desired attributes include**
  - Proliferation resistance through long core lifetime with no on-site fuel handling, passive safety, modular factory construction, semi-autonomous load following, simplified operation with small staff.
- **R&D elements are focused on: definition of the reference system design, coolant and materials issues unique to the LFR, evaluation of the safety case and 'license by test' approach, and understanding deployment and institutional issues unique to small transportable systems.**



LFR Research Includes the Following

- **System Design & Evaluation**
  - Long-life core design, near unity conversion ratio
  - Thermal hydraulic design for passive safety, natural circulation, and autonomous load following
- **Materials**
  - Material challenges: Pb/LBE, fast neutron fluence, time/temp
  - Corrosion testing & modeling, radiation damage models, material design
- **Coolant Technology**
  - Instrumentation, testing, modeling: flow, chemistry control, …
  - Thermal & hydraulic properties
- **Institutional and Deployment Issues**
  - Deployment analysis: factory production, transportation, economics
  - Non-proliferation requirements & assessment

FY04 ACCOMPLISHMENTS

- **Reactor Design & Coolant Technology**
  - Design requirements for 10-25 MWe system developed
  - Natural circulation, autonomous feedback design scaled down from larger system to 45 MWth/18 MWe with CO_2 Brayton cycle. Trade-off between burn-up reactivity swing and compensation rod worth points to tight core with large fuel pin diameter, modest flux, nitride fuel.
  - Cartridge core change-out conceptual design.
- **Materials & Coolant Technology**
  - 1000 hr corrosion test in flowing 450C LBE completed (Delta loop).
  - Initial 'LFR Materials & Coolant Technology Plan' drafted.
  - Materials Screening Tests
    - 1000 hr test of MA957 ODS and SiC-SiC to 650°C in lead
- **Institutional & Deployment Issues**
  - Economic factors for factory production/modular installation evaluated for cost/benefit guidance to system design.

WORK IN PROGRESS FOR FY05

- **Preconceptual design:**
  - Development of integrated point design to serve as basis for material, fuel, design sensitivity and systems trade studies
  - Component evaluations: compact steam generators, secondary systems, seismic issues, safety systems, …
- **Materials & Coolant Technology:**
  - Higher temperature DELTA loop testing (520C):
    - Screen new materials (amorphous …), coatings (aluminide...), treatments (laser peening …)
  - Design requirements for Pb/LBE engineering test facility.
- **Deployment, Institutional & International**
  - Evaluate NRC licensing and safety approach developments.
  - Deployment cost/benefit
  - Start-up GIF LFR System Steering Committee

PLANS FOR FY06-08 (milestones)

- **FY06**
  - **Preconceptual design viability evaluations including reactivity control, system, heat transport and emergency heat removal.**
  - **Initiate studies of potential alloy modification, surface treatments and advanced materials for LFR environments.**
  - **Complete design & start construction of Pb Engineering Test Facility.**
- **FY07**
  - **Complete preliminary selection of primary candidate materials for LFR system, including assessment of mechanical and corrosion properties of primary candidate LFR materials.**
  - **Preconceptual design viability evaluations including structural assessment, containment approach and transient/safety analysis.**
- **FY08**
  - **Establish reference cladding design and material specifications.**
  - **Complete construction of Pb Engineering Test Facility & start testing.**
  - **Preconceptual design viability evaluation including core refueling and transport approach and integrated system viability.**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Kevan D. Weaver

Gas-Cooled Fast Reactor (GFR)

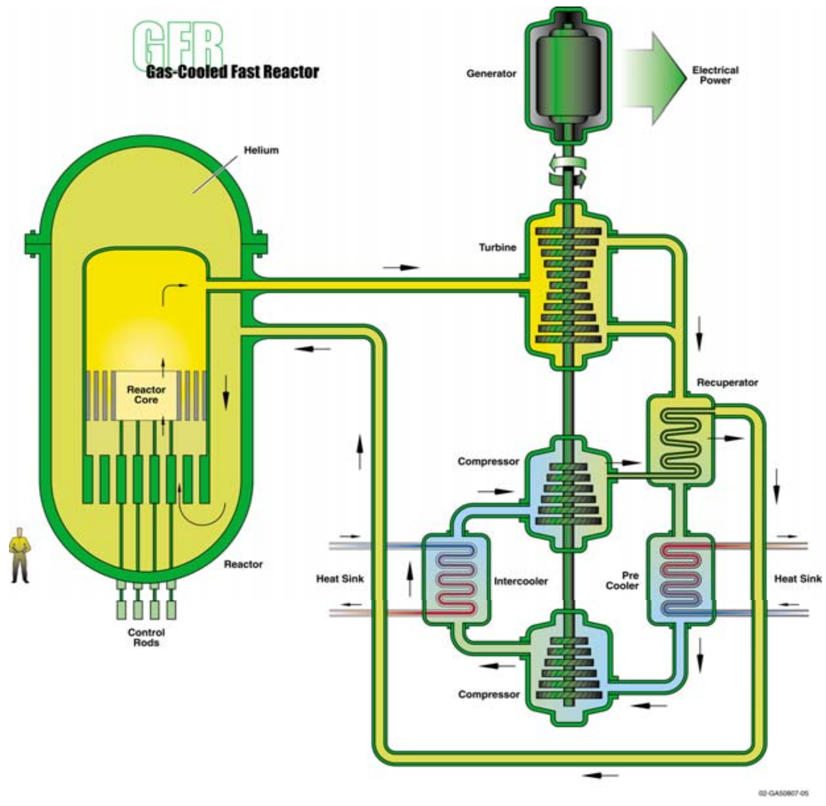
Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **GFR Design and Safety**
  - Define GFR reference design features (fuel technology, coolant, unit power) and operating parameters (power density, temperatures)
  - Identify safety systems capable of effective decay heat removal

- **GFR Fuels, In-Core Materials, and Fuel Cycle Processes**
  - Identify fuels and core materials capable of high temperature operations, excellent fission product confinement, and reasonable burnup/fluence
  - Identify and test fuel treatment and refabrication processes



FY04 ACCOMPLISHMENTS

♦ System Design and Safety

- Documented initial requirements for GFR system design, performance, and safety analysis models

♦ Materials

- Issued GFR material selection and qualification program plan
- CO_2 decomposition studies
  - Completed design of in-pile loop
- Continued ODS joining studies
  - Successfully joined end-caps to pins, and bulk ODS material

♦ Fuels and Fuel Cycle (under AFCI)

- Modeling of actinide bearing fuel
  - Demonstrated that particle fuel can handle ~5% ^{241}Am

WORK IN PROGRESS FOR FY05

8 organizations

- **System Design and Safety**
  - Optimization of candidate safety systems for 600MWt design (combined active/passive)
  - Initial designs of 2400MWt system
- **Materials**
  - Testing (including ion and neutron irradiation) of high temperature candidate ceramics
  - Continuation of ODS joining studies using TLP bonding
- **Energy Conversion (under Energy Conversion)**
  - Development of supercritical-CO_2 power conversion cycle
- **Fuels and Fuel Cycle (under AFCI)**
  - Continued fuel modeling
  - Fabrication of UC particle fuel
  - Irradiation of candidate fuel matrix materials
  - Particle fuel coating
  - Preparation of candidate fuel irradiations

PLANS FOR FY06-08

- Thermal-hydraulic and physics analysis
- Design and testing of safety systems
- Measurement of thermo-mechanical (physical properties) of candidate materials
- Materials irradiation and testing
- Fuel performance modeling, fabrication, and irradiation

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

John H. Kolts

Nuclear Hydrogen Thermochemical Cycles

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Sulfur – Iodine Thermochemical Cycle**
  - Laboratory scale process testing
  - Membrane development
  - Catalyst development
  - Engineering and process optimization
- **Hybrid Sulfur Cycle**
- **Alternative Thermochemical Cycles**
  - Calcium – Bromine Cycle
  - New Cycles

FY04 ACCOMPLISHMENTS

- **Completed design of Integrated Laboratory Scale system for the Sulfur – Iodine process**
- **Completed design and flowsheet analysis for reactive and extractive decomposition of HI**
- **Initiated catalyst studies for the decomposition of SO_3**
- **Completed initial analysis of Calcium – Bromine cycle**

WORK IN PROGRESS FOR FY05

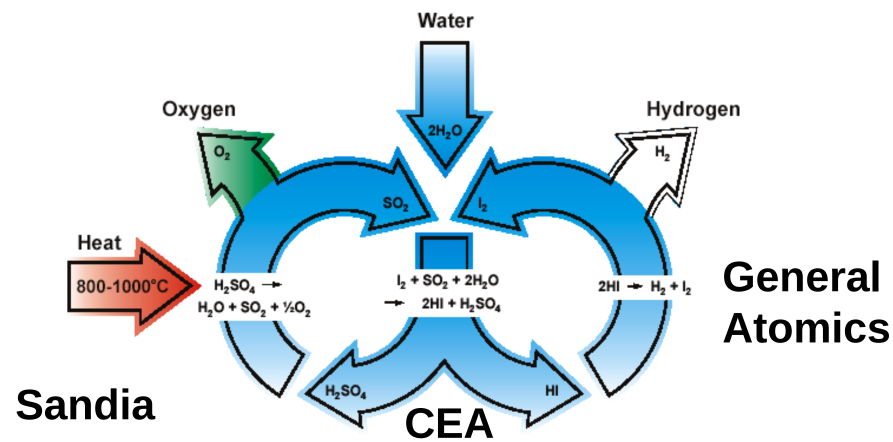
- **Sulfur Iodine Cycle**
  - Fabricated and initiated testing of reactive and extractive decomposition of HI to iodine and hydrogen
  - Fabricated in metal components sulfuric acid decomposition reactor and started testing
  - Continued efforts with CEA, France on fabrication and testing of Bunsen reactor ($SO_2 + I_2 + 2H_2O \rightarrow H_2SO_4 + 2HI$)
  - Initiated comprehensive materials testing program
  - Initiated heat exchanger design, modeling, and testing program
  - Continued catalyst and initiated membrane test programs
- **Hybrid Sulfur Cycle**
  - Completing conceptual design for hybrid sulfur system
  - Conducting ambient pressure testing of H_2SO_3 electrolysis

WORK IN PROGRESS FOR FY05

- **Alternative Thermochemical Cycles**
  - Developing sound criteria for selecting alternative thermochemical processes
  - Potential efficiency
  - Favorable thermodynamics
  - Favorable kinetics
  - Chemistry (completing reaction pathways, phase changes ---)
- **Alternative Cycle Calcium – Bromine**
  - Conducting analysis and design of cold plasma dissociation of HBr
  - Evaluating feasibility of using molten spray contactors to stabilize Ca on support surface

PLANS FOR FY06-08

- Complete construction of integrated laboratory scale S-I loop components.



PLANS FOR FY06-08

- Ship individual S-I laboratory scale loops to INL, assemble and begin testing of fully integrated system
- Complete high temperature, long term, physical property tests for HI and H_2SO_4 compatible materials
- Complete demonstration tests for high temperature membranes to shift SO_3 decomposition equilibrium
- Complete SO_3 decomposition catalyst efforts
- Complete HI decomposition catalyst efforts
- Complete membrane separation efforts for removal of water from iodine solutions
- Conduct optimization tests with single cell electrolyzer for hybrid sulfur cycle

PLANS FOR FY06-08

- Conduct and test lab-scale multi-cell electrolyzer for hybrid sulfur system
- Complete proof of process tests for SO_2/O_2 separation
- Complete preliminary design for S-I and hybrid sulfur cycle
- Complete testing required to determine feasibility and performance of Ca-Br cycle

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Steve Herring

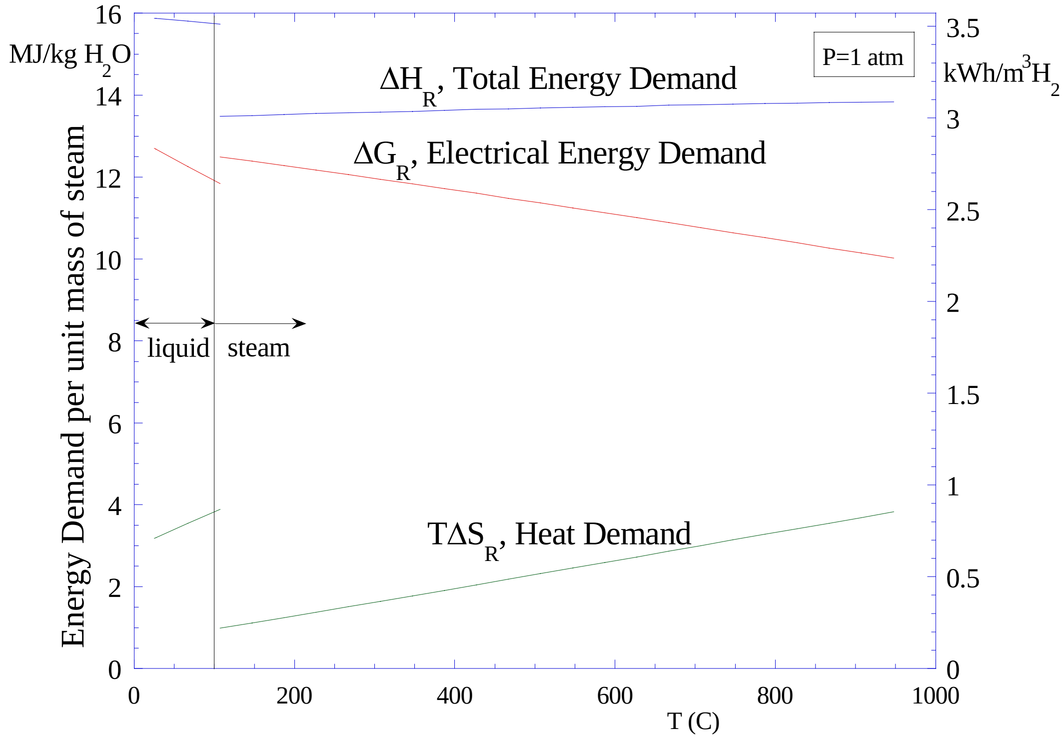
Nuclear Hydrogen High Temperature Electrolysis

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Why use High Temperature Electrolysis?

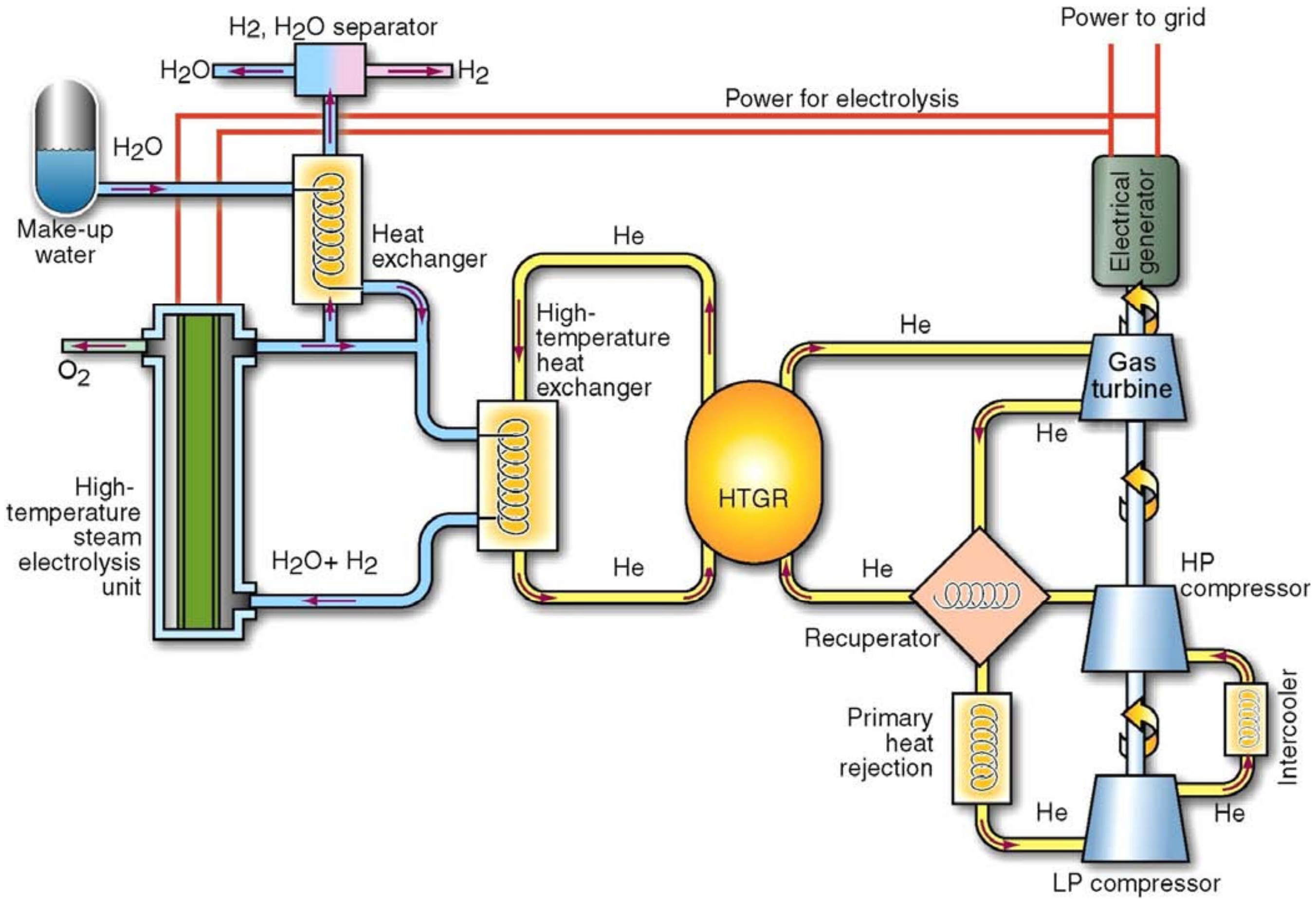
Energy Input to Electrolyser



In addition:

- **Higher electrical generation efficiency**
- **Faster kinetics**
- **Production of co-products (CO/H_2)**

High Temperature Electrolysis Plant



Structure of the Work

1. **HTE System Definition**

   Plant conceptual design [INL]

   CFD and Plant Modeling [INL and ANL]

   Athabasca oil-sand upgrading (I-NERI, INL,ANL with AECL)

2. **HTE Experiments**

   Button Cell and Stack fabrication [Ceramatec, Inc., SLC]

   Advanced electrodes and electrolytes [ANL]

   HTE test stand operation [INL]

   Plasma deposition of cells [INL]

   High temperature H_2/H_2O membrane separations [ORNL]

High Temperature Electrolysis Overview

HTE

Technical Area Objectives (FY05)

- **Develop and demonstrate energy-efficient, high-temperature, solid-oxide electrolysis cells (SOECs) and stacks for hydrogen production from steam**
- **Demonstrate technology at progressively larger scales**
- **Perform flowsheet analyses of systems-level HTE processes to support planned scale-up to Integrated Laboratory Scale, Pilot-Scale and Engineering Demonstration Scale**
- **Develop detailed CFD models of operating SOECs; validate with experimental data**
- **Investigate alternate cell materials (e.g., alternate electrode materials), alternate cell configurations (e.g., porous-metal substrates, Tuff Cell), and applications of inorganic membranes**

High Temperature Electrolysis Overview

Key Milestones (Level 2)

- Demonstrate high-temperature electrolysis stack testing at a production rate of 50 normal liters per hour of hydrogen. [INL, 12/31/04]
- Develop engineering process model for HTE system performance evaluation. [ID, 5/17/05]
- Demonstrate high-temperature electrolysis stack testing at a production rate of 100 normal liters per hour of hydrogen. [INL, 8/1/05]
- Develop conceptual design documentation for the 200 kW high-temperature electrolysis pilot-scale experiment. [INL, 8/15/05]
- Complete Annual Report of CFD and Flowsheet Analyses of High Temperature Electrolysis Plant. [ANL, 9/15/05]
- Complete analyses of membrane applications to High Temperature Electrolysis. [ORNL, 9/1/05]
- Successful fabrication of 15 button cells based on the INL porous-metal substrate design. [INL, 1/15/05]

HTE Research Priorities
Laboratory Scaling Phase (10 Year Plan)

Key technical issues:

- Cell Sealing – For planar electrolysis stacks, edge and manifold sealing is a critical issue, both for stack performance and to enable efficient collection of the hydrogen product.
  - **Glass ceramic seals**
  - **Compression seals (e.g., mica)**
  - **Ceramic pastes**
  - **Significant research has been performed on stack sealing under the DOE SECA program for the fuel cell mode of operation.**
  - **Design studies and laboratory tests are needed to address these issues**
- Interconnections – The use of metallic interconnection between planar cells would result in lower ohmic losses, improved resistance to thermal and mechanical shock, and reduced manufacturing costs.
  - **Metallic interconnections must operate at lower temperatures than ceramic interconnections.**
  - **Chromium mobilization**
  - **Contact resistance (bond layers)**

Key Technical Issues (cont)

- Electrolyte Performance – Methods for increasing electrolyte performance are under investigation
  - **higher ionic conductivity materials with comparable cost are being developed and will be examined for this application**
  - **thin electrolytes with very high ion mobility may be produced using Thermal Spraying and/or Chemical Vapor Deposition (CVD) techniques**
  - **methods for the production larger cells may also reduce the overall cost of cells and stacks**
- Cathode and Anode Materials – Electrodes optimized for larger cells and for more economical production techniques will reduce the capital cost of the electrolyzer.
  - **Graded porosity electrodes**
  - **Thermal spray techniques**
- Materials costs – The use lower cost materials and the use of reduced amounts of intrinsically costly materials will reduce the overall capital cost of the cells

Milestones
HTE Systems Analysis and Experiments

FY2005

- Complete HTE conceptual design and system cost assessment
- Define HTE cell/module options, develop cell / module test plan for FY05-07
- Develop engineering model for HTE system performance evaluation
- Complete button cell experiments
- Continue stack experiments

FY2006

- Design HTE integrated laboratory-scale experiments
- Develop conceptual HTE pilot-scale experiment design
- Construct stack /module arrays for integrated laboratory-scale experiments
- Develop conceptual pilot scale module design

FY2007

- Begin HTE integrated lab-scale experimental operations
- HTE Pilot-scale experiment preliminary design
- Complete HTE cell testing
- Conduct HTE stack / module tests
- Candidate pilot scale module tests

FY2008

- Pilot-scale experiment final design
- Complete HTE integrated lab-scale experimental operations
- Implement cell/module technology improvements

FY2009

- Pilot scale experiment decision

Major Issues in HTE Materials Needs

- **Cost of materials and cell fabrication**
- **Lifetime of the module**
  - Performance – lifetime tradeoff
  - Limiting number of thermal cycles/transients
- **Uniformity and quality of cell manufacturing**
- **Maximum temperature of interconnects**
- **Sealing, especially in planar configuration**
- **Manufacture of thin electrolytes**
- **Matching coefficients of thermal expansion**
- **Shrinkage during manufacture**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Steven R. Sherman, Ph.D.

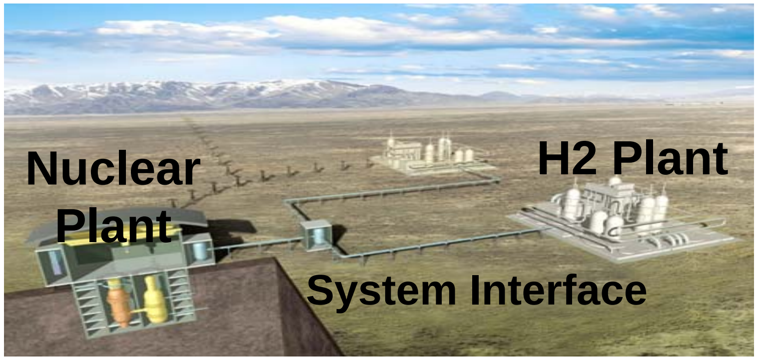
Reactor-Hydrogen Production Process Interface

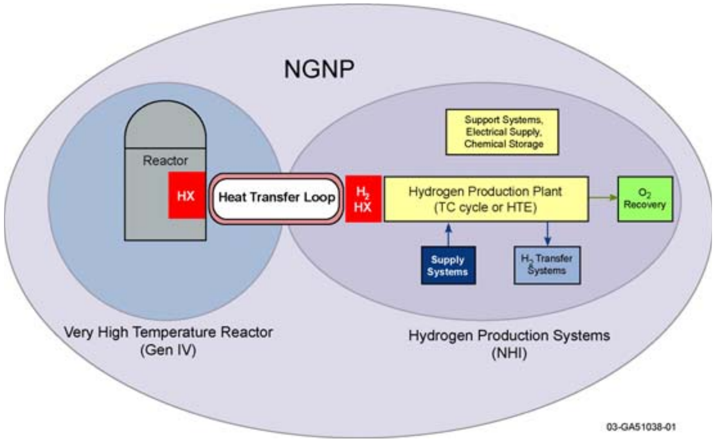
Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Development of high-temperature heat transfer network to enable linkage of nuclear plant to hydrogen production plant**
  - Materials (structural and fluids)
  - Heat exchanger design and development
  - Modeling and simulation
  - Experimental testing
- **Development of hydrogen plant ancillary systems and infrastructure for pilot-scale and engineering-scale nuclear hydrogen production plants**





FY04 ACCOMPLISHMENTS

- **Defined project scope of system interface area**
  - Defined initial infrastructure requirements for pilot-scale hydrogen plant (thermochemical or HTE)
  - Defined initial balance-of-plant requirements for hydrogen plant
  - Defined technical issues and barriers for high-temperature system interface
- **Developed 2-D and 3-D FLUENT models of compact heat exchanger that used helium and molten salt**
- **Performed materials testing and corrosion testing on high temperature alloys (e.g., Waspaloy, Inconel 617)**
- **Initiated work on C-C/Si-C composites and other non-metallic materials for high-temperature heat exchangers**
- **In summary, FY04 work indicated that biggest challenges lie with the system interface, and most resources in this area must be focused on materials, heat exchanger development, and modeling**

WORK IN PROGRESS FOR FY05

- **Funding**
  - Direct, $790K
  - Indirect (through UNLV RF), $1.9M
- **Project areas**
  - Nuclear plant/hydrogen plant spacing requirements
  - Thermal-hydraulic study of heat transfer fluids and effects on system interface configuration
  - Study of individual high-temperature heat exchanger functional and material requirements
  - High-temperature metallic materials mechanical properties measurements
  - Corrosion behavior of materials in S-I process
  - Studies of non-metallic heat exchanger materials and heat exchanger manufacturing techniques
  - Continuation of heat exchanger modeling efforts

PLANS FOR FY06-08

Focus of the program for the next 3 years will be on:

- **Determine candidate materials of construction and heat transfer fluid(s) for system interface components**
- **Study nuclear reactor-hydrogen process isolation methods and equipment**
- **Complete baseline designs for system interface and high temperature heat exchangers**
- **Perform lab-scale testing of components and heat exchangers**
- **Development of integrated simulation of system interface and hydrogen plant**
- **Determine baseline balance-of-plant components and configurations for proposed thermochemical and HTE plants**
- **Initiate permitting activities for pilot-scale plant**
- **Complete assessment of applicable codes and standards**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Dr. Kathryn McCarthy

Advanced Fuel Cycle Initiative Systems Analysis

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

AFCI Systems Analysis Includes

AFCI Systems Analysis has tasks in:

- **Broad System Studies**
  - AFCI Criteria and Assessment
  - Scenario Evaluations and Trade Studies
  - Repository Benefits
  - Economic Benefits
- **Transmutation Systems Studies**
  - Transmutation Criteria
  - Transmutation Analyses
  - Nonproliferation and Safeguards

Participating Labs are:

ANL, INEEL, BNL, LANL, LLNL, ORNL, PNNL, SNL and SRS

Relationship to NE Program Priorities

Systems Analysis is an integrating activity

- Combines technical options to address overall program goals

♦ Global Studies

- Development of fuel cycle strategies and objectives
- Dynamic studies of fuel cycle scenarios

♦ Transmutation Options

- Actinide management via thermal and fast reactor recycle

♦ Benefits Studies

- Repository – Heat load and source term reduction
- Economic – Systematic fuel cycle cost analyses
- Non-proliferation – Intrinsic features of fuel cycle approaches

Systems Analysis Objectives

- Develop deployment strategies for the best fuel cycles in the intermediate- and long-term based on environmental, nonproliferation, energy, and economic benefits of advanced fuel cycles, balanced by the understanding of development costs and technology risks

- Assess transmutation approaches and optimize a preferred nuclear fuel cycle for the U.S, including major alternatives and options

- Assess and optimize individual Generation IV systems for the purpose of comparison and technology selection

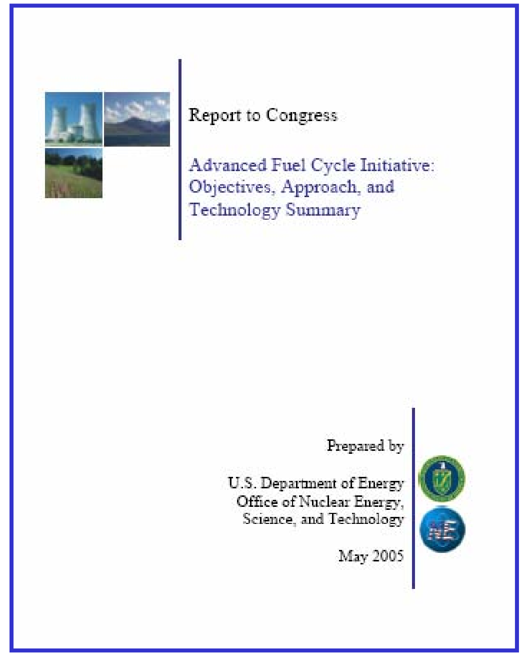
- Assess performance for specific technology options and facility alternatives that support the program

Major Systems Accomplishments

- Transmutation analyses for a range of fuels and reactor loadings enabling the evaluation of the impact of transmutation approaches on geologic disposal
- Complete fuel cycles analyses for a range of transmutation and separation options to assess spent fuel management strategies
- Gathering of economic data for all fuel cycle process steps to support future defensible life cycle cost analyses
- Dynamic analyses of a range of fuel cycle deployment scenarios to address requests from Congress
- Development of quantitative systems goals and evaluation of their achievability via the above assessments to support congressional requests
- Development of models to support the above activities

Major FY-05 Deliverables

- Prepare the 2005 Report to Congress
- Prepare the 2005 Comparison Report
- Complete first draft of Cost Basis Report
- Prepare report documenting the recommendation on the thermal recycle option
- Prepare white paper on approach to nonproliferation
- Complete first draft of the Technical Options Report (input to Secretarial recommendation)



Report to Congress

Advanced Fuel Cycle Initiative: Objectives, Approach, and Technology Summary

Prepared by

U.S. Department of Energy
Office of Nuclear Energy, Science, and Technology

May 2005

AFCI Criteria and Assessment

- **Develop AFCI content and process criteria, assess and report progress.**
- **Content development -**
  - Assist integration of AFCI and Gen IV goals and develop associated AFCI criteria and metrics.
  - Define the potential ranges of nuclear energy demand, reactor types, fuel types, separations options, transmutation options, and waste disposal options.
- **Process development -**
  - Identify U.S. nuclear fuel cycle development pathway, including activity timeframes and decision points.
- **Assessment and reporting -**
  - Consolidate AFCI information and produce annual Comparison Report.
  - Develop the AFCI 2005 Report to Congress and organize the 2007 Secretarial repository recommendation report.

AFCI Objectives

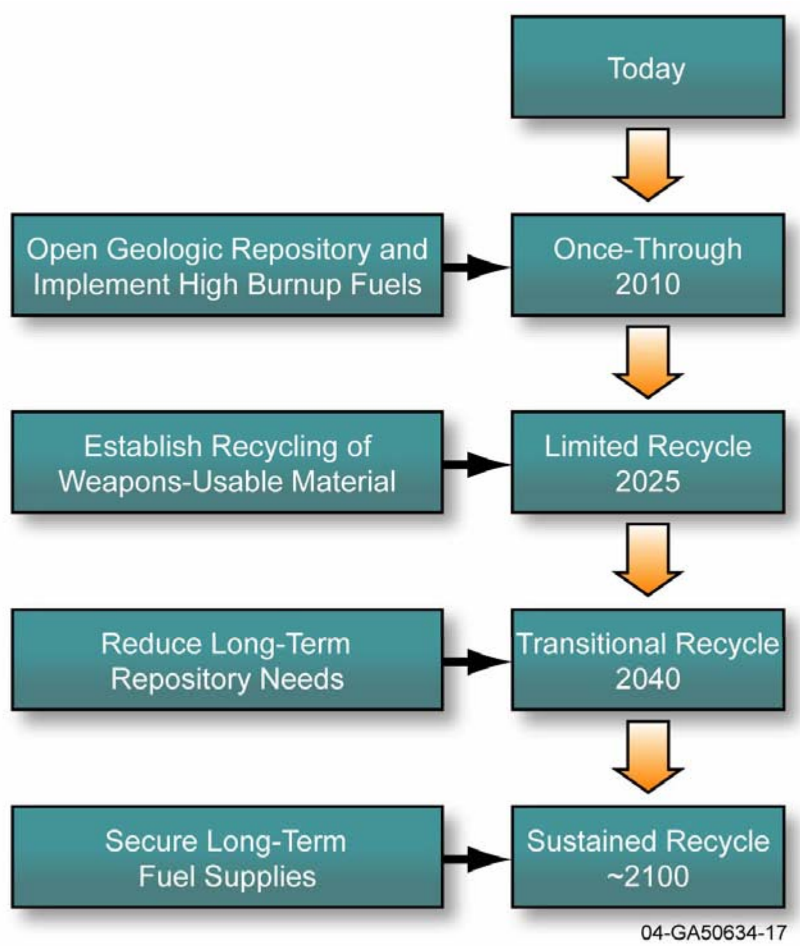
Objective 1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.

Objective 2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for spent fuel management.

Objective 3. Enhance energy security by extracting energy recoverable in spent fuel and depleted uranium, ensuring that uranium resources do not become a limiting resource for nuclear power.

Objective 4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.

Fuel Cycle Evolution Strategy

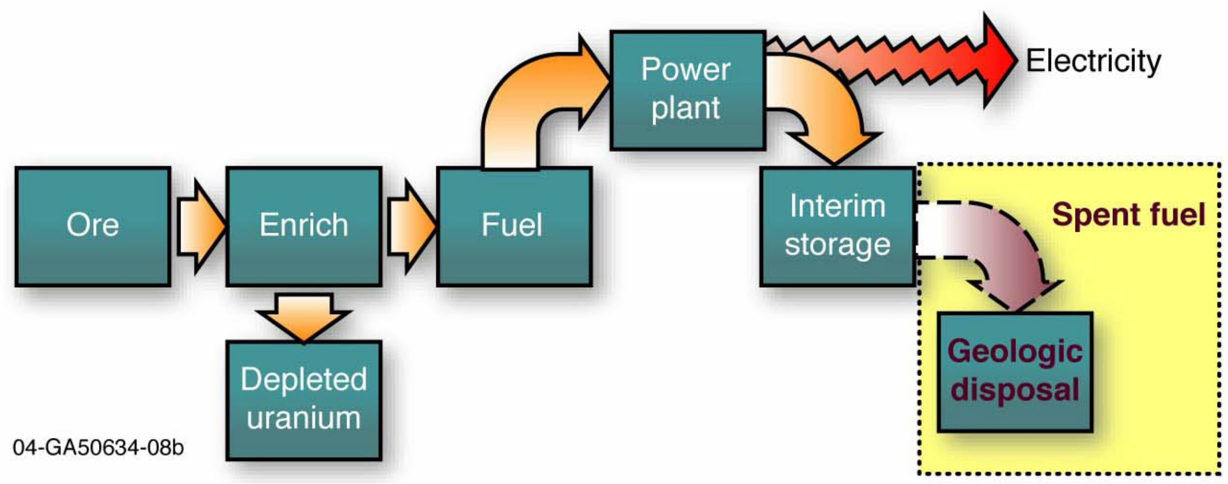


- **Today** – Spent fuel is stored indefinitely at the reactor sites.
- **Once-Through** – Spent fuel is directly disposed.
- **Limited Recycle** – Spent fuel is recycled a few times, then directly disposed.
- **Transitional Recycle** – Spent fuel is recycled continuously and only fission products are disposed.
- **Sustained Recycle** – Depleted uranium is used for fuel, only fission products are disposed.

AFCI Strategy

Once-Through Phase

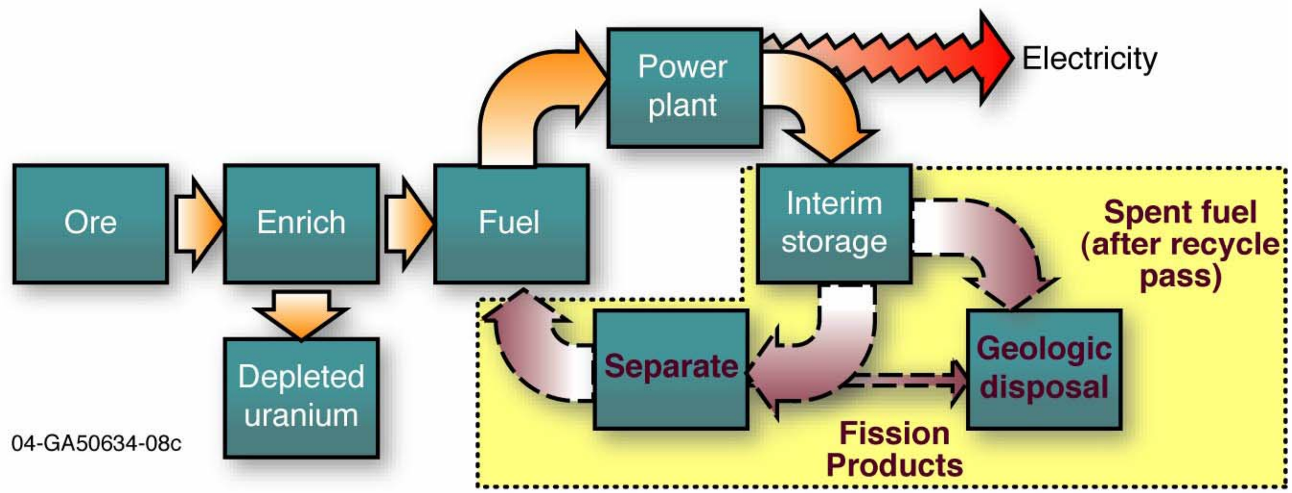
- Support opening of the geologic repository
- Develop high burn-up fuels to reduce spent fuel production rates



AFCI Strategy

Limited Recycle Phase

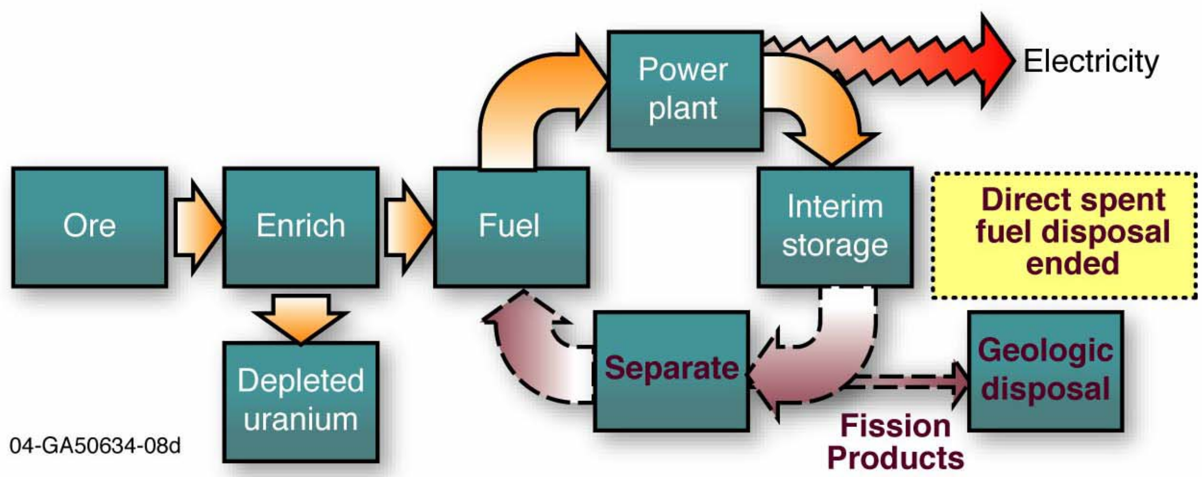
- **Establish commercial spent fuel recycling**
- **Begin destruction of weapons-usable materials**



AFCI Strategy

Transitional Recycle Phase

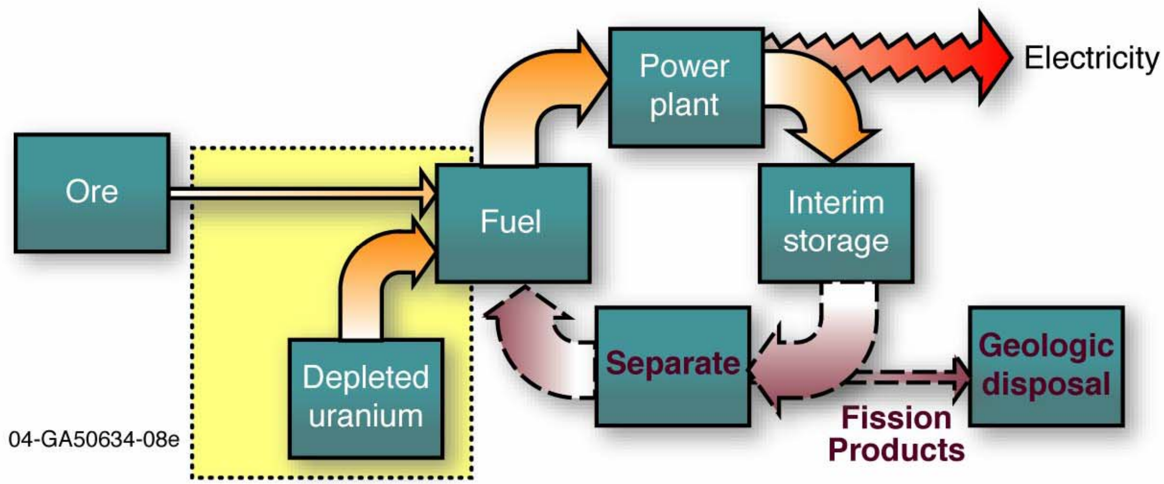
- ♦ **Enhance destruction of transuranics via inclusion of fast reactors**
- ♦ **End direct disposal of spent fuel**



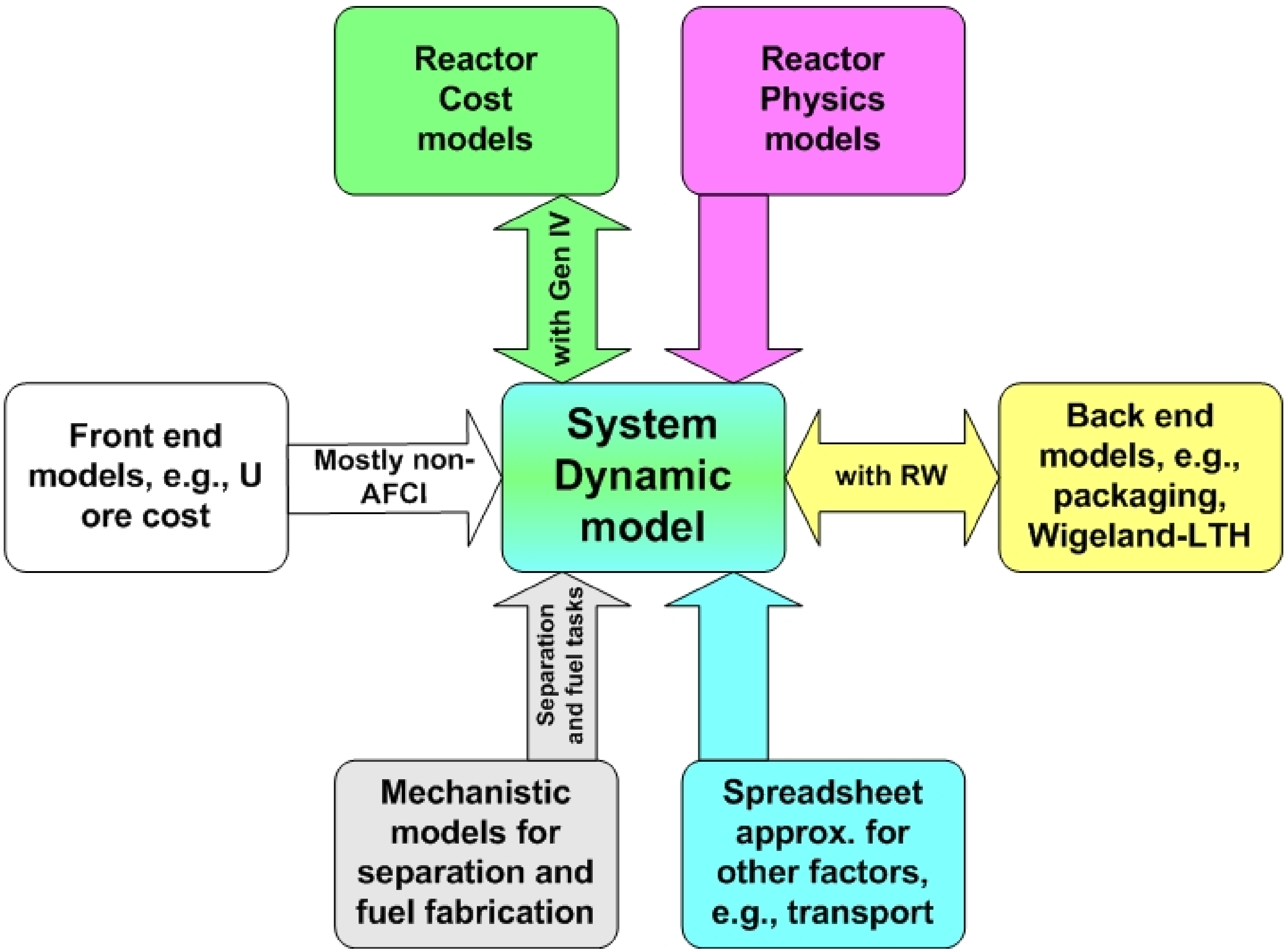
AFCI Strategy

Sustained Recycle Phase

- ♦ **Increase number of fast reactors**
- ♦ **Transmute waste uranium to create new fuel**
- ♦ **Eliminate need for mining and enrichment**



Scenario Evaluation and Trade Studies



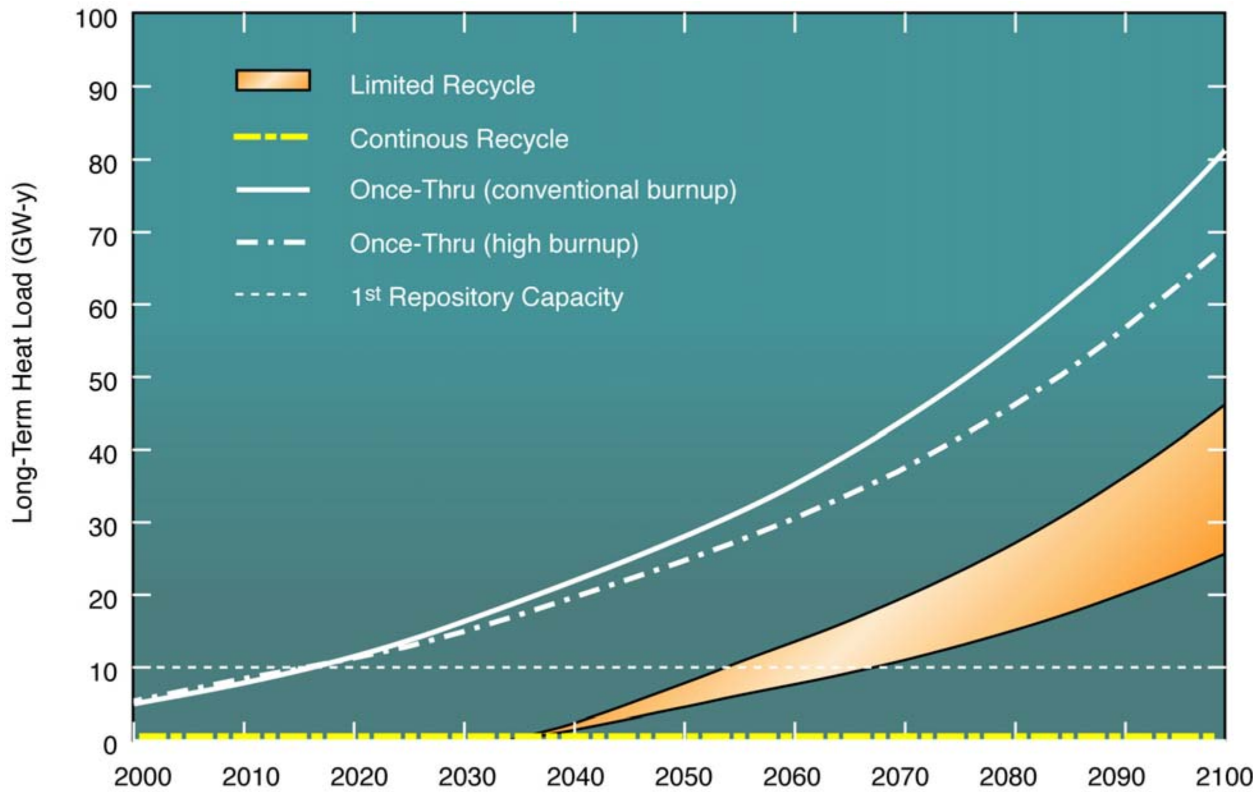
Waste Management Objective

Nuclear Futures		Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share Generation	Growing Market Share Generation
Cumulative discharged fuel in 2100 (metric ton)		100,000	120,000	250,000	600,000	1,400,000
		Existing Reactors Only		Existing and New Reactors		
Fuel Management Approach		Number of Repositories Needed at 70,000 Metric Ton Each				
No Recycle	Once-Through	2	2	4	9	20
No Recycle	Once-Through, High Burnup Fuels	2	2	3	7	17
Reprocess & Recycle	Limited Recycle, High Burnup Fuels	Recycle not applicable		2	5	10
Reprocess & Recycle	Continuous Recycle			1	1	1
Reprocess & Recycle	Sustained Recycle			1	1	1

04-GA50634-15

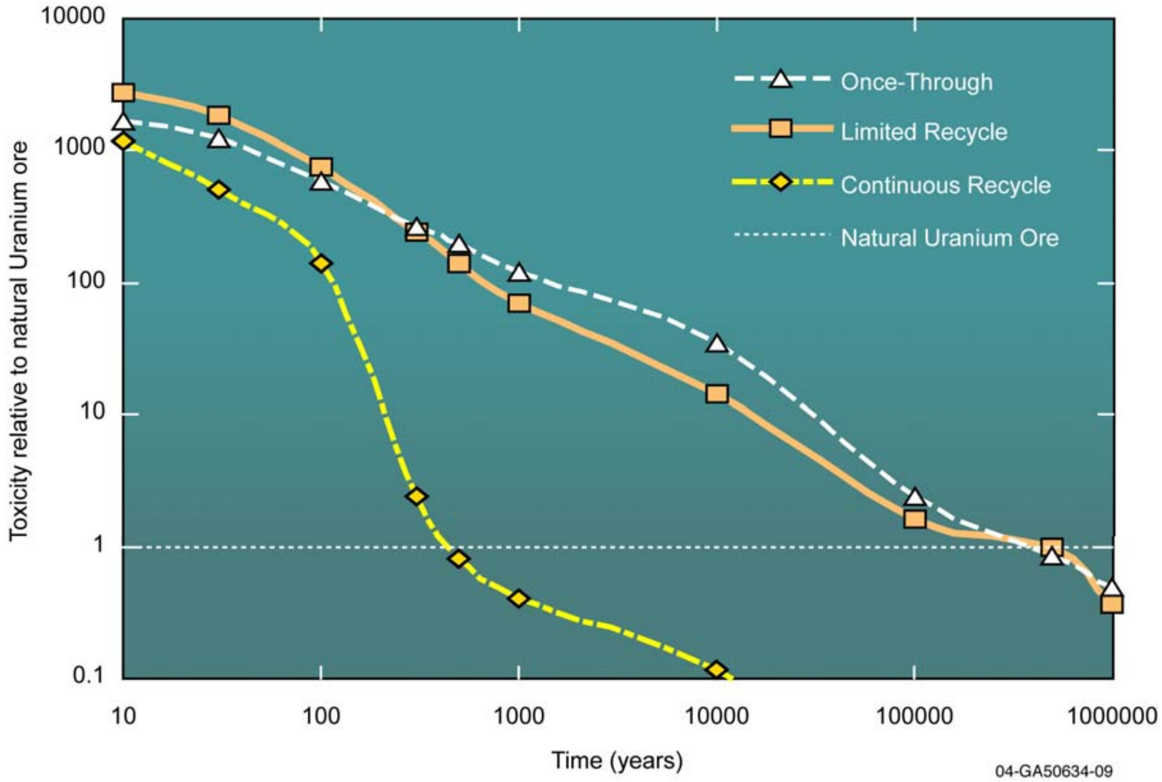
- **For a continued once-through fuel cycle, vastly expanded permanent disposal space will be needed this century**
- **Imminent decisions regarding either the expansion of geologic disposal space and/or implementation of recycle are required**

Waste Management Objective: Long-Term Heat Load Reduction



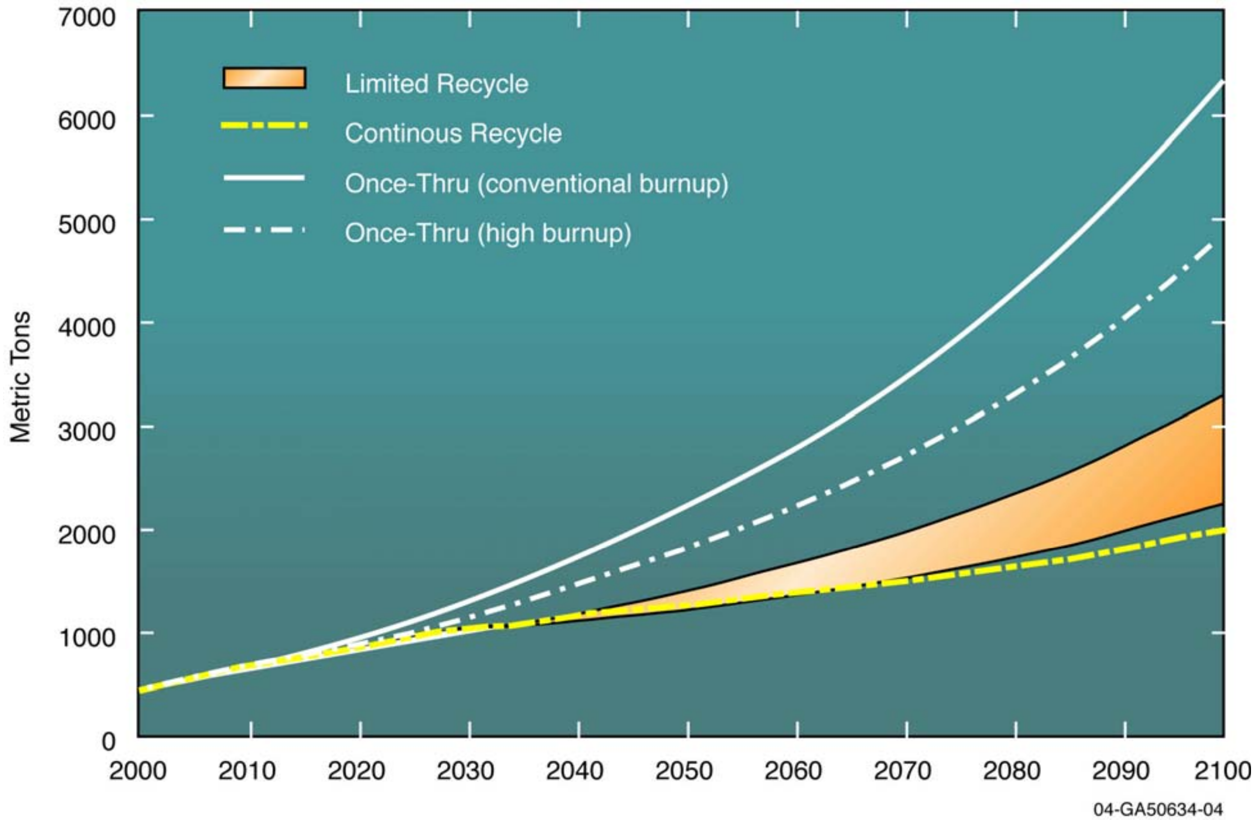
- **Continuous recycle (no direct disposal of spent fuel) yields large reductions in the inventory of long-term heat producers**
- **Limited recycle serves as a delay line for disposal of long-term heat, however, eventual reduction is only ~1/2, at best**

Waste Management Objective: Radiotoxicity Reduction



- **Once again, continuous recycle required for significant reduction**
- **The AFCI Continuous Recycle strategy can significantly improve the basic nature of nuclear waste disposal (thermal load and isolation time frame)**

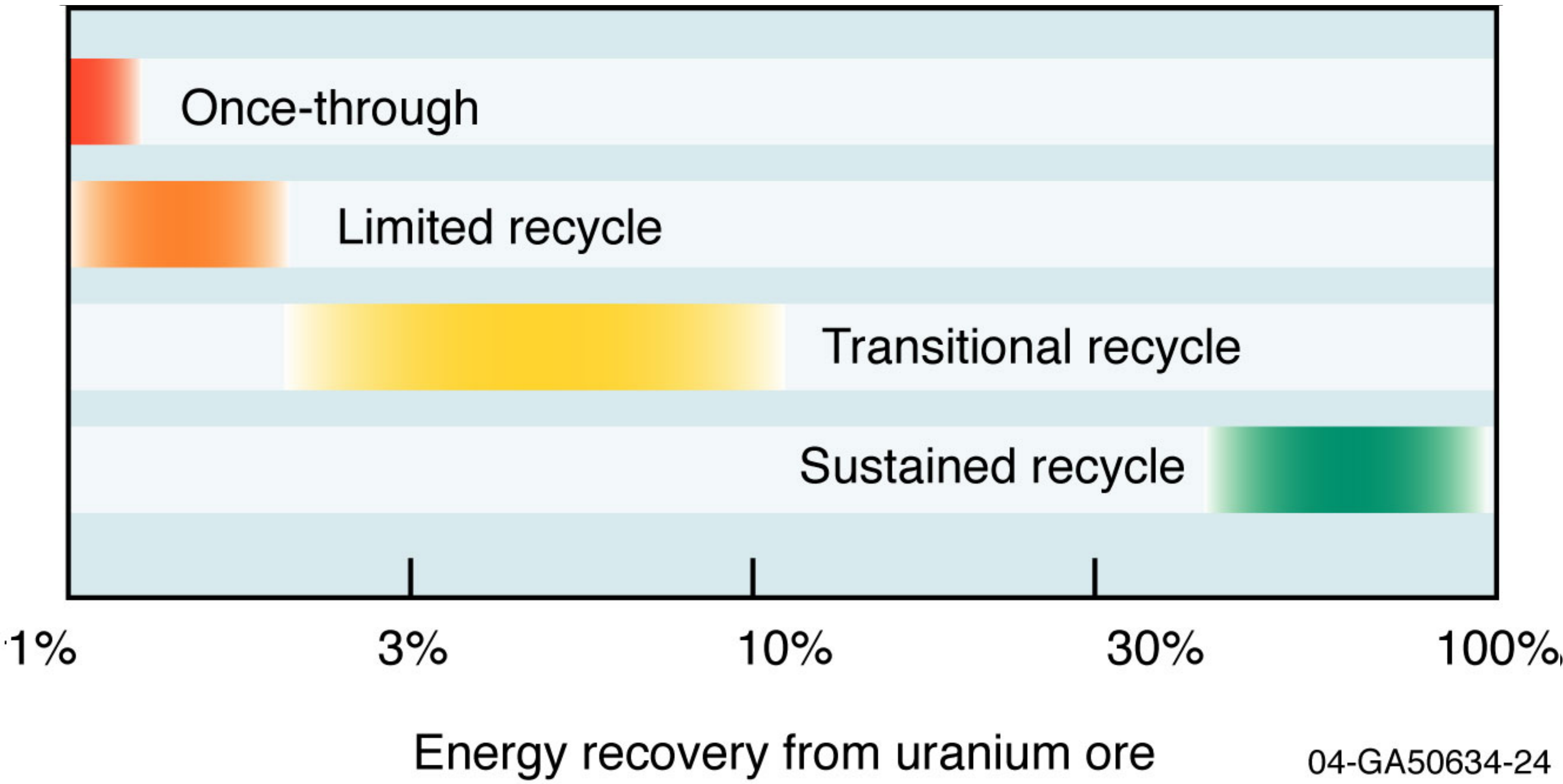
Proliferation Objective
Plutonium Inventory



- All AFCI strategies will reduce the plutonium inventory compared to a continued once-through fuel cycle
- A more aggressive implementation of AFCI technology could be employed to stabilize or decrease the plutonium inventory

Resource Objective
Uranium Needs

- **AFCI and Generation IV technologies can eliminate uranium supply concerns**



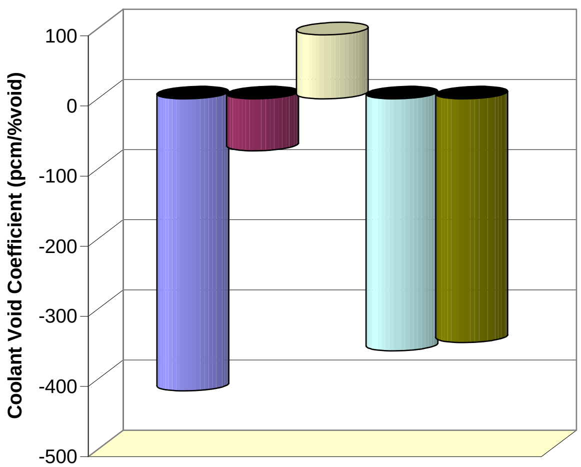
Transmutation Options and Analysis Approach

- **Systematic assessment of transmutation system technology and implementation options**
  - Large body of existing international work on transmutation
  - Most research focused on details of specific options
  - Develop techniques to consistently compare diverse options
- **Respond to inquiries regarding transmutation strategy**
  - Construct and evaluate fuel cycle scenarios
  - Integrate work to respond to NERAC and external questions
  - Address key systems planning/direction issues
- **Perform detailed fuel cycle analyses to address key issues**
  - Transmutation performance of promising options
  - Spent fuel characterization for repository benefits analysis
  - Quantitative evaluation of fuel cycle impacts of specific options

Overview of Recent Transmutation Analyses

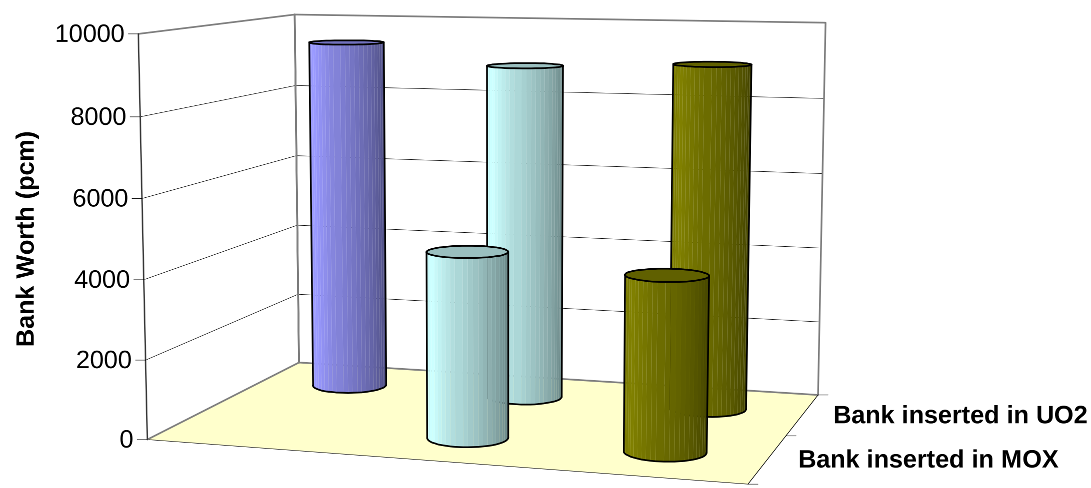
- **Systematic comparison of transmutation options**
  - Investigation of dynamic fuel cycle scenarios
  - Comparison of transmutation potential for thermal/fast systems
- **Thermal reactor transmutation options**
  - Completion of LWR recycle assessment with CEA
  - Impact of high burnup LWR fuel on waste management
  - Consideration of Am/Cm management options
    - Both homogeneous and heterogeneous approaches
  - Impact of inert matrix fuel (IMF) on LWR recycle performance
- **Fast reactor transmutation options**
  - Completion of low conversion ratio SFR safety assessment
  - Adaptation of Gen-IV concepts for transmutation
    - Development of GFR burner designs

Reactivity Coefficient Estimates for MOX and Standard Fueled LWRs

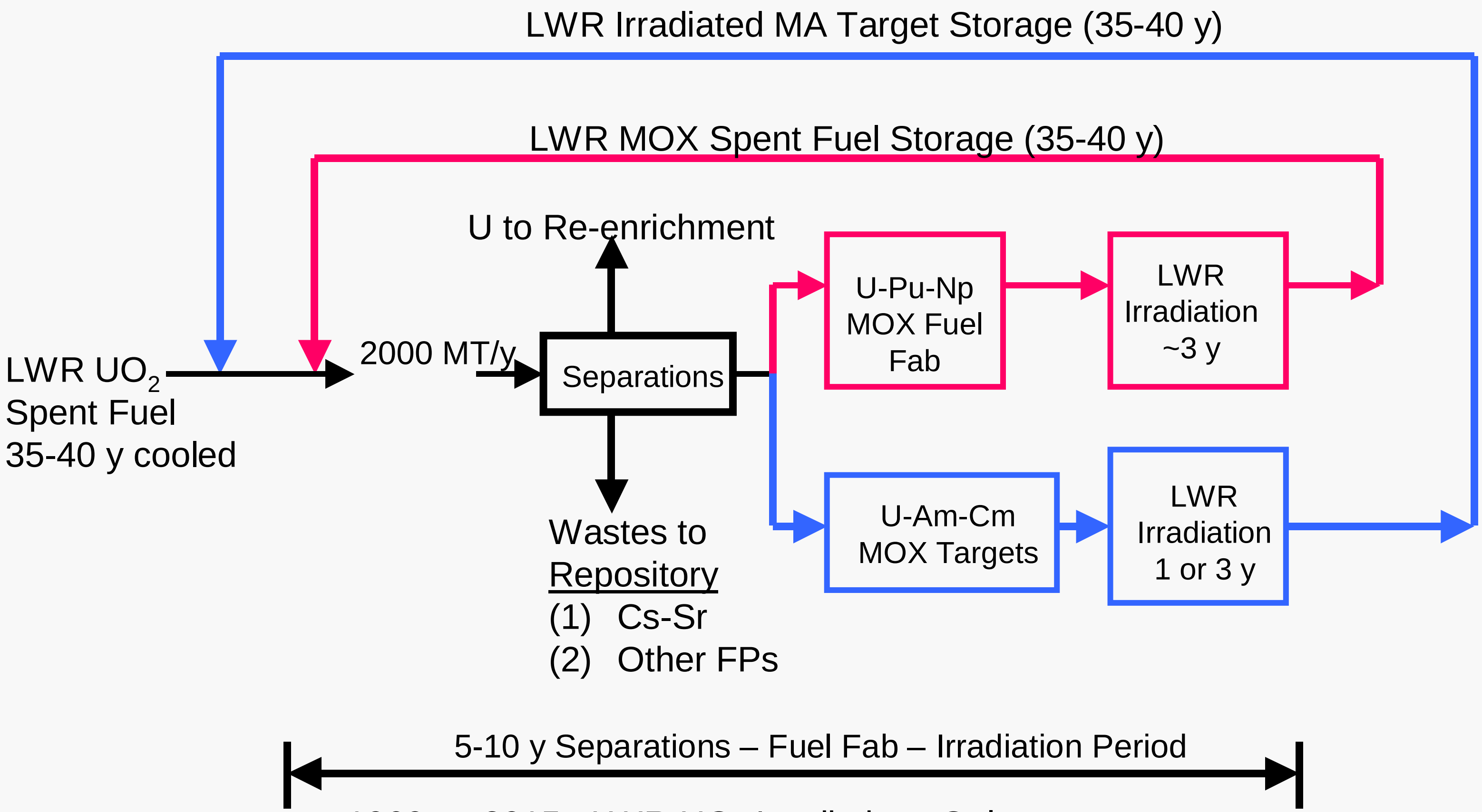


- **All UO_2 core: 3.85 wt.%U-235**
- **All Pu-MOX core: 9.5%Pu/HM**
- **All Pu+Np-MOX core; 14.0%Pu+Np/HM**
- **¼ Pu-MOX core: 8.00%Pu/HM (avg) in MOX, 3.85 wt.%U-235**
- **¼ Pu+Np-MOX core: 8.35%Pu+Np/HM (avg) in MOX, 4.10 wt.%U-235**

- **Coolant void coefficient (shown at left)**
  - Compared with all UO_2 core, void coefficient is 15-20% less negative for partial MOX core
  - All Pu+Np-MOX core has positive void coefficient
- **Control bank worth (shown below)**
  - Estimates based on standard bank (B_4C material) inserted in 48 core locations
  - Control bank worth in UO_2 is 5% lower in mixed core
  - Control bank worth 30-50% lower when inserted in MOX assembly



Am/Cm Transmutation Strategy



1960s – 2015: LWR UO_2 Irradiations Only
2015 – 2055: LWR UO_2 + LWR MOX 1st Cycle Irradiations
2055 – 2095: LWR UO_2 + LWR MOX 2nd Cycle Irradiations

Fast Spectrum Transmutation Options

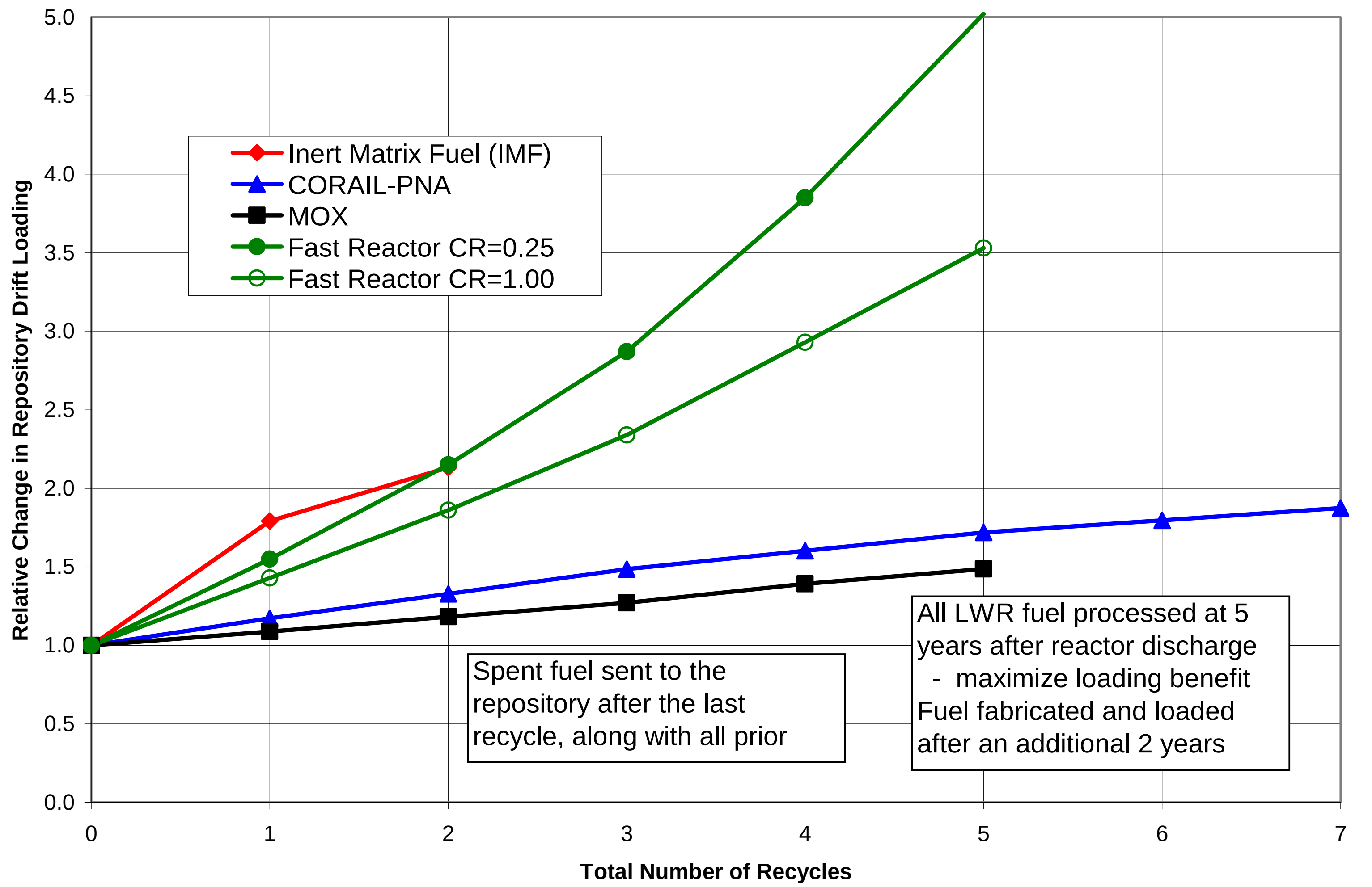
System	Conventional SFR Burner	Low CR SFR Burner	ADS
TRU Conversion Ratio	0.55	0.25	0.00
Net TRU consumption rate (kg/yr)	108	193	270
Fuel Volume Fraction,	0.38	0.22	0.19
Fuel Enrichment, % TRU/HM	27/33	44/56	100
TRU Inventory, MT of TRU	4.36	2.25	2.66
Burnup Swing (%Δk)	1.35	4.26	4.14

- TRU consumption rate significantly higher for low conversion ratio systems
- Enrichment is roughly 50% for low conversion ratio burner
- Burnup reactivity loss rate much faster at low conversion ratio
- Favorable passive safety performance is retained at low conversion ratio
- GFR burner designs have been developed with similar performance

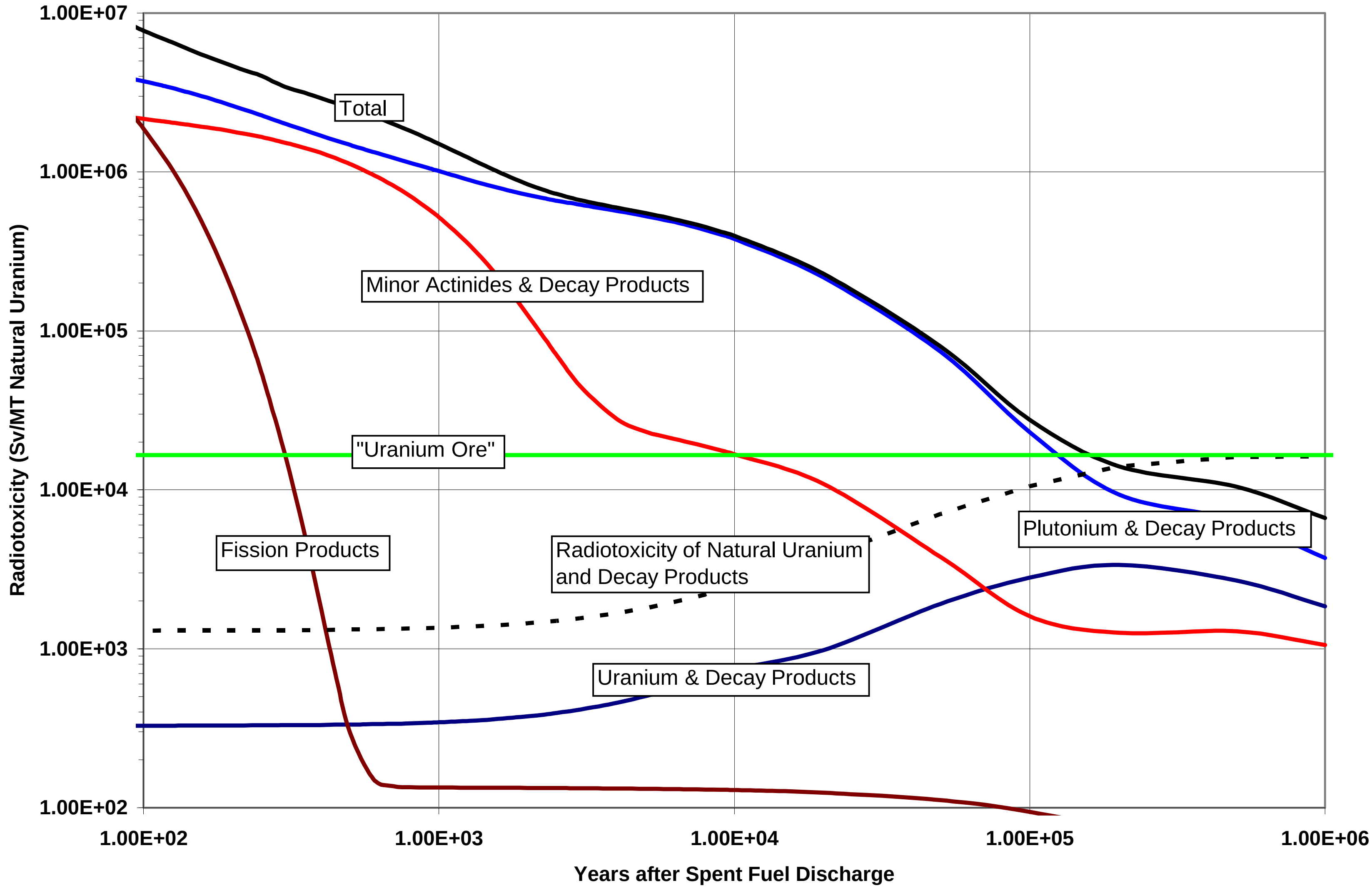
Key Repository Performance Criteria

- **Thermal criteria set temperature limits for the repository and its contents; still evolving, including**
  - peak temperature below the local boiling point (96 ºC) at all times midway between adjacent drifts (tunnels)
  - peak temperature of the drift wall below 200 ºC at all times
- **Dose criteria set limits on the peak dose rate during the regulatory period of 10,000 years**
  - limit of 15 mrem/year for the maximally reasonably exposed individual
- **For each set of criteria, what separations are of benefit?**
  - What additional processes, such as transmutation, are required?

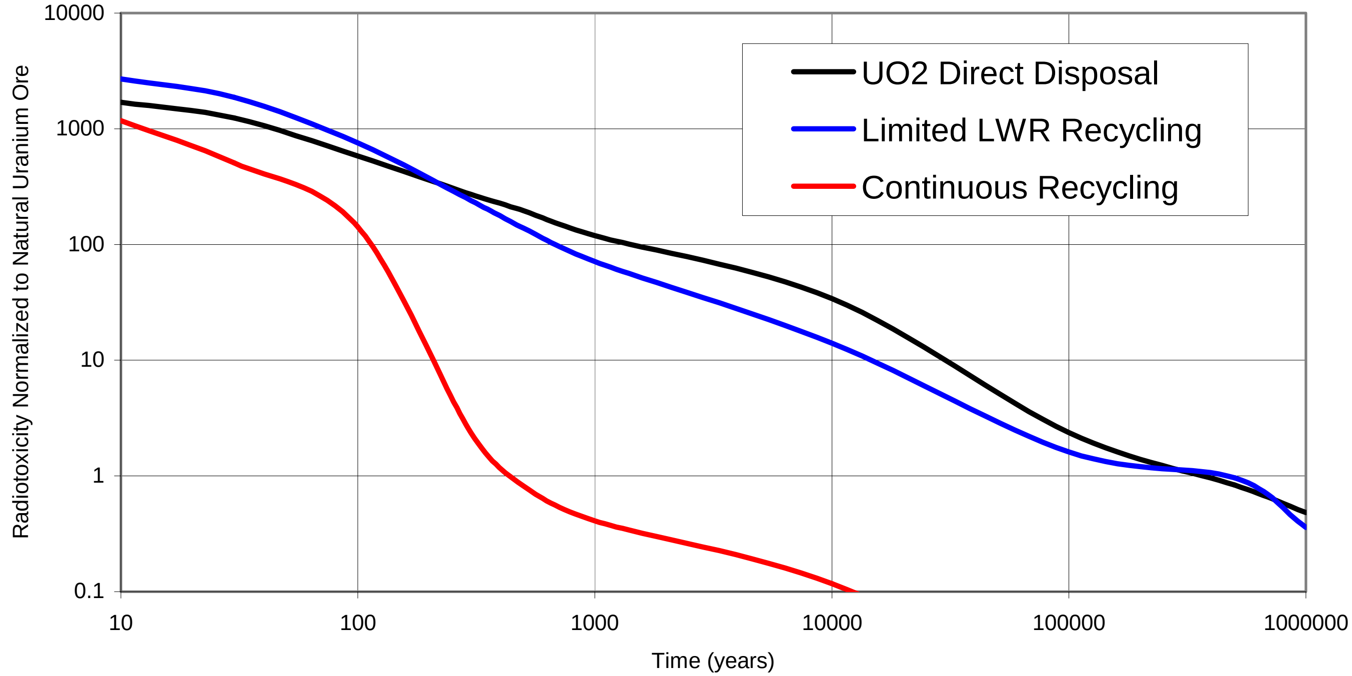
Recycle Impacts on Repository Capacity



Spent PWR Fuel Radiotoxicity



Repository Radiotoxicity Reduction



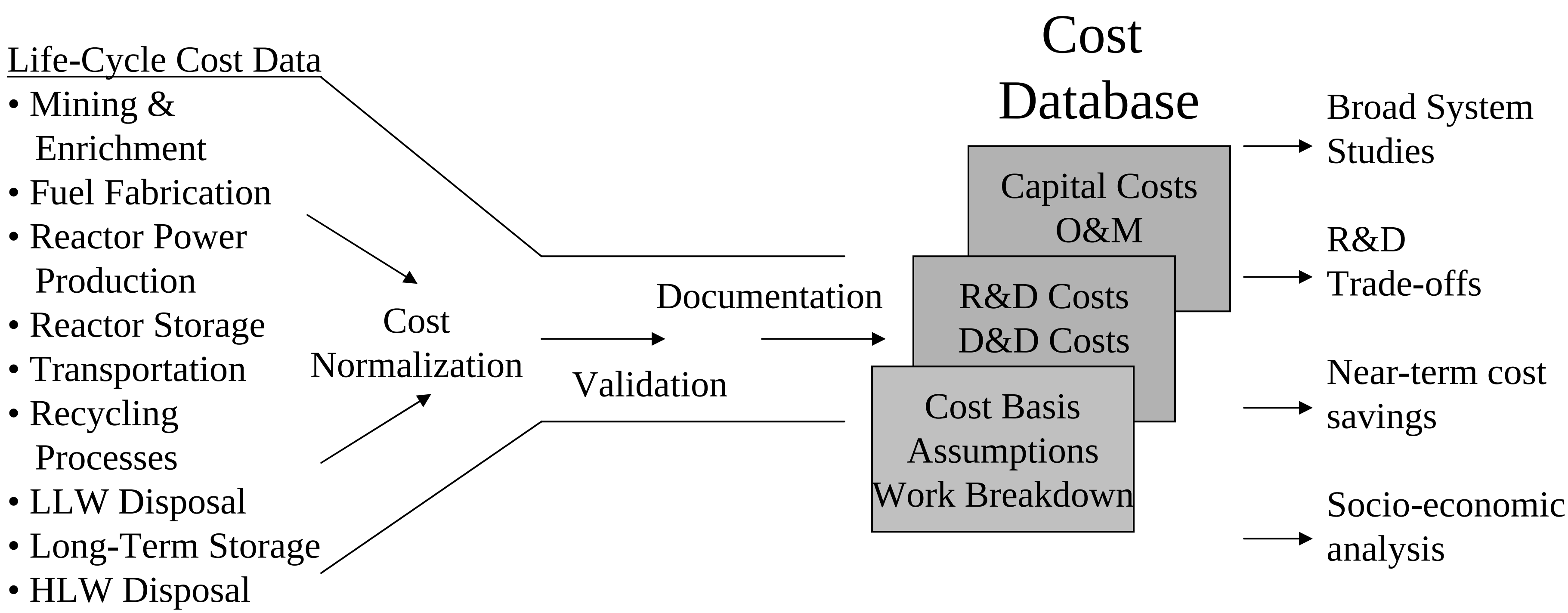
- Limited LWR recycling provides little radiotoxicity benefit when most or all components of the waste stream are considered
- Only disposing of process waste provides a large benefit by keeping actinide inventory, including uranium, very low
- Radiotoxicity reduction achieved only by actinide consumption

Summary of Repository Benefit Analysis

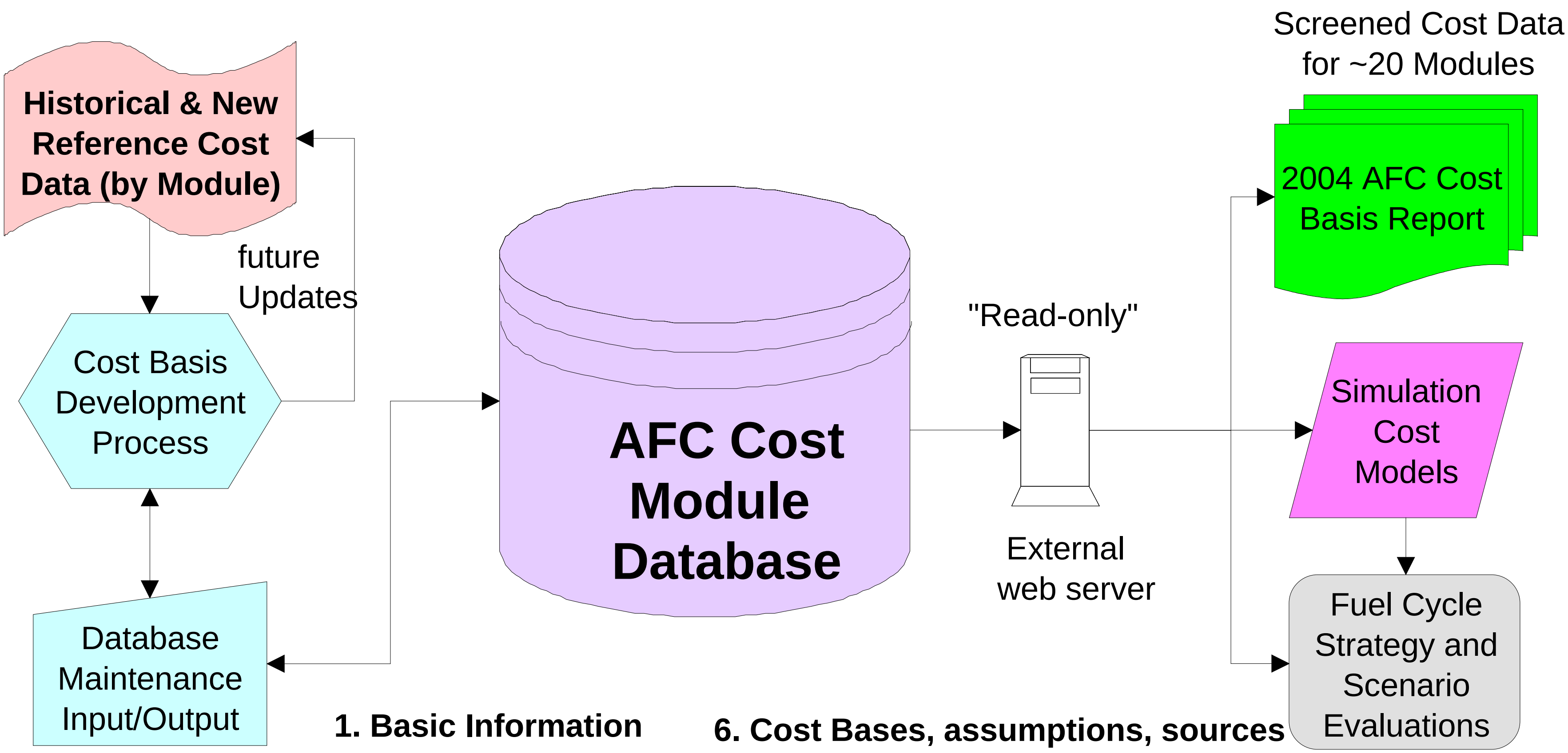
- **Separation of certain elements can substantially benefit the design and operation of the Yucca Mountain repository**
  - Plutonium and americium separation and transmutation to provide increased loading (reduced repository area)
    - factor of 2-6, depending on separation efficiency and timing
  - Subsequent cesium and strontium removal for further increases in loading (or reductions in repository area)
    - factors upwards of 60, again depending on separation efficiency and timing
  - Neptunium separation and transmutation for long term dose rate reduction
    - reduced from hundreds of mrem/year to < 15 mrem/year
- **Efforts are now underway in several areas**
  - Separation of other elements, such as curium
  - Evaluation of the benefits of specific strategies (MOX, etc.)
  - Quantify impact of potential loading strategies for waste
  - Direct interaction with DOE-RW

Economic Benefits

Establish a credible cost basis that provides consistent comparison of AFCI & Gen IV deployment options

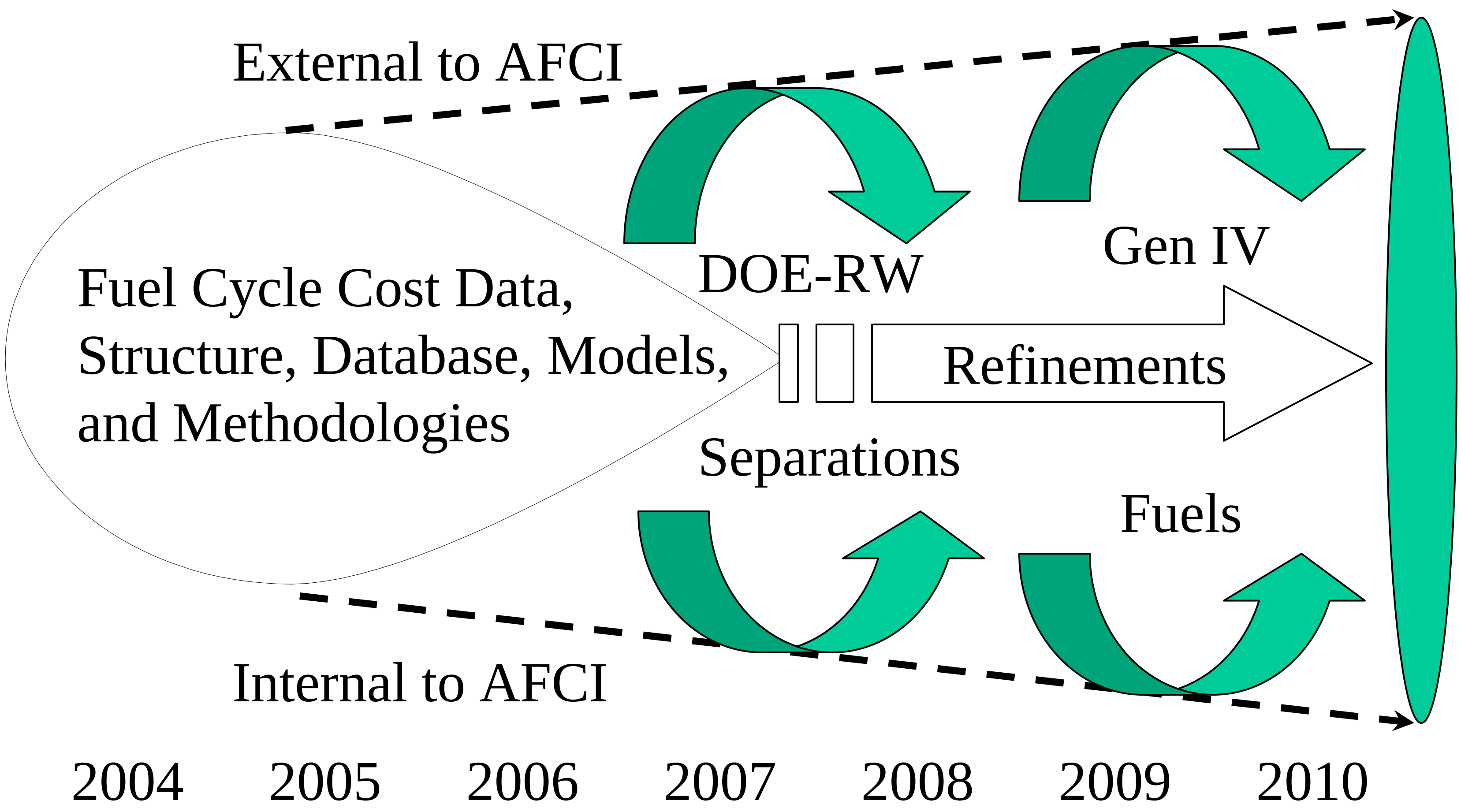


Advanced Fuel Cycle Cost Database



1. Basic Information
2. F&OR Description
3. Picture/Schematics
4. Interface Definition
5. Scaling Factors
6. Cost Bases, assumptions, sources
7. Data Limitations
8. Cost Summaries
9. Sensitivity and Uncertainty Analysis
10. References

Economic Benefits – Vision to 2010



Systems Analysis Deliverables

Annual	**Provide Report to Congress comparison matrix to indicate R&D status and program directions**
FY 2005	**Report to Congress "AFCI Objectives, Approach and Technology Summary"**
FY 2005	**Initial Technical Options Report (input to Secretarial Recommendation)**
FY 2006	**Develop safeguards by design methodology**
FY 2006	**Initial report on proliferation resistance of a reprocessing facility**
FY 2007	**Complete total fuel cycle cost uncertainty analysis**
FY 2007	**Provide necessary information for a Dec 2007 Secretarial recommendation on the need for a second repository with a final report on a recommended US transmutation approach**
FY 2008	**Complete national and global materials simulation capability and apply to limited recycle management strategy**
FY 2008	**Initial report on industry view of fuel cycle management options**

PLANS FOR FY06-08

- **Collaborate with DOE-RW on technical options for future commercial SNF management**
- **Provide recommendations on fuel types and reactor systems from the standpoint of the overall fuel cycle (continuing)**
- **Provide analyses as input to recommendation on need for second repository (2007)**
- **Provide system impact assessments to support reprocessing technology decisions (2007)**
- **Develop Simulation Institute for Nuclear Energy Modeling and Analysis**

Simulation Institute for Nuclear Energy Modeling and Analysis

- ***Revolutionize the nuclear energy enterprise by creating a powerful, science-based, fuel-cycle modeling and simulation capability***
  - *Create new tools for designing and developing innovative nuclear technologies*
  - *Guide choices for technology development to meet the society's needs*
  - *Reduce cost of R&D and deployment*
  - *Ensure high fidelity and reduce uncertainty*
  - *Shorten the time of development and deployment*
  - *Rebuild the R&D infrastructure in an optimal way*
  - *Conduct virtual experiments where physical experiments are impossible or too expensive*

SINEMA Mission

- ***Establish a powerful network for computational capability***
- ***Avoid expensive and lengthy development process for multiple fuel cycle and reactor options***
  - *Optimize systems in the computational domain and select the most promising candidates*
  - *Shorten the development time by avoiding complex full-scale experiments*
- ***Analyze all elements of the fuel cycle for nuclear energy production and nuclear materials management***
  - *Economics*
  - *Safety and environmental*
  - *Proliferation*
  - *Sustainability*
- ***Provide objective input to National and International decision makers***
- ***Build joint projects among universities, national laboratories, industrial and regulatory agencies***

Systems Analysis Budget

FY 2004 $4.33M total

FY 2005 $5.78M total

$533K - AFCI Criteria and Assessment

$950K - Scenario Evaluations and Trade Studies

$425K - Repository Benefits

$875K - Economic Benefits

$1259K - Transmutation Criteria (includes Deep Burn analyses)

$1402K - Transmutation Options and Analyses

$200K - Nonproliferation and Safeguards

$131K - Integration

FY 2006-10 $6.5M/yr total

Major Conclusions

1. We are establishing systematic understanding of the feasibility of transmutation in reactor systems.
2. We are developing a portfolio of waste management strategies.
3. We understand how to make optimal use of existing infrastructure and technologies.
4. We are now moving into better utilizing this knowledge in a set of global studies that will support the 2007 Secretarial recommendation.

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Hussein S. Khalil
Director, Nuclear Engineering Division

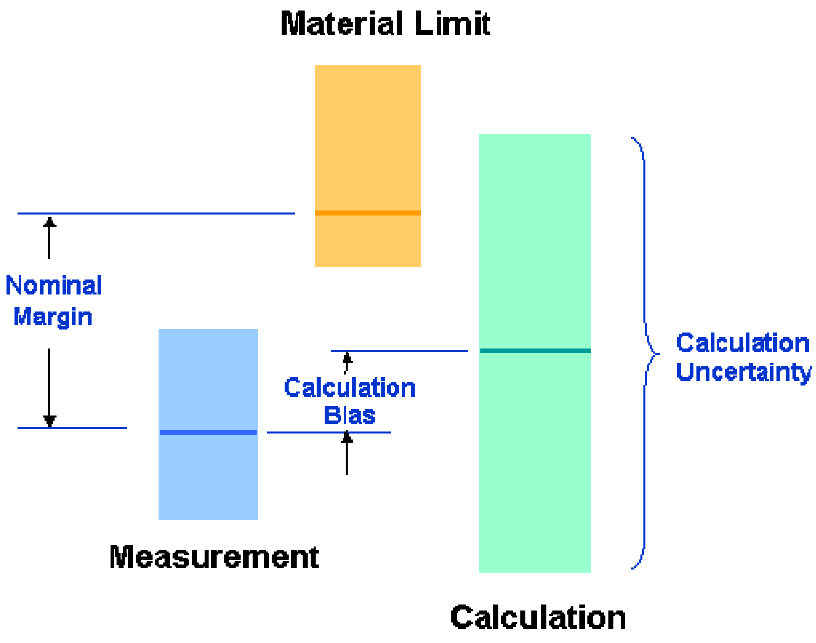
Generation IV Design and Evaluation Methods

Argonne National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

TOPICS: Analysis Methods

- **Rationale**
  - **Enable accurate predictions of system performance**
    - **Confirm viability of new technologies and design features; requires credible analyses verified with experimental data**
    - **Quantify performance advances relative to current generation systems**
  - **Reduce modeling uncertainties that necessitate conservatism in design**
    - **Increase assurance of performance gains, *prior to system operation***
    - **Avoid potentially costly efforts to improve upon the capabilities of available technologies**
  - **Provide the tools needed for future regulatory reviews and licensing of Generation IV systems**



TOPICS: Analysis Methods

- **Approach**
  - Specify analytical capabilities needed to design Gen IV systems and characterize their performance
  - Identify and document relevant measurements
  - Assess the adequacy of existing measurements and simulation tools
  - Implement and qualify required improvements
- **Examples of modeling needs**
  - Validation of nuclear data for minor actinides, non-standard reactor materials
  - Representation of double heterogeneity of coated particle fuels
  - Accurate modeling of spectral transition regions at core/reflector interface
  - Simulation of small cores with significant global transport effects

TOPICS: Analysis Methods

- **Examples of modeling needs (cont'd)**
  - Reliable estimation of materials damage parameters for in-core and ex-core structures
  - Accurate resolution of SS and transient power, flow, and temperature distributions (reduce hot channel factors)
  - Simulation of systems with moving fuel (PBRs, MSRs)
  - Modeling of natural and mixed convection flows and flow regime transitions
  - Reliable estimation of reactivity feedback from expansion or displacement of reactor components
  - Confirming effectiveness of passive decay heat removal paths and systems (RCCS, RVACS)
  - Adequately simulating the progression and consequences of accident scenarios (e.g., SG tube rupture in LFR, air or water ingression in VHTR)

TOPICS: Evaluation Methodologies

- **Methodologies for evaluating system performance against the Generation IV goals**
  - Economics—need consistent and comprehensive methodology to address new features of Gen IV systems and their deployment
  - Proliferation Resistance & Physical Protection—need to establish and gain consensus for a systematic methodology
  - Risk & Safety—need to establish technical basis for safety review, licensing and regulation of Gen IV systems
  - Others (e.g., more comprehensive evaluation of sustainability)

TOPICS: Evaluation Methodologies, cont'd

- **Key needs in economics evaluation**
  - Standardized approach, yet with flexibility to treat specific features of different markets and systems
  - Represent new characteristics of Gen IV systems, e.g.,
    - Generation of hydrogen and other energy products
    - Effect of plant size (small vs. large units)
    - Closed fuel cycles with new processes implemented in centralized or on-site recycle facilities
    - Actinide management
  - Support evaluations of system cost estimates (proponents' claims) and optimization of economic performance

TOPICS: Evaluation Methodologies, cont'd

- **Key needs in PR&PP evaluation**
  - International consensus methodology (and related terminology)
  - Systematic and comprehensive treatment
    - Consider the relevant proliferation and security threats
    - Account for intrinsic system features and protection/safeguards measures
    - Consider the entire system and life-cycle of materials
  - Identify features of materials, processes, and facilities that contribute to increased PR&PP
  - Formulate limited number of high-level indicators ("measures") of a system's PR and PP

TOPICS: Evaluation Methodologies, cont'd

- **Key needs in risk & safety evaluation**
  - Define approach to safety and safety compliance verification for Generation IV systems
    - Foster safety enhancement as an integral part of system design
    - Define risk and safety evaluation methodology
  - Explore the potential of risk-informed, technology-neutral safety criteria
  - Establish the feasibility of a common technical basis for regulatory review of Gen IV systems in different countries or regions

Relationship to DOE-NE Program Priorities

- **Thrust of the Generation IV International Forum (GIF) is to develop and demonstrate the Generation IV systems**
  - vs. generic R&D
  - System Research Plans (SRPs) developed for VHTR(NGNP), SCWR, GFR, SFR; initiated for LFR and MSR
  - U.S. participates in all, with VHTR receiving highest priority and funding
- **Early focus is on resolving key viability/feasibility questions**
- **Analysis of system concepts is an integral part of the R&D**
  - Provide focus for technology development (fuels, materials, processes)
  - Insure compatibility/integration of different technologies
  - Provide basis for evaluating performance

Relationship to DOE-NE Program Priorities

- **Analysis methods improvements needed to**
  - Accommodate new features of Gen IV systems
  - Reduce bias and uncertainty in calculated performance parameters
  - Reduce need for experimental programs and measurement of differential and integral characteristics
  - Support design development, assessment, and optimization
  - Characterize safety performance and accident behavior
- **Evaluation methodology advances needed to**
  - Measure performance relative to Gen IV goals
  - Provide basis for selecting among system options (e.g., prismatic vs. pebble VHTR, PV vs. PT SCWR, …)
  - Support funding requests (e.g., to OMB, Congress)
  - Attract commercial participation in system design and demonstration

ACCOMPLISHMENTS: Analysis Methods

- **Three workshops on Gen IV analysis needs and capabilities were conducted in FY 2003**

Reactor physics design analysis	Feb 18-19, at ANL
T-H and safety analysis	Mar 18-19, at INEEL
Nuclear data needs	Apr 24-25, at BNL

  - Attended by lab, university and industry representatives
  - Conclusions and recommendations documented
  - Outcome factored into ten-year program plan for Gen IV D&EM

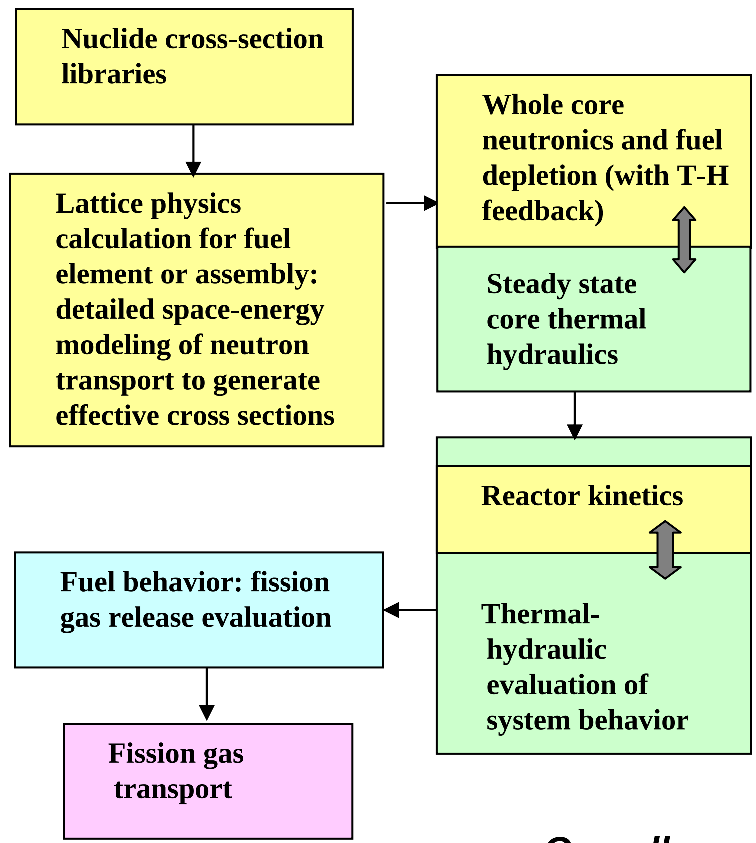
- **Subsequent workshops**
  - International workshop on reactor physics, at PHYSOR Topical Meeting (April 2004, Chicago)
  - International workshop on nuclear data needs (April 2005, Antwerp)
  - Workshop on requirements and capabilities for CFD analysis of advanced, gas-cooled reactors, at ASME Fluids Engineering Summer Conference (June 22, 2005, Houston)

Workshop Findings and Conclusions

- **Existing tools can be adapted for for pre-conceptual design and viability phase analyses**
- **Some modeling improvement are needed, e.g.,**
  - Modeling of natural and mixed convection flows
  - Modeling of flow mixing in outlet plenum of VHTR and GFR
  - Representation of double heterogeneity of coated-particle fuel in neutronic calculations
  - Re-evaluation of minor actinide data for high burnup and closed fuel cycle evaluations
  - Uncertainty propagation in Monte Carlo depletion analysis
  - Better representation of coupled phenomena
- **Need to preserve specifications and measured results of past experiments**
- **Need for systematic approach to prioritize future efforts**
  - International standard problems
  - Use of sensitivity and uncertainty analysis tools
  - Expert identification and ranking of phenomena

Modeling Improvement

- **Activities directed mainly to meeting design development and safety confirmation needs for gas-cooled reactors, particularly the VHTR**
  - Identify and document integral benchmarks applicable to validation of VHTR physics predictions
  - Assess and improve deterministic reactor physics tools for application to VHTR
  - Identify important phenomena and databases for T-H and system dynamics codes
- **Also participated in the planning of GFR critical experiments at CEA Cadarache**



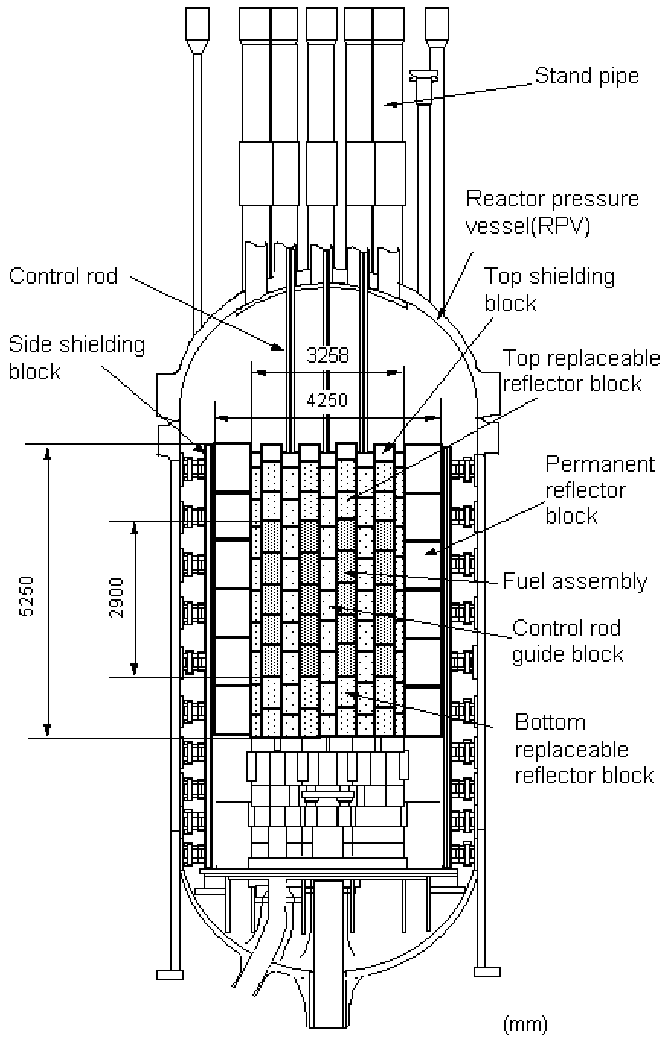
Overall Analysis Process

Neutronic Benchmarks

- **Identified and assessed integral measurements for qualifying VHTR/NGNP physics predictions**
  - Critical experiments: VHTRC (Japan); HITREX-1 (UK); ASTRA, GROG (RF); KAHTER (Germany); HTR-PROTEUS (Switzerland)
  - Reactor measurements: HTR-10 (China); HTTR (Japan); DRAGON (UK); Peach Bottom, FSV (US)
- **Assessment of measurements considered**
  - Similarity/relevance of experiments to VHTR (geometry, fissile material, enrichment, operating conditions)
  - Physical parameters measured and inferred
  - Availability/quality/pedigree of data; adequacy of uncertainty characterization
- **Priorities were identified for PBR and PMR benchmark problem development**
  - HTR-10 (10 MWt, 6-cm pebble core, H/D=2.0/1.8m; 17% enriched UO_2)
  - ASTRA (6-cm pebble core, H/D=1.8-3.8/0.9-1.8m, 17% enriched UO_2)
  - HTTR (30 MWt prismatic block core, H/D=2.9/2.3m, 3-10% enriched UO_2)
  - VHTRC (prismatic block core, H/D=2.4/2.4m, 2-6% enriched UO_2)
- **Development of peer reviewed compilation of selected benchmarks to start in FY-05**
  - Through participation in OECD/NEA IRPhE project
  - Compilation to employ rigorous QA standards of ICSBEP project

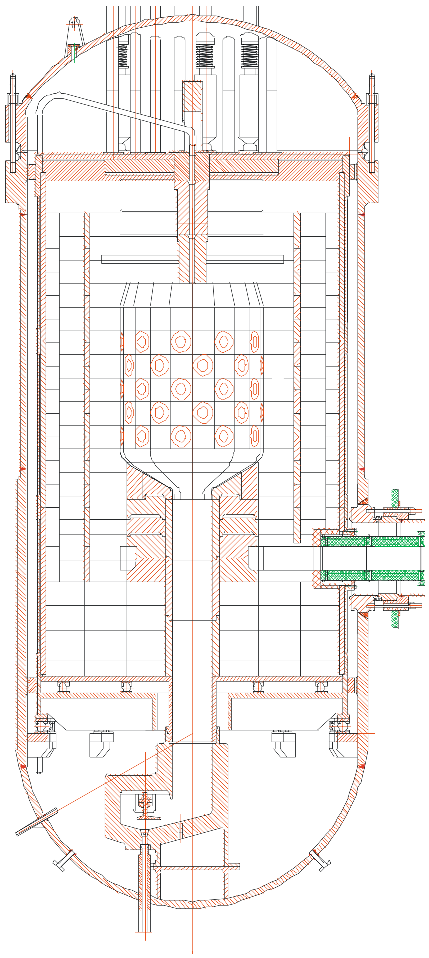
Neutronic Benchmarks, cont'd

- **High Temperature Test Reactor (HTTR)**
  - JAERI has released to the IAEA HTTR start-up core physics data
    - Annular-core data representative of block-type VHTRs are available
  - Data is being used to validate lattice and whole-core analysis code suite proposed for VHTR analysis
    - DRAGON or WIMS and DIF3D
    - MCNP4 for high fidelity solution
  - Core excess reactivity calculated by DRAGON/DIF3D in good agreement with results obtained by European and Japanese groups
  - Solution refinements (group structure and geometry) ongoing to determine improvements required to obtain better agreement with experimental results
  - MCNP4 used to determine impact of approximations



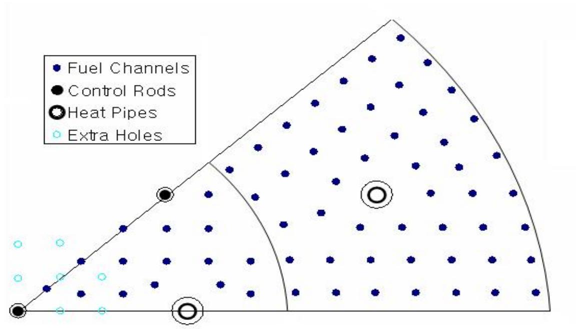
Neutronic Benchmarks, cont'd

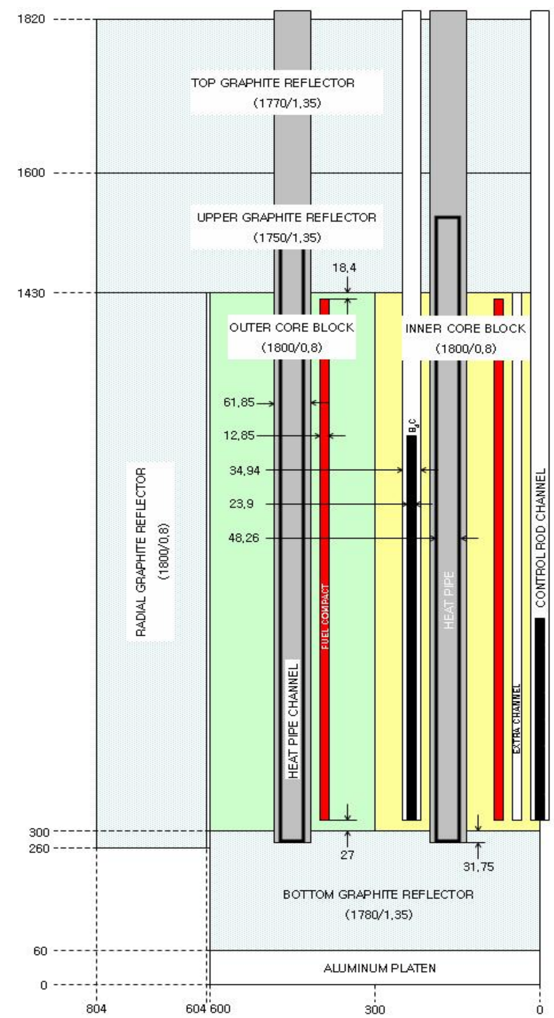
- **HTR-10 Benchmark (China)**
  - Specifications provided by the HTR-10 project in IAEA-TECDOC-1382
  - Available configuration information and experimental data have been collected and reviewed
  - A PEBBED model of the HTR-10 reactor has been developed and successfully run, with an artificial set of cross sections
  - COMBINE models are currently being developed to generate an appropriate set of effective cross section



Neutronic Benchmarks, cont'd

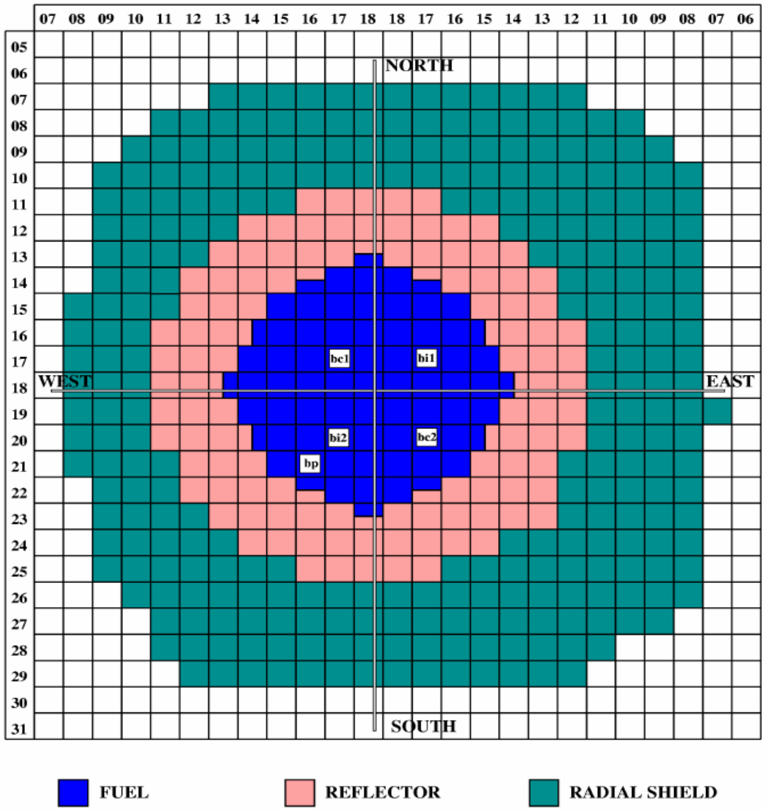
- **Compact Nuclear Power Source (CNPS)**
  - Preliminary evaluations indicated that the CNPS physics experiments conducted at LANL in the 1980s are useful for validating codes for VHTR systems
    - Small reactor (20 KWe) for remote sites
    - Design used TRISO fuel particles (LEU) and core had inherently large negative temperature reactivity coefficient
  - Development of Stochastic and deterministic core models ongoing





GFR Integral Physics Measurements

- **Participated in planning of the "ENIGMA" GFR critical experiments at the CEA MASURCA facility**
  - **Experiments motivated by significant differences between GFR cores and SFR cores previously studied**
- **Goal was to achieve neutronic similarity of experimental configuration with reference GFR design—within constraints on core size, geometry, and materials**
  - **Key needs: neutron streaming, spectral transition at core/reflector interface, and inclusion of appropriate materials (ZrC, Zr_3Si_2, SiC)**
- **Analysis quantified achievable reduction in GFR physics uncertainties achievable with the measurements**
- **Status of ENIGMA**
  - **First experiments now scheduled for late 2006/early 2007 to accommodate planned facility upgrades (fire safety, new control rods, and seismic evaluations)**
  - **Safety tests are required in middle 2006 to demonstrate adequacy of new control rods (worths)**



Preliminary GFR Test Configuration

Physics Code Assessment

- **Performed assessment of codes for VHTR physics analysis with aim of identifying and implementing the capabilities needed for NGNP development**
  - Key near-term need is to configure and qualify a reasonably automated code system that conforms with modern standards
- **Assessment covered following components of such a code system:**
  - Nuclear data processing: preparation of fine-group libraries from evaluated nuclear data
    - Available tools (e.g., NJOY) judged adequate
  - Lattice physics: determination of effective multigroup cross sections for each core region as function of burnup, temperature, control insertion, etc.
    - WIMS8 and DRAGON codes selected because they contain double heterogeneity models for coated-particle fuels
    - Main needs are to develop and validate models and application procedures
  - Whole core neutronics and burnup simulations: neutron flux calculation and modeling of fuel depletion
    - DIF3D/REBUS-3 code system identified for prismatic block VHTR; PEBBED for pebble bed VHTR
    - Key needs include representation of core thermal conditions, core-reflector coupling effects, and neutron streaming
- **Required improvements of numerical standard codes such as Monte Carlo (MCNP) and whole-core MOC analysis capability (DeCART) also defined**

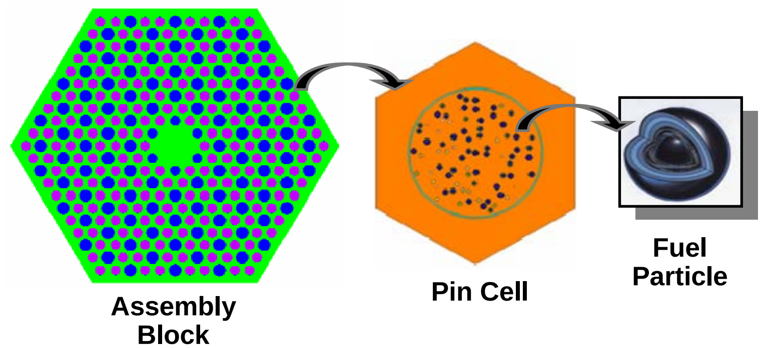
Assessment of Lattice Physics Codes

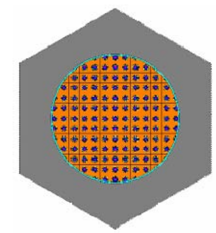
Comparison of Lattice Codes for Cross Section Generation for NGNP

	MICROX	DELIGHT	WIMS8	DRAGON	APOLLO	MCNP
Solution Options	CP	CP	CP/ S_n /MOC	CP/MOC	CP/MOC/ S_n	Monte Carlo
Energy Groups	193 (pointwise data for resonances)	111	172 or 69	172/69 or variable	172	Continuous
Problem Geometry	Unit cell	Unit cell	Whole assembly	Whole assembly	Whole assembly	General
Treatment for double heterogeneity (DH)	Yes	Yes	Yes	Yes	Yes	Explicit particles
Geometry of DH treatment	1-D cylindrical cell	1-D cylindrical cell	1-D cylindrical cell	Assembly	Assembly	Full geometry
Usage Issues	Lacks assembly modeling	Lacks assembly modeling. ENDF/B-III and IV data	Separate unit cell and whole assembly calculations	No library distributed (allows various types)		σ's at limited temperatures. Lattice modeling of particles
State of Use	Used for high temperature reactor designs	Used for HTTR (Japan) design	Not extensively used	Not extensively used	Not extensively used	Generally used
Availability	At code centers	At code centers	Commercial code	Open distribution	CEA-France code	At code centers
Testing of Model, V&V	IAEA benchmarks. Reactor startup. Critical experiments	Critical experiments and reactor startup	IAEA benchmarks	Comparison to MONK and WIMS8. IAEA benchmarks	Code-to-code comp. IAEA benchmarks. Critical experiments	IAEA benchmarks. Critical experiments and reactor startup

CP: Collision probability
MOC: Method of Characteristics
S_n: Discrete Ordinates

Assessment of Lattice Physics Codes, cont'd



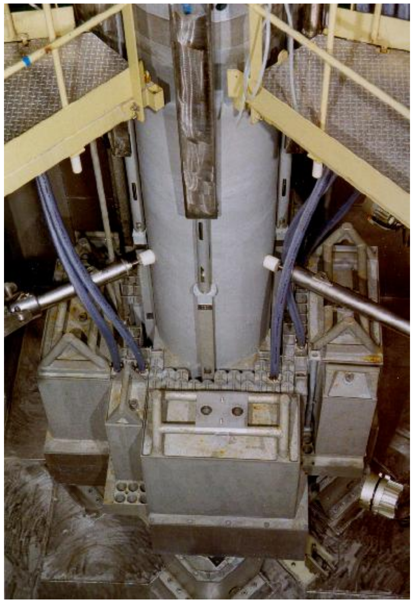


Regular (lattice) distribution of particles typically used to approximate actual stochastic distribution

k_∞ Estimates:

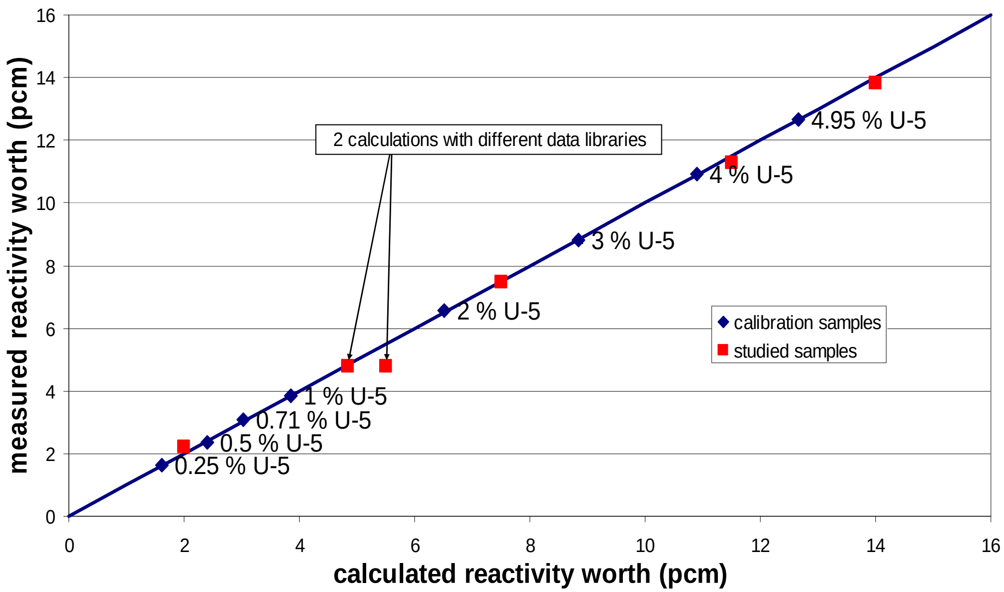
			GT-MHR, UO_2	GT-MHR, $(TRU)O_{1.7}$	VHTR, $UC_{0.5}O_{1.5}$
MCNP4C	Random Distribution		**1.57335 ± 0.00040**	**1.25838 ± 0.00040**	**1.53280 ± 0.00082**
	Lattice Distribution	SC	1.57279 ± 0.00039	1.25427 ± 0.00071	1.52978 ± 0.00071
		BCC	1.57118 ± 0.00041	1.25317 ± 0.00070	1.53160 ± 0.00071
		FCC	1.57276 ± 0.00041	1.25192 ± 0.00077	1.52890 ± 0.00073
DRAGON			1.57565 (93)	1.26794 (599)	1.54393 (470)
WIMS8			1.57121 (-86)	1.25326 (-325)	1.52993 (-122)
Double heterogeneity effect			1.4 % $\Delta\rho$	13.1 % $\Delta\rho$	2.3 % $\Delta\rho$

OSMOSE: CEA/DOE I-NERI Project



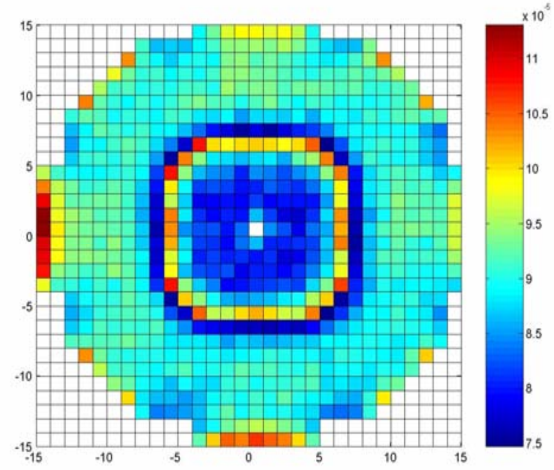
- Goal: To measure integral reaction rates very accurately, in representative spectra for the actinides important to future nuclear system designs
- Outcome:
  - Database of reactivity-worth measurements for ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm
  - Identification of deficiencies in data libraries of minor actinides

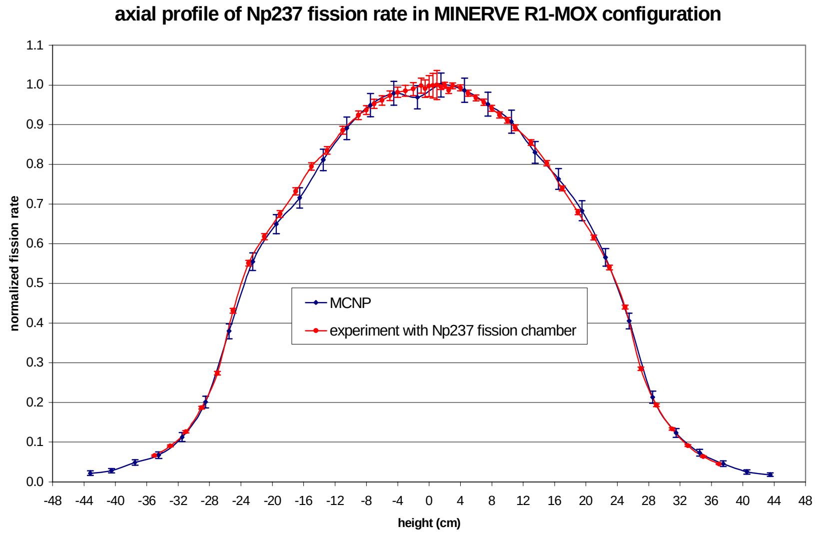
- **Oscillation Measurements**
  - MINERVE Reactor
    - fundamental mode
    - well-characterized
  - Small samples of separated isotopes
  - Very accurate measurement technique



OSMOSE: CEA/DOE I-NERI Project, cont'd

- **The MINERVE Reactor has been characterized in the R1-UO2 and R1-MOX configurations – includes experimental measurements and calculations for:**
  - k-eff; conversion ratio; CR reactivity, calibration sample worths; axial and radial fission rates; spectral indices
- **Oscillation measurements using the calibration samples were performed and analyzed**
- **Pellets for 5 test samples have been fabricated; purification and analysis of the feedstock materials for the other samples has been completed**





FY	Measurement Spectra
2005	PWR UO2
2006	PWR UO2, PWR MOX
2007	PWR MOX, Epithermal
2008	Epithermal
2009	Hard Epithermal
2010	Overmoderated UO2
2011	Moderated Fast
2012	Fast

Measurements carried out for: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm

VHTR T-H and Safety Modeling

- **Initiated process of assessing, selecting and qualifying models for VHTR T-H and safety analysis**
  - Considering both quasi 1-D system codes and 3-D CFD software

- **Overall process**
  - Identify operational and accident scenarios and safety related thermal hydraulic phenomena
  - Perform scaling analysis of phenomena to identify dimensionless numbers and their ranges
  - Review of applicability of T-H and system code models given scaling analysis results; identify areas for improvement
  - Screen existing experiments for use in developing improved correlations for codes; identify new experiments
  - Compare code calculations against experiments to identify model uncertainties and provide guidance for experiment plan and model improvement

VHTR T-H and Safety Modeling

Operating Regimes in Block-Type VHTR

						Upsets													
						Protected							Unprotected							
			Full Power Operation	Operational Transients	Refueling	Loss of Generator Load	Reactivity Insertion	Loss of Cooling	Shaft Breakage	Loss of Coolant	Overcooling	Flow Blockage	Loss of Generator Load	Reactivity Insertion	Loss of Cooling	Shaft Breakage	Loss of Coolant	Overcooling	Flow Blockage
Values of Operating Regime Variables	Pressure	Normal Pressure	X	X		X	X	X	X		X	X	X	X	X	X		X	X
		Depressurized			X					X							X		
	Cooling	Forced Convection	X	X	X	X	X		X		X		X	X		X		X	
		Conduction						X		X		X			X		X		X
	Heating	Neutronic	X	X									X	X	X	X	X	X	X
		Decay Heat			X	X	X	X	X	X	X	X							

VHTR T-H and Safety Modeling

Identification of Design/Safety Issues in Block-Type VHTR

Operating Regime	Location	Phenomena	Design/Safety Issue
OR4 - *Normal Pressure/* *Conduction Cooling/* *Neutronic Power*	Air Duct in Reactor Cavity Cooling System	P3 - *Laminar-Turbulent Transition* Transition region flow lies in the continuum between the laminar and turbulent flow regions.	Heat transfer and pressure drop in air duct may be underestimated resulting in reduced heat removal capability.
OR4 - *Normal Pressure/* *Conduction Cooling/* *Neutronic Power*	Air Duct in Reactor Cavity Cooling System	P4 – *Mixed Convection* Significant buoyancy-driven flow may develop taking friction pressure drop and heat transfer out of either the turbulent convection or laminar convection regime into the mixed convection regime.	Heat transfer and pressure drop in air duct may be underestimated resulting in reduced heat removal capability.
OR7 - *Depressurized/* *Conduction Cooling/* *Neutronic Power*	Inlet Plenum	P1 - *Thermal Stratification*. Hot plumes rising from prismatic blocks will create stratified region at top of inlet plenum.	Poor mixing at top of inlet plenum may result in elevated temperatures and material creep.

VHTR T-H and Safety Modeling

- **Developed first-cut PIRT for screening of T-H and safety analysis tools and identification of relevant experimental data; identified issues include**
  - Core peaking factors and thermal margins
  - Inlet plenum mixing (flow reversal in LOFC accident)
  - Outlet plenum jetting
  - Effectiveness of RCCS during accident

Scenario	Inlet Plenum	Core	RCCS	Outlet Plenum
DCC		i. Neutronics behavior ii. Bypass at operating conditions iii. Hot channel characteristics at operational conditions. iv. Air & water ingress. v. Potential fission product transport	i. Laminar-turbulent transition flow ii. Forced-natural mixed convection flow	Mixing at operational conditions
PCC	Mixing	i. Neutronics behavior ii. Bypass iii. Laminar-turbulent transition flow iv. Forced-natural mixed convection flow v. Hot channel characteristics at operational conditions	i. Laminar-turbulent transition flow ii. Forced-natural mixed convection flow	Mixing at operational conditions

VHTR T-H and Safety Modeling

Example of VHTR Measurements for T-H Model Qualification

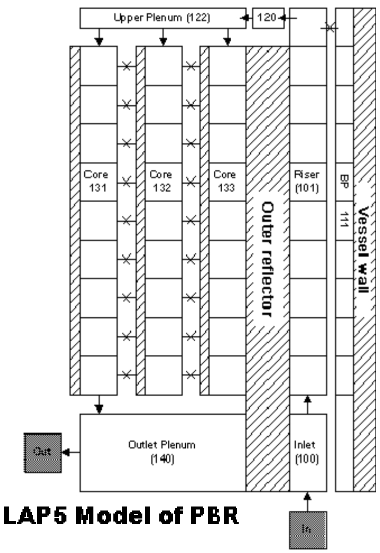
Section	Facility or Event	Scenario or Phenomena	Comments	Reference
3.1	Arbeitsgemeinschaft Versuchsreaktor (AVR)	LOCA test with main circuit valves open	Test performed in May, 1988 to measure effectiveness of natural convection of helium through main loop.	Krüger, et al, 1991. Bäumer, et al, 1990.
		LOCA test with main circuit valves closed	Test performed in May, 1988 to provide comparison to above LOCA test performed with main circuit valves open.	Krüger, et al, 1991. Bäumer, et al, 1990.
		LOCA test with main circuit valves open	Test performed in October, 1988 to measure effectiveness of natural convection of helium through main loop. This experiment had 110 temperature monitor elements in core.	Krüger, et al, 1991. Bäumer, et al, 1990.
		LOCA test with main circuit valves closed	Test performed in October, 1988 to provide comparison to other LOCA test performed in October, 1988 with main circuit valves open. This experiment had 110 temperature monitor elements in core.	Krüger, et al, 1991. Bäumer, et al, 1990.
3.2	Fort Saint Vrain	Scram test on August 6, 1977.	A scram was initiated from 28% power. Core power, inlet temperature, flow and outlet temperature data are available.	Ball, et al., 1977. Ball, et al., 1978a Ball, 1980
		Scram test on October 25, 1977.	A scram was initiated from 40% power. Core power, inlet temperature, flow and outlet temperature data are available.	Ball, et al., 1978b
		Scram test on July 23, 1977.	A scram was initiated from 30% power. Core power, inlet temperature, flow and outlet temperature data are available.	Ball, et al., 1978c Ball, 1980
		Scram test on July 8, 1980.	A scram was initiated from 35% power. Core power, inlet temperature, flow and outlet temperature data are available.	Ball, et al., 1981
		Scram test on May 8, 1978.	A scram was initiated from 50% power. Core power, inlet temperature, flow and outlet temperature data are available.	Ball, 1980
3.3	G1 Reactor Accident	Severe accident	Qualitative assessment of core damage. Opportunity to perform validation calculations regarding oxidation of graphite, aluminum cladding, and metallic uranium.	Wichner & Ball, 1999.
3.4	HTR-10 Experiments	Loss of primary flow without scram	At operational conditions the primary helium blower is quickly stopped and the primary isolated from the water cooling systems on the secondary side of the steam generator. All control units remain where they were before the transient begins. The transient is run at partial load. These data not available until an agreement is reached with Generation IV Program.	Sun & Gao, 2003
		Control rod withdrawal without scram	A control rod is withdrawn at operational speed so that positive reactivity is introduced; although the reactor protection system responds, the reactor is intentionally not tripped. The transient is run at partial load. These data not available until an agreement is reached with Generation IV Program.	Sun & Gao, 2003

WORK IN PROGRESS FOR FY05: Analysis Methods

- **Reactor physics benchmarks**
  - Perform sensitivity studies for high-priority VHTR benchmarks to confirm relevance and potential for uncertainty reduction
  - Document PMR and PBR benchmark specifications and measured results for use in validation and quality assurance of VHTR physics analysis tools
  - Organize the OECD/NEA International Reactor Physics Benchmark Evaluation Project (IRPhEP); peer reviewed compilation of benchmarks to high standard of QA
- **Integral neutronic measurements**
  - Perform reactor modeling and pre-analysis for OSMOSE, support OSMOSE measurements in MINERVE, inter-compare code results for the purpose of code validation
  - Define GFR integral measurements and participate in analysis of ENIGMA GFR experiments in MASURCA
- **T-H and safety analysis tools**
  - Further develop first-cut PIRT for a limiting set of VHTR events
  - Perform scoping analyses to determine the important parameters and their impact upon key safety criteria
  - Identify and assess separate effects measurements relevant to model validation for inlet and outlet plenum modeling

WORK IN PROGRESS FOR FY05: Analysis Methods

- **As part of an I-NERI collaboration, ANL, INL and KAERI are jointly**
  - Specifying generic PBR and PMR configurations for a VHTR
  - Defining limiting VHTR events and associated safety criteria
  - Developing list of classification of components.
  - Specifying key accident phases and phenomena for PIRTs
  - Carrying out sensitivity analyses for the key phenomena
- **Preliminary tables of phenomena were produced without the rankings**
- **Next step is to specify the rankings for the PIRTs**



RELAP5 Model of PBR

Phases	Phase ID	Event Scenarios and Major Processes
1	Loss of Forced Circulation	• Event initiated by loss of offsite power and/or turbine trip + failure of SCS to start + reactor shutdown • Rapid increase of fuel temperature and vessel temperature drops rapidly
2	Heat-up	• Core heat-up slows down by natural circulation inside core • Heat removal by conduction cooling and radiation to RCCS is smaller than core afterheat
3	Cooldown	• Heat removal by conduction cooling and radiation to RCCS is larger than core afterheat • Core cools down to safe shutdown states

Phases for HPCC Transient

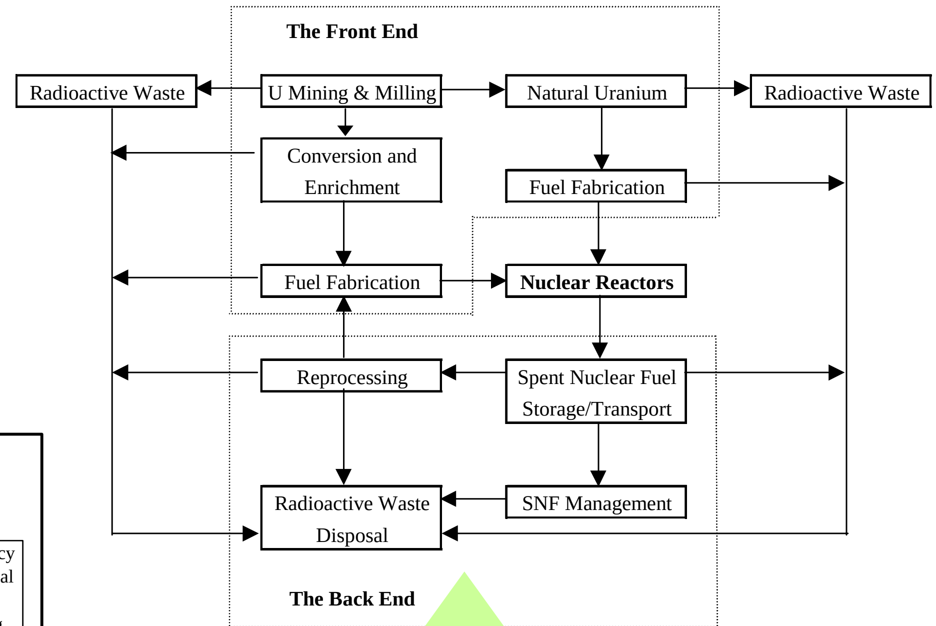
Phenomena	HPCC			DCC			FT	
	1	2	3	1	2	3	1	2
flow distribution	o	o	o	o	o	o	o	o
heat transfer (forced convection)	o			o			o	o
heat transfer (mixed and free convection)		o	o		o	o		
pressure drop (forced convection)	o			o			o	o
pressure drop (mixed and free convection)		o	o		o	o		
Decay heat	o	o	o	o	o	o		
Reactivity feedback							o	o
Fuel/reflector conductivity	o	o	o	o	o	o	o	o
Fuel/reflector specific heat	o	o	o	o	o	o	o	o
Multi-D heat conduction including contact	o	o	o	o	o	o	o	o
Gas conduction		o	o		o	o		
Radiation heat transfer		o	o		o	o		
Graphite oxidation						o		
Homogeneous chemical reaction						o		

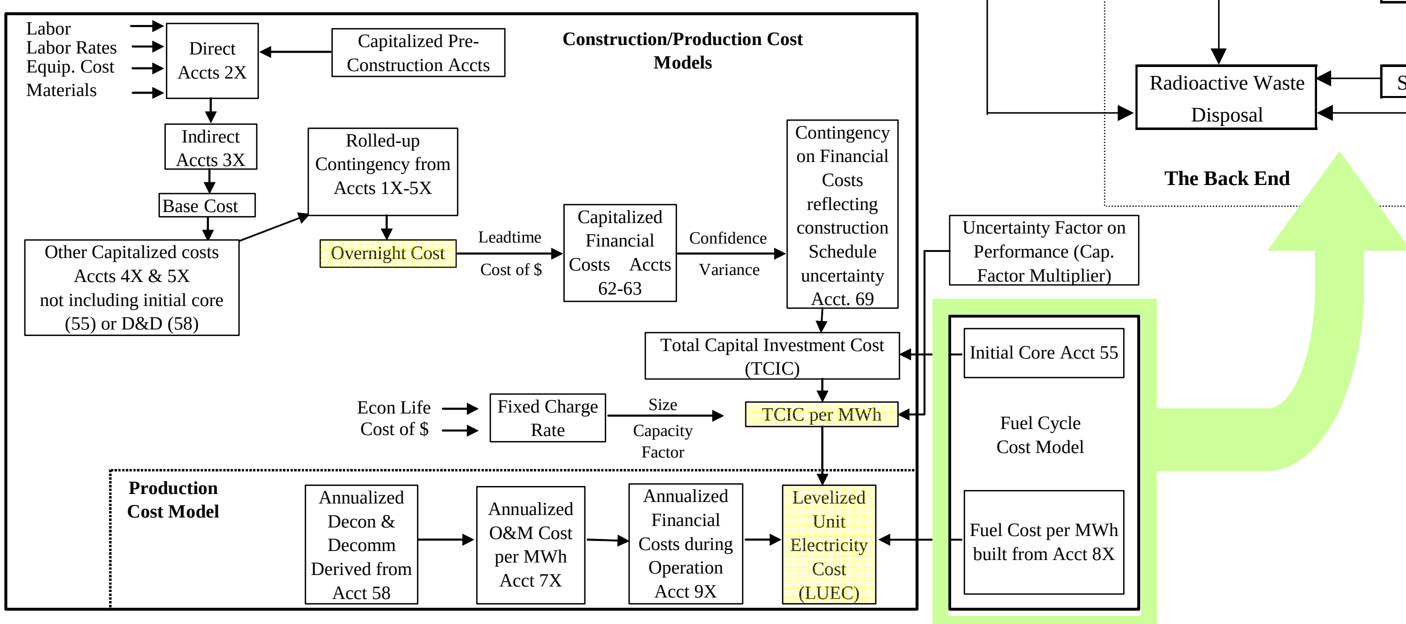
ACCOMPLISHMENTS: Economic Evaluation

- **Adapted capital and production cost models for international applications**
  - Eliminated or generalized system- and country-specific assumptions (e.g., treatment of taxes and depreciation)
  - Adopted uniform assumptions, e.g., concerning cost categories, commodity prices, labor rates, financial discount rates, licensing approach/costs, site requirements, plant capacity factor
  - Standardized approach for calculating total and levelized capital costs, O&M costs
  - Documented in "Cost Estimating Guidelines for Generation IV Nuclear Energy Systems"
- **Completed draft specifications for an integrated nuclear economics model (completed 6/30/04)**
- **Implemented the Guidelines in developmental software**
- **Verified software using example ALWR case**

Integrated Model Specification

- Expands on the Gen IV Roadmap description of economic models
- Serves as a guide as model development proceeds





ACCOMPLISHMENTS: Economic Evaluation

- **New revision (Rev. 1) of Guidelines recently completed**

- **Rev. 1 highlights**
  - Cost estimating process clarified
  - GIF Code of Accounts (COA) defined based on IAEA COA, with correlation to the EEDB
  - Top-down estimating process described and guidance for its application provided
  - RD&D Code of Accounts added to standardize the display and tracking of planned RD&D costs
  - Treatment of contingency added

WORK IN PROGRESS FOR FY05: Economics Evaluation

- **Worked to collect labor and commodity data for European and Asian markets**
- **Initiated hydrogen production cost model**
  - Performed literature search on hydrogen economics
  - Used EMWG software to repeat GA cost calculation for a hydrogen generation plant
- **Initiated plant size model**
  - Prepared white paper on modularity
  - Performing case study on large vs. small plant installation
- **On-going refinement of COA and Guidelines based on application experience**

Model for Hydrogen Economics

- ♦ **Guidelines were applied for calculation (display) of costs for a hydrogen generation system**
  - Using GA data for a modular helium reactor
  - Using corresponding cost categories for an electricity generating plant
- ♦ **Parametric variations explored sensitivities to**
  - Financing charges
  - Contingency

Annualized and Levelized Cost for 4-unit MHR

Case:	GT-MHRx4	GT-MHRx4 adjusted	PH-MHRx4 CFR=10.5%	PH-MHRx4 w/First Core CFR=10.23%	PH-MHRx4 adjusted CFR=10.23%
Annualized Cost in $M/yr					
Capital Cost incl Financing	145.785	181.469	169.197	151.450	231.067
Operations Cost	30.110	30.110	73.981	73.981	73.981
Fuel Cycle Cost	66.834	66.834	66.834	66.834	66.834
D&D Cost	0.000	0.593	0.000	0.000	0.443
Totals	**$242.729**	**$279.006**	**$310.013**	**$292.265**	**$372.325**
Mills/kwh or $/MWh			Electricity Equivalents		
Capital Cost incl Financing	16.15	20.10	21.29	19.06	29.08
Operations Cost	3.34	3.34	9.31	9.31	9.31
Fuel Cycle Cost	7.40	7.40	8.27	8.27	8.27
D&D Cost	0.00	0.07	0.00	0.00	0.06
Totals	**$26.89**	**$30.91**	**38.87**	**36.64**	**46.71**
Annual Production of H2 in kMt			201.982	201.982	201.982
Cost of H2 in $/kg			**$1.53**	**$1.45**	**$1.84**

- ♦ **Required extensions of Guidelines were determined, e.g.,**
  - Direct cost accounts
  - O&M cost components; better understanding needed
  - Spending profile during construction of chemical plant

ACCOMPLISHMENTS: PR & PP Evaluation

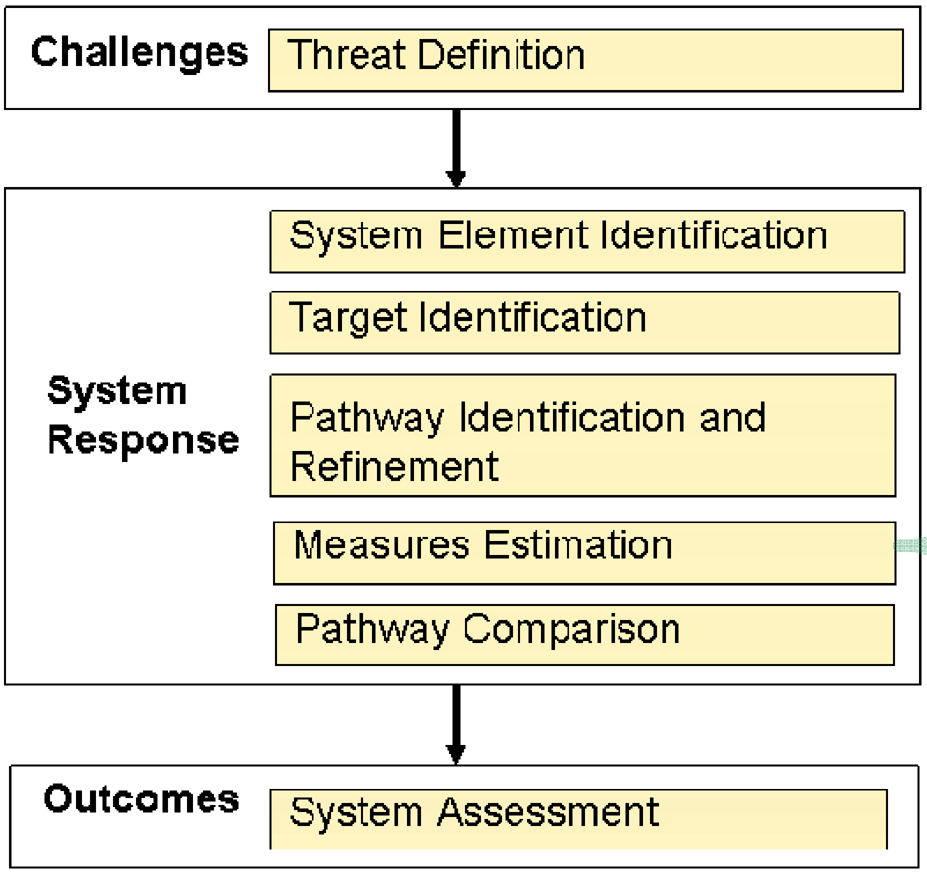
- **Formulated and documented methodology requirements**
- **Developed/adopted consensus terminology (consistent with IAEA)**
  - PR, PP, intrinsic features, extrinsic measures, …
- **Developed initial PR&PP assessment methodology**
  - Threat characteristics (actor, motivation, aspirations, capabilities)
  - High level measures to express a system's PR and PP
  - Probabilistic pathway analysis method for assessing system response
  - Progressive evaluation approach (qualitative → increasingly quantitative)

Threats	⟶	***System Characteristics***	⟶	***Outcomes***

PR&PP Measures

- **Initial methodology documented in "*Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems*"**
  - Rev. 2 issued September 30, 2004

ACCOMPLISHMENTS: PR & PP Evaluation



- **Initial PR measures**
  - **Proliferation technical difficulty**
  - **Proliferation resources**
  - **Proliferation time**
  - **Fissile material quality**
  - **Detection time (safeguardability)**
  - **Detection resources**
- **Initial PP measures**
  - **Operational accessibility**
  - **Adversary delay**
  - **Consequences and mitigation potential**
  - **Detection time**
  - **Interruption delay**
  - **Physical protection resources**

ACCOMPLISHMENTS: PR & PP Evaluation

- **Completed a "development study" for a hypothetical system—Example Sodium Fast Reactor (ESFR)**
  - Considered limited set of threats and pathways
  - Verified basic applicability of methodology
  - Refined or formalized several methodology aspects (e.g. target specification as key to pathway definition)
  - Identified methodology gaps
    - Systematic process for choosing pathways relevant to a particular threat
    - Derivation of measures from specific system characteristics
    - Aggregation of measures over multiple pathways and their segments
    - Incorporation of uncertainty in analysis
    - Prevalent subjectivity and need for expert elicitation

ACCOMPLISHMENTS: PR & PP Evaluation

- **Conducted a workshop (Nov. 2004, in Washington, D.C.) to obtain feedback from prospective users**

  Feedback highlights

  - Good progress; reasonable methodology
  - Significant challenges and questions remain
    - Adequacy of measures
    - Need for implementation details (how to apply)
    - Need for a reference or baseline
    - How to balance with other Gen IV goals
    - Prevalent subjectivity and need for expert elicitation

WORK IN PROGRESS FOR FY05: PR&PP Evaluation

- **Worked on ESFR "demonstration study" as means of further detailing methodology and addressing gaps**
  - Extend threat characteristics
  - Identify full set of pathways for threats
  - Improve the estimation of measures
    - Revisions to measures proposed, some adopted
  - Specify methods for aggregation
- **Initiated a software-based Implementation Guide for users**
  - Detailed planning underway
- **Established interface with AFCI**
  - Joint development and use of cycle cost data
  - Assess proliferation resistance of separations processes under development (e.g., UREX+)
- **Draft of Rev. 3 methodology report planned for Sep. 2005**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

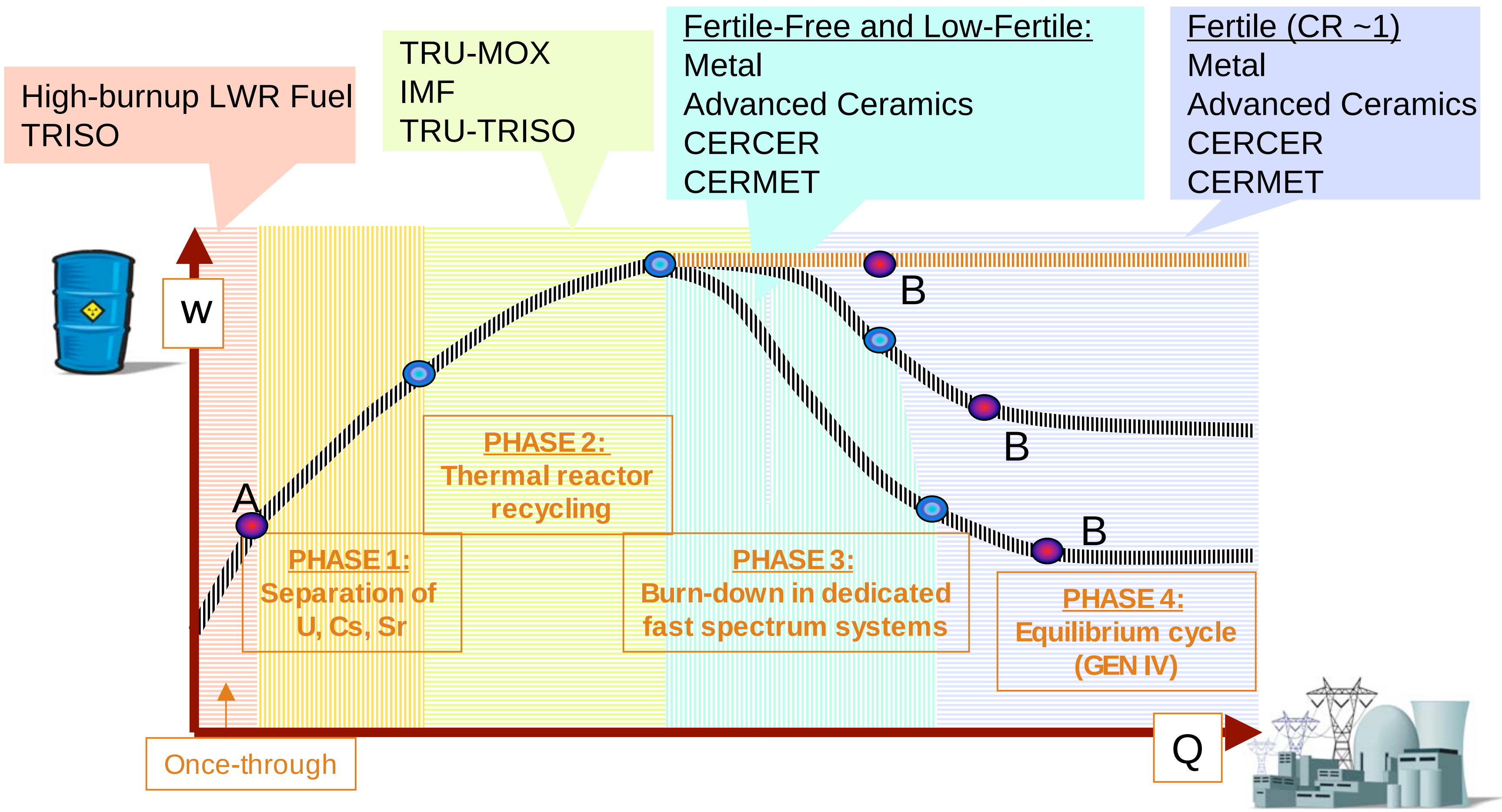
Kemal O. Pasamehmetoglu

National Technical Director (kop@inel.gov)

Advanced Fuel Development

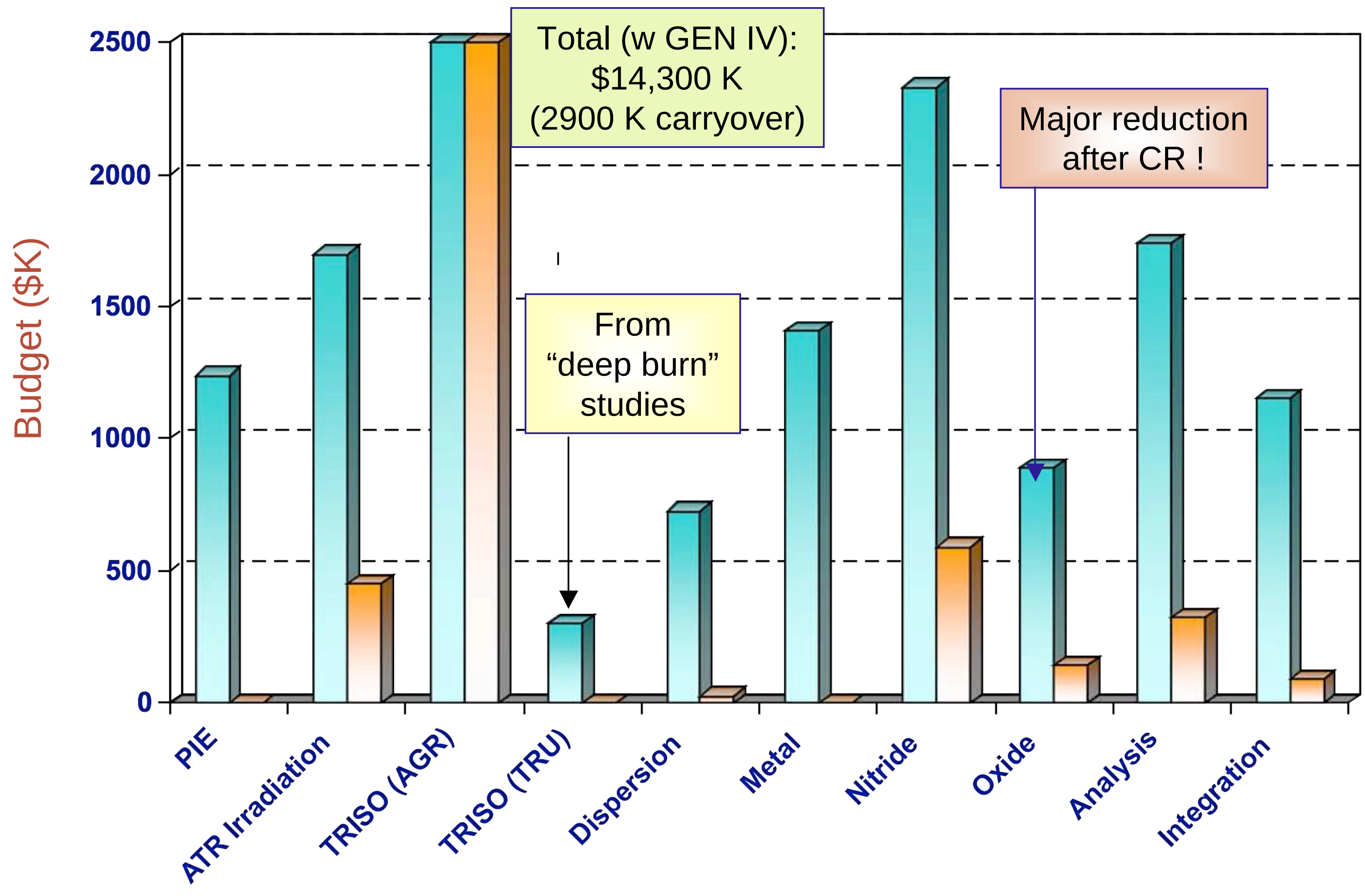
Workshop for Universities
Double Tree Hotel, Rockville,MD
June 16, 2005

The fuel development program covers fuels and cladding research for multiple phases of the fuel cycle evolution .

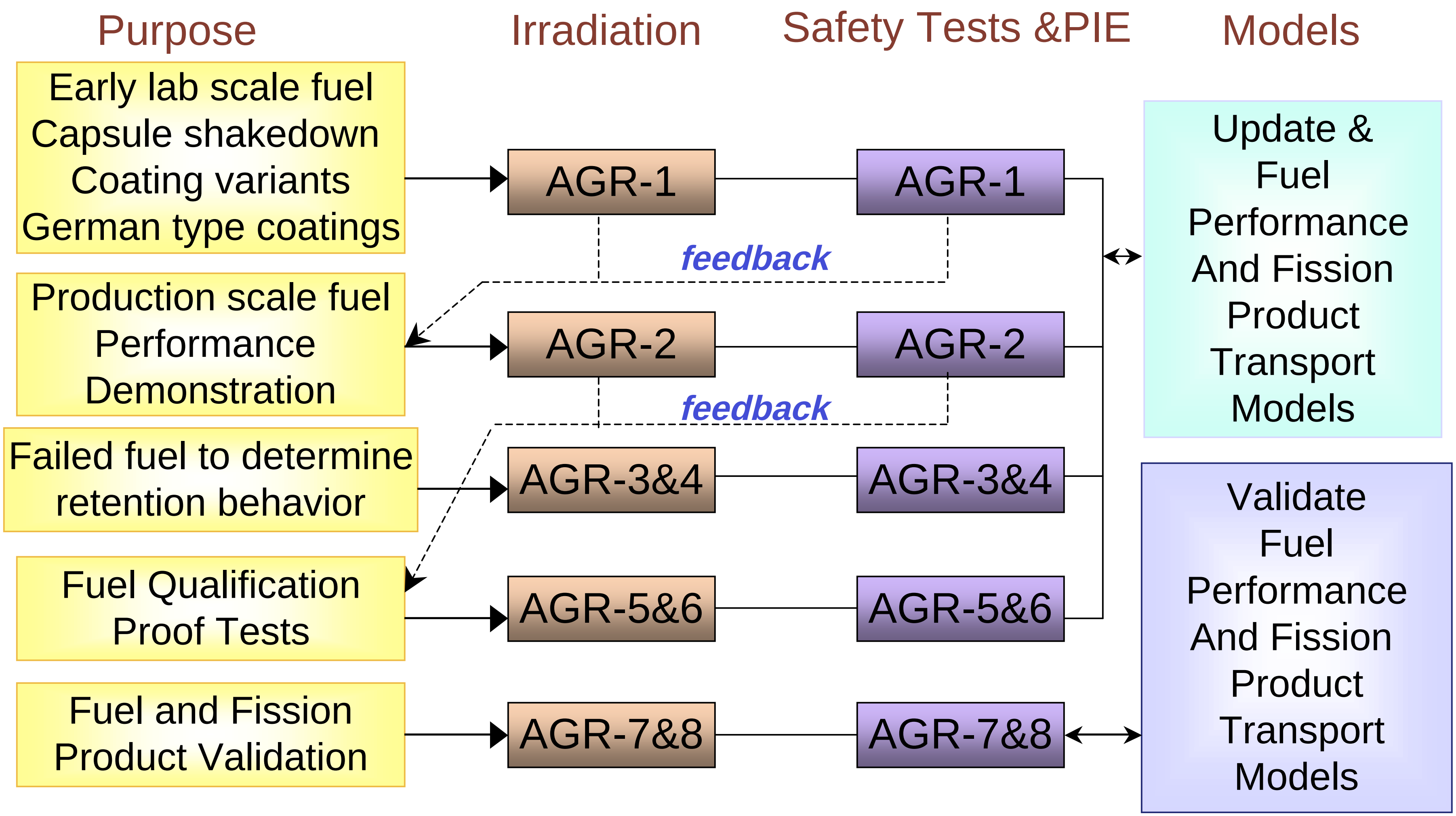


W= F(volume, radiological risk, short-term heat load, long-term heat load,plutonium mine)

In FY05, the total budget of fuel development activities is $26.2 M, including $4.4 M carryover



Overview of AGR Program Activities



Contact for additional info: Dave Petti at David.Petti@inl.gov

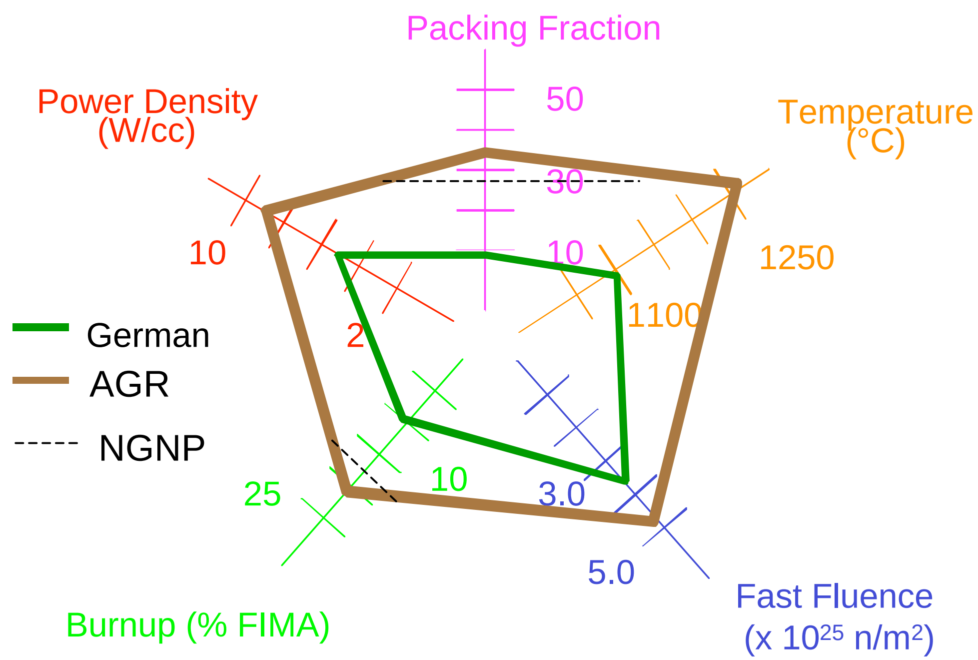
AGR Program Irradiations

Capsule	Task/Purpose	Cells	Location
AGR-1	Shakedown and early fuel - confirm understanding from historical database and provide feedback to fabrication	multi	large - B
AGR-2	Performance test fuel - provide feedback to fabrication for a large coater (6”)	multi	large - B
AGR-3	Fission product transport - 1	multi	large - B
AGR-4	Fission product transport - 2	single	small - B
AGR-5	Fuel qualification - 1 - statistics important	multi	large - B
AGR-6	Fuel qualification - 2 - statistics important	multi	large - B
AGR-7	Fuel performance model validation	multi	large - B
AGR-8	Fission product transport - 3	multi	large - B

Contact for additional info: Dave Petti at David.Petti@inl.gov

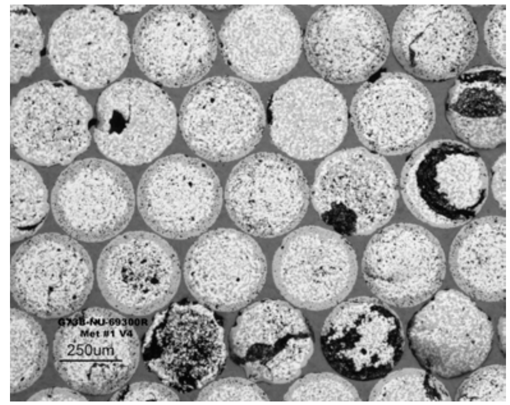
Comparison of Fuel Service Conditions: The Challenge for VHTR

- Germans qualified UO_2 TRISO fuel for pebble bed HTR-Module
  - Pebble; 1100°C, 8% FIMA, 3.5 x 10^{25} n/m^2, 3 W/cc, 10% packing fraction
- Japanese qualified UO_2 TRISO fuel for HTTR
  - Annual compact; 1200°C; 4% FIMA, 4x10^{25} n/m^2, 6 W/cc; 30% packing fraction
- Eskom RSA is qualifying pebbles to German conditions for PBMR
- Without an NGNP design, the AGR program is qualifying a design envelope for either a pebble bed or prismatic reactor
  - 1250°C, 15-20% FIMA, 4-5x10^{25} n/m^2, 6-12 W/cc, 35% packing fraction
  - UCO TRISO fuel in compact form

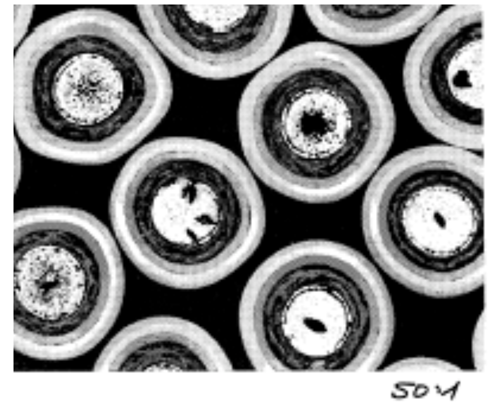


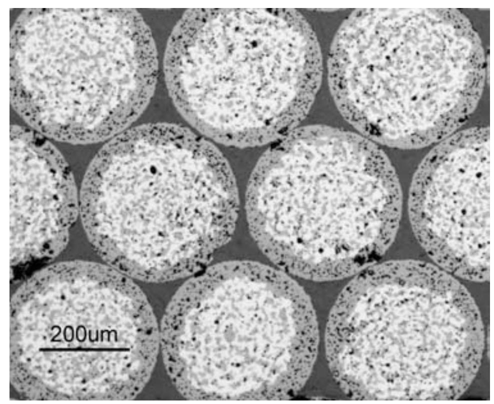
UCO Kernel Development at BWXT

- Specification requires
  - C/U = 0.5 ± 0.2 , O/U=1.5 ± 0.2, r > 10.4 g/cc
- 5 kg NUCO kernels fabricated for coating tests in FY-04
  - C/U = 0.4, O/U=1.3, r = 10.7 g/cc
- Chemistry improvement program initiated to increase O/U ratio while maintaining C/U ratio
- Development effort then focused on improving carbon dispersion using ultrasonics to obtain higher density ($\geq$ 10.5 g/cc)
- LEU UCO kernels for AGR-1 irradiation are complete.

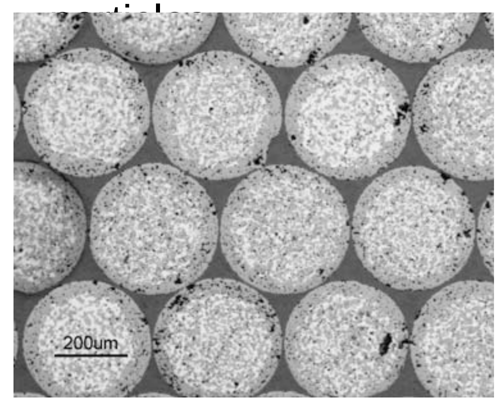


Initial NUCO kernels


German UCO



NUCO kernels from AGR chemistry improvement activities

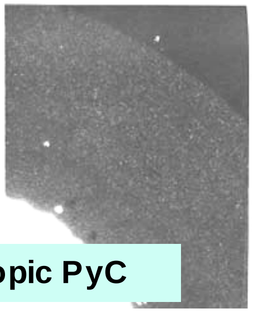


LEUCO kernels for AGR-1

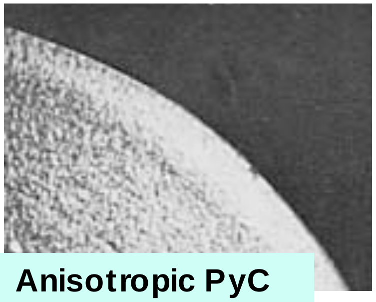
Key Differences between German and US fuel

- Coating rate used to make PyC (affects permeability and anisotropy of layer; US is low which reduces permeability and increases anisotropy; German is high which reduces anisotropy and decreases permeability)
- Nature of the interface between SiC and IPyC (German fingered interface is strong and US is weak which causes debonding)
- Microstructure of SiC (German is small grained and US is large columnar grained; difference is largely due to temperature used during SiC coating step)

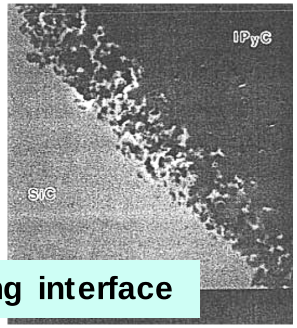
German	US



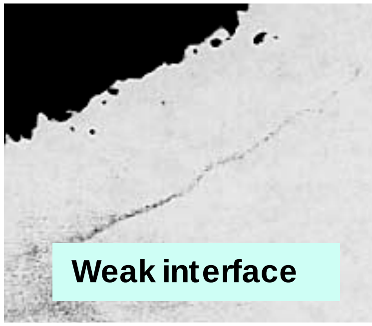
Isotropic PyC



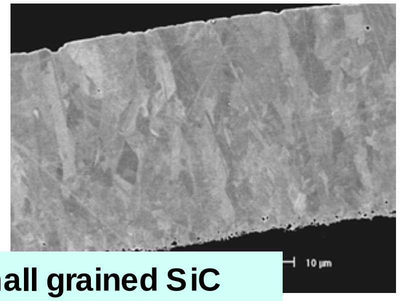
Anisotropic PyC



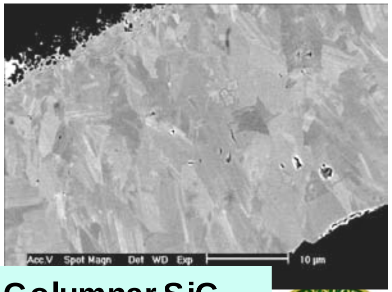
Strong interface



Weak interface



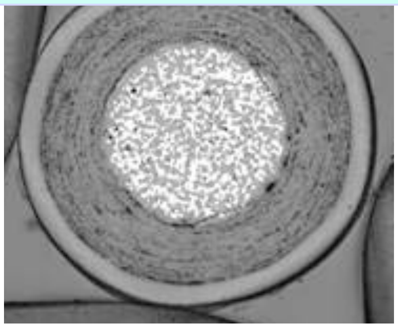
Small grained SiC



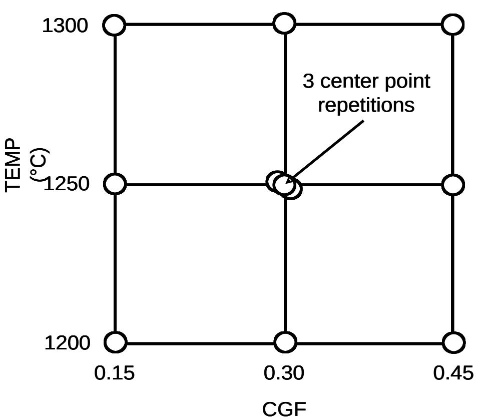
Columnar SiC

Final optimization of coater conditions using NUCO is underway prior to producing fuel for AGR-1

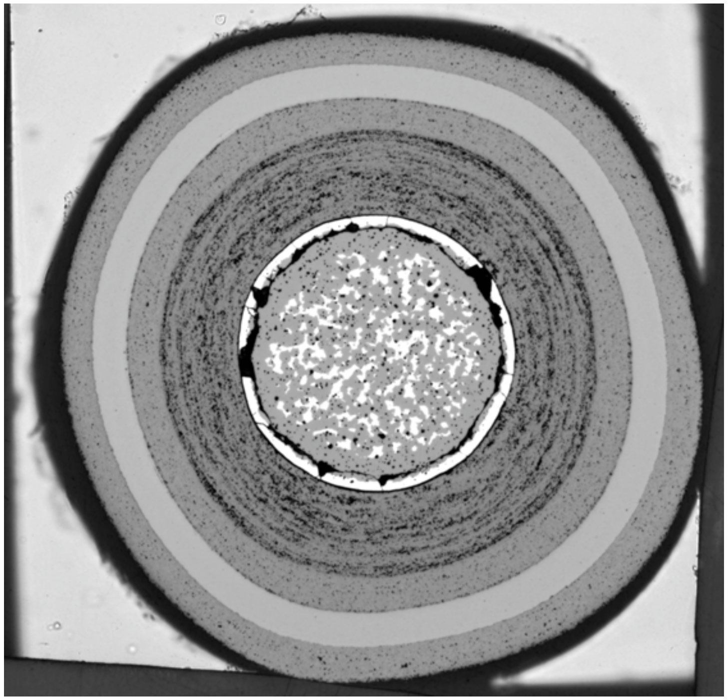
IPyC parametric study to determine baseline and variant processes for irradiation



Buffer/IPyC on 350 µm NUCO



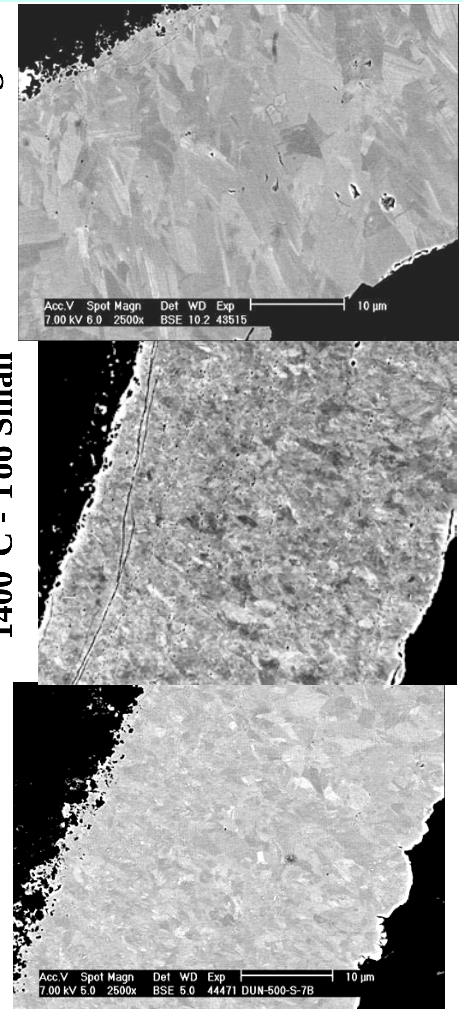
TRISO Layers Have Been Deposited on 350 µm NUCO Kernels



SiC grain size adjustments

1510°C - Too Big

1400°C - Too Small

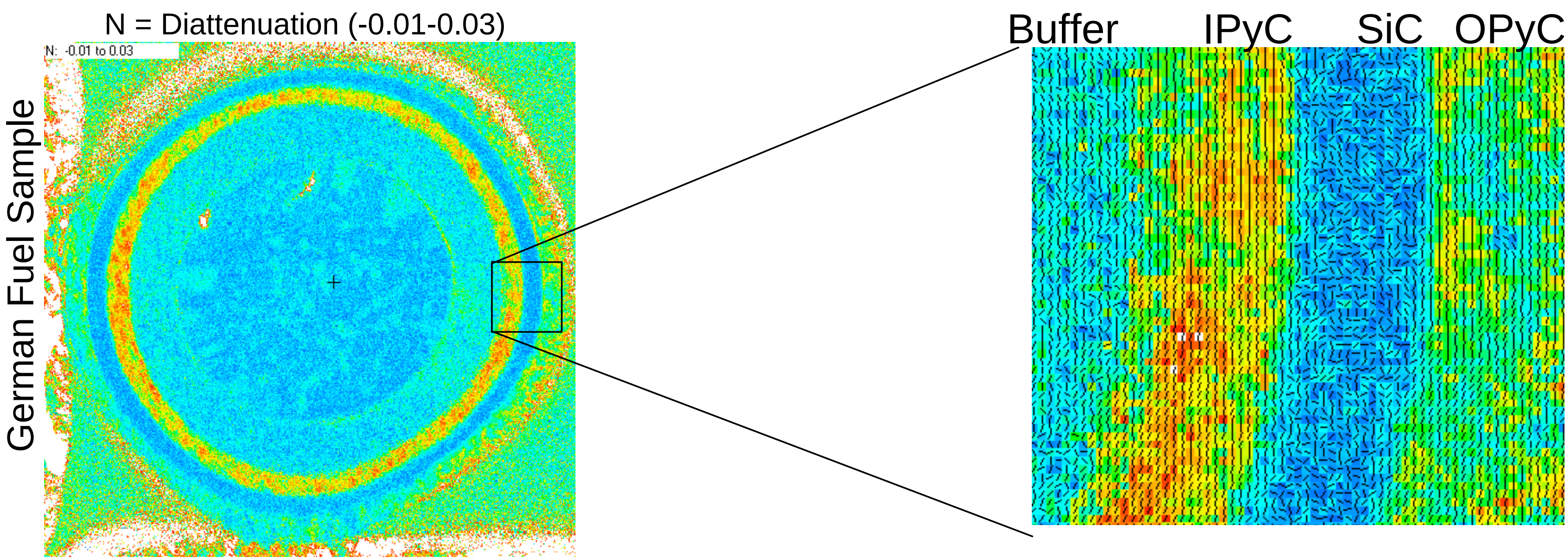


Example:1400°C with Ar dilution

Improved Pyrocarbon Anisotropy Measurement

- **Improved best resolution to ~4 micron (2 micron pixel size).**
- **Improved data analysis to calculate the standard deviation of the principal angle from the normal to the radial direction.**
- **Observed non-uniformity in the diattenuation of the IPyC.**
- **Measured crystalline rutile (TiO_2) and calcite ($CaCO_3$) using the 2-MGEM obtaining expected diattenuation.**
- **Improved calibration resulting in improved accuracy in diattenuation.**

N = Diattenuation (-0.01-0.03)



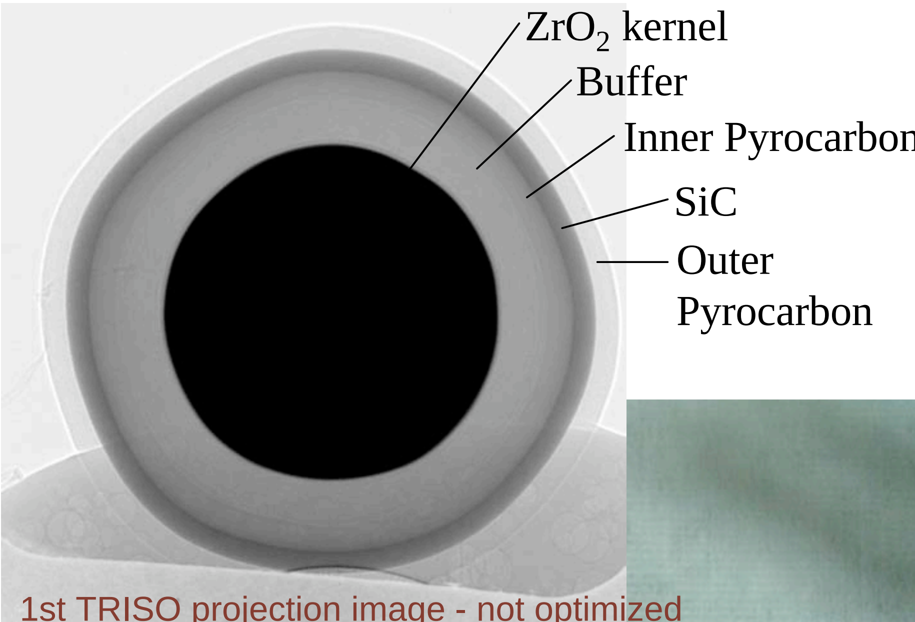
Pixel Size: 2 μm

OAF ≅ 1+2N = 1.028 for German IPyC

Diattenuation + Direction of Principal Axis

Line segments indicate the preferred fast axis (ab-plane) orientation

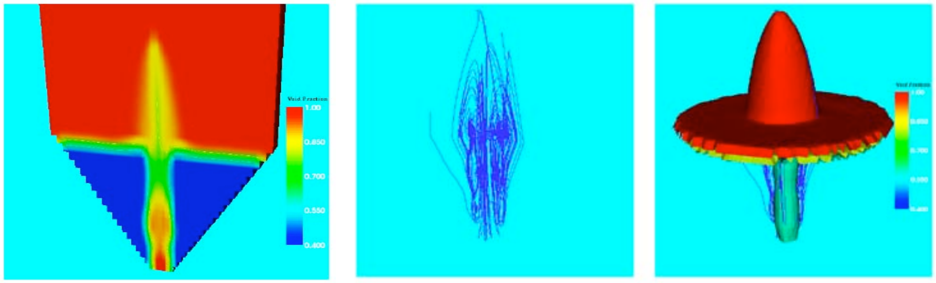
ORNL Micron-resolution x-ray analysis



1st TRISO projection image - not optimized

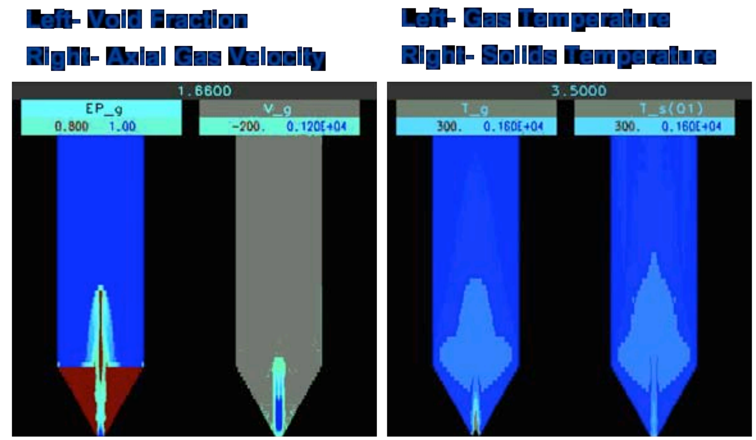
- **Delivered 10 Jan 05**
- **Installation and staff training underway**
- **1.6 µm resolution**
- **Tomographic capability**

FBCVD Process Model Development



- 3D simulations reveal non-symmetric circulation oscillations observed in the experiments including sloshing, spout rotation
- Such oscillations may broaden the range of local particle conditions
- 3D simulations must be used carefully because they are much more time consuming (~10 times longer on 4-8 processors)

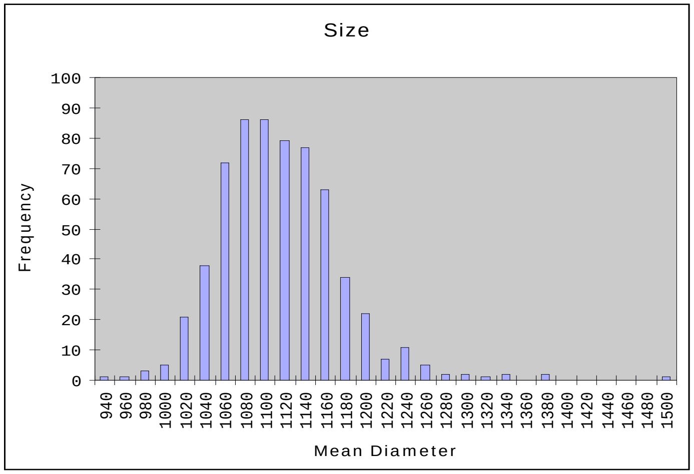
- Added tracer capability to study the particle path during coating
- Examining impact of varying cone angle
- Using UT-Knoxville mockup to validate model



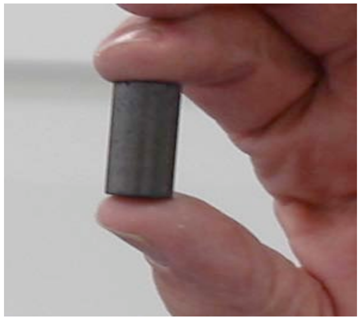
- Preliminary runs of the high-temperature 50-mm surrogate coater
- Inlet gas temperature at the start of the cone is assumed to be 25°C (ambient) and the walls are at 1300°C
- Gas velocity goes negative away from the centerline due to drag from falling particles
- Inlet gas heats very quickly from ambient to the furnace temperature unlike pure gas flow

Progress in Overcoating and Compacting

- Improved control and yield of overcoating process
  - -can consistently overcoat surrogates to desired overcoat thickness (~165 μm)
  - -efficiency in overcoating a batch of surrogates increased from 50% to ~92%
  - -designing two stage overcoating process where stage 1=overcoating; stage 2=sizing/smoothing
- Developed warm pressing process methodology
  - -35% packing fraction has been achieved
  - -compact characterization on-going in order to determine the quality of these compacts
  - -higher packing fractions may be possible

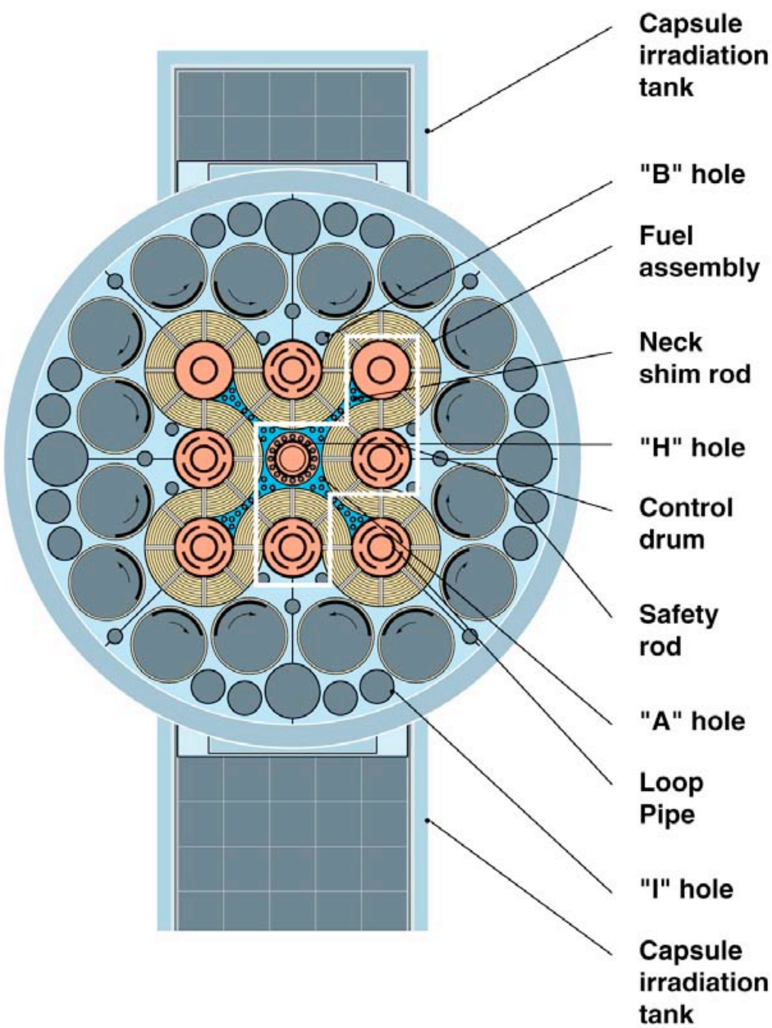


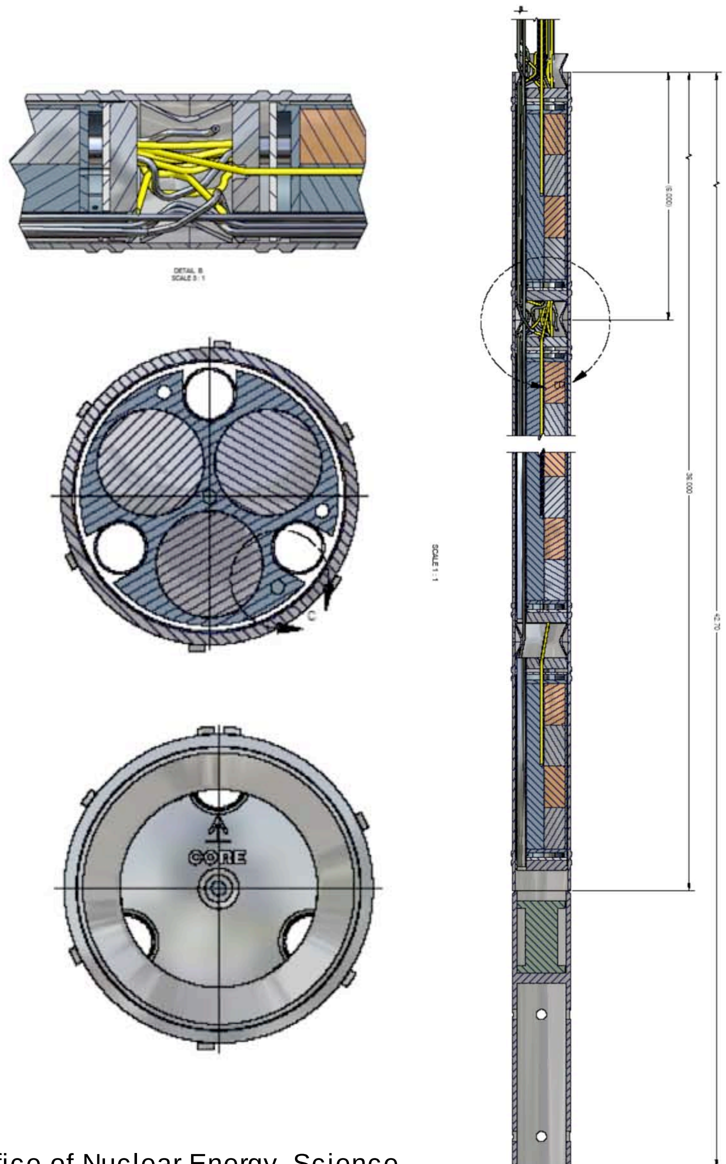
Size data showing average mean diameter of overcoated particles ≅ 1100 μm, which corresponds to an overcoat thickness of 165 μm



Carbonized compact with 35% packing fraction

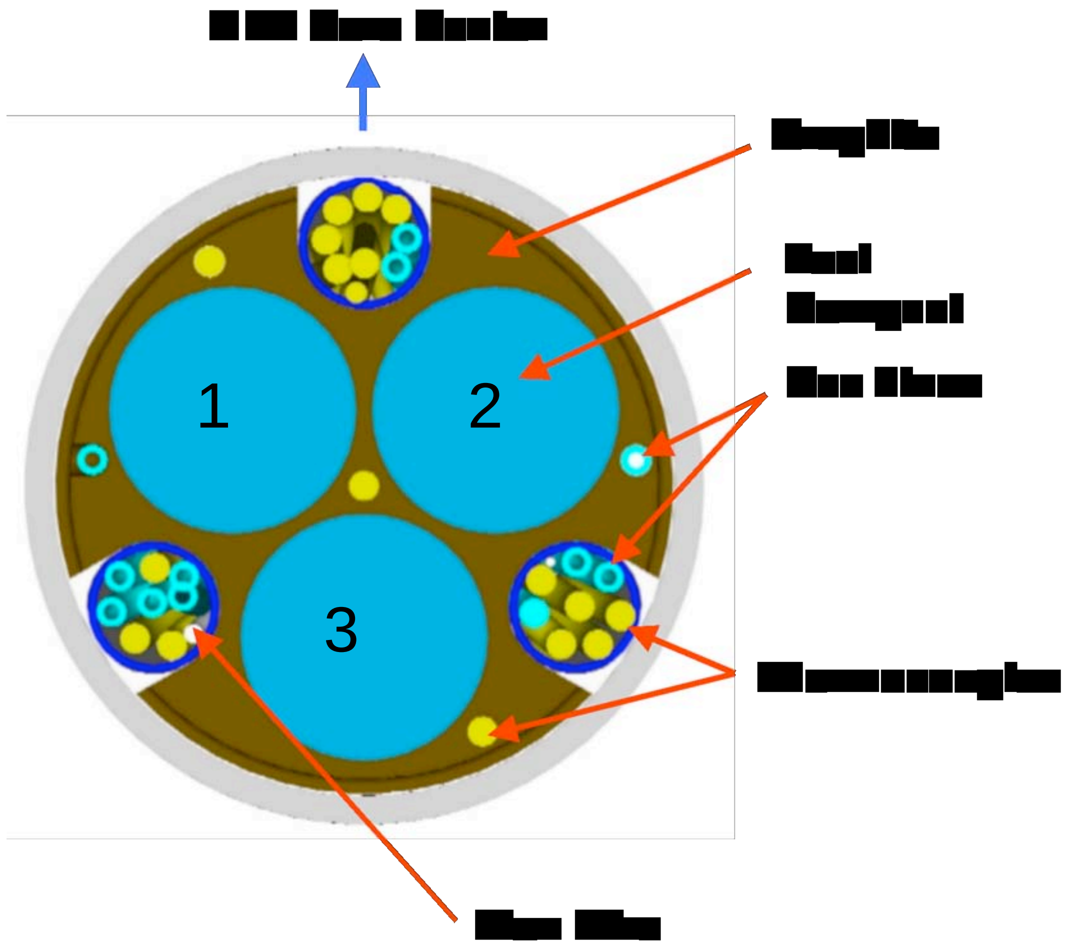
The AGR irradiations will be conducted in the large B positions in ATR at the INL





AGR-1 Fuel Irradiation Capsule

- Six compartments. Hafnium outer shroud and boronated graphite bodies to manipulate flux and power in fuel over life
- 12 one-half inch diameter, one inch long compacts per compartment
- 3 fuel compacts per level encased with graphite containing B_4C (5 to 7 wt%)
- 3 thermocouples per capsule for temperature monitoring and control
- Flux wires for fast and thermal flux monitoring
- Gas gap between graphite and capsule shell sized to provide temperature control



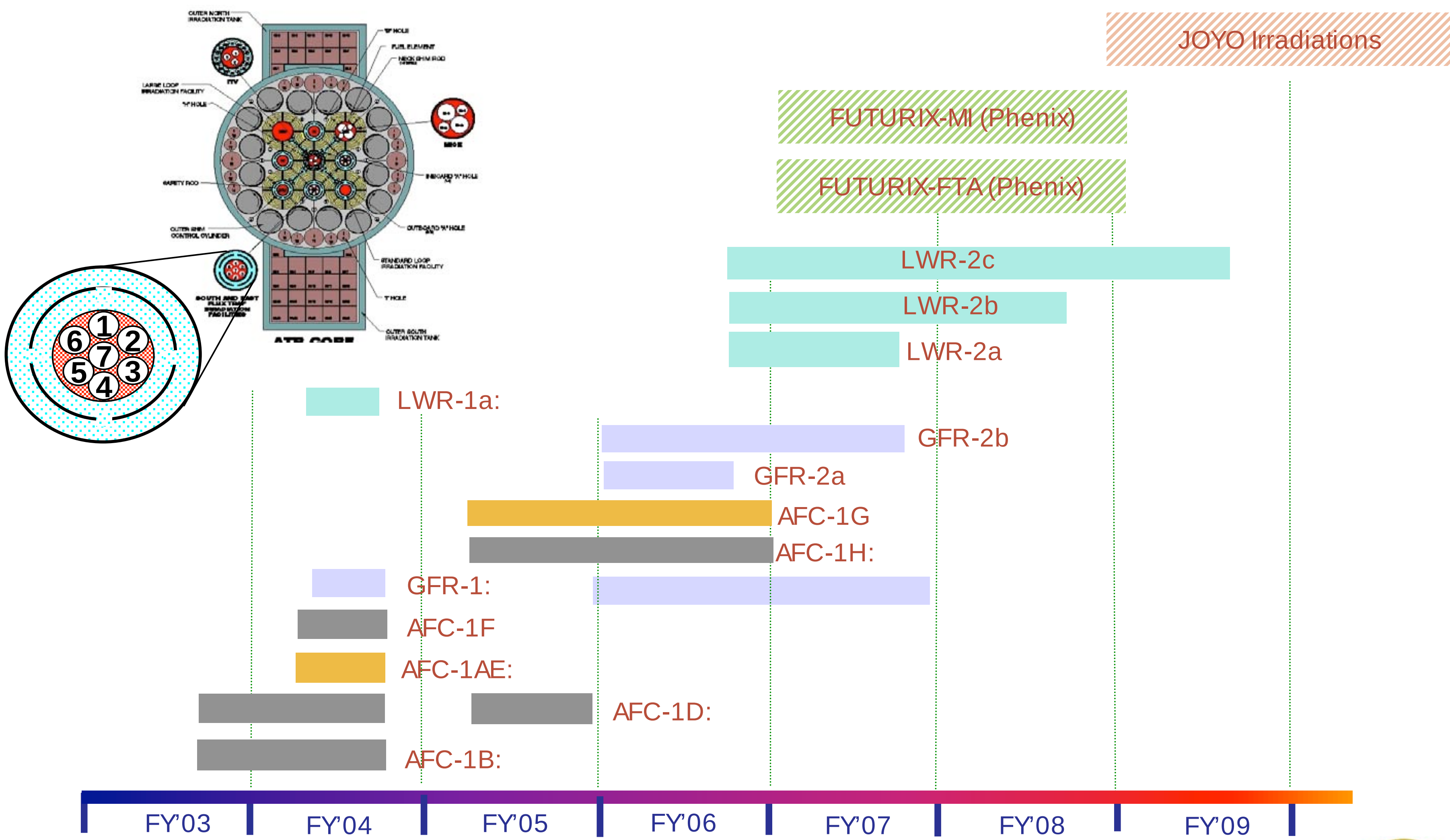
Fuel Performance Modeling: PARFUME Capabilities

Structural	Service Conditions	Physio-chemical Models	Layer Interactions	Failure Evaluation
Intact particles Cracked layers Debonded layers Faceted particles	Any user specified temperature, fluence, burnup history Improved thermal model for element and particle Accident conditions	Booth equivalent sphere fission gas release using Turnbull diffusivities HSC thermo-dynamic based for CO production for any fuel composition Redlich-Kwong EOS Fission product transport across each layer	Amoeba effect Fission product SiC interactions (e.g. Pd, Cs) Thermal Decomposition	Monte Carlo based statistical sampling Direct numerical integration

Some specific University research areas of interest for the TRISO fuel development are

- Evaluation and Development of Thermocouples for High Temperature Nuclear Environments in the Advanced Test Reactor, Idaho National Lab.
- Transport behavior of Fission Products in TRISO Coated Particles
- Methods to evaluate and limit fission product plating in ATR fission product monitoring system tubing
- Thermochemical Analysis of Multi-Composition Kernels for Coated Particle Fuel
- Irradiation Creep Measurements of Pyrolytic Carbon
- Evaluation of natural graphite properties after adsorption of fission products
- Compacting development evaluation and innovative methods

PLANNED IRRADIATIONS FOR TRANSMUTATION FUELS



Metallic Fuel Development

AFC-1 metal fuel fabricated and characterized (U, Pu, Am, Np, Zr)

FUTURIX-FTA R&D report and fuel fabrication report completed.

FUTURIX-FTA samples are being fabricated

AFC-1G metal fuel fabrication

Metal fuel thermal characterization successful

FCI work ongoing

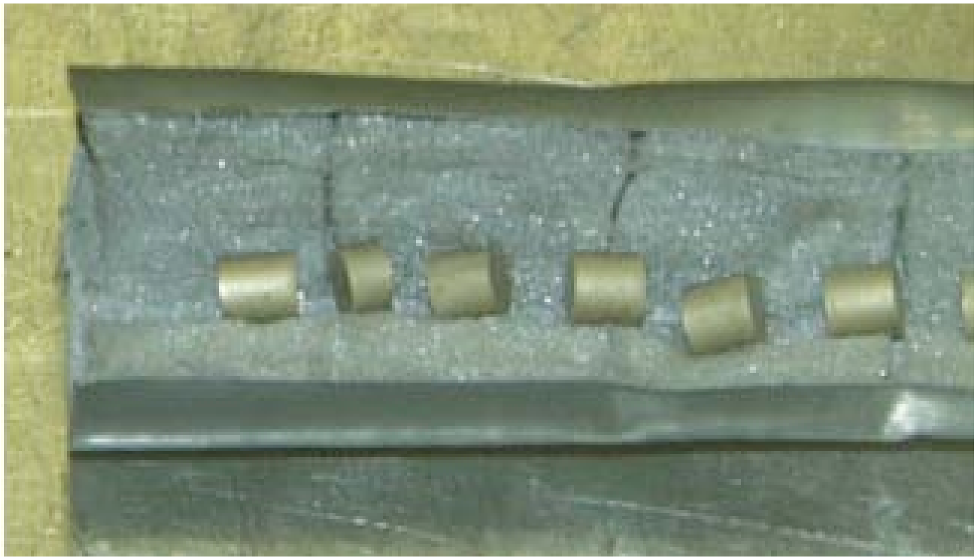
Nitride Fuel development

Nitride process testing completed and documented

AFC-1G pellets fabricated and characterized (6 rodlets)

FUTURIX FTA pellet fabrication continues

The thermal properties measurements will be repeated

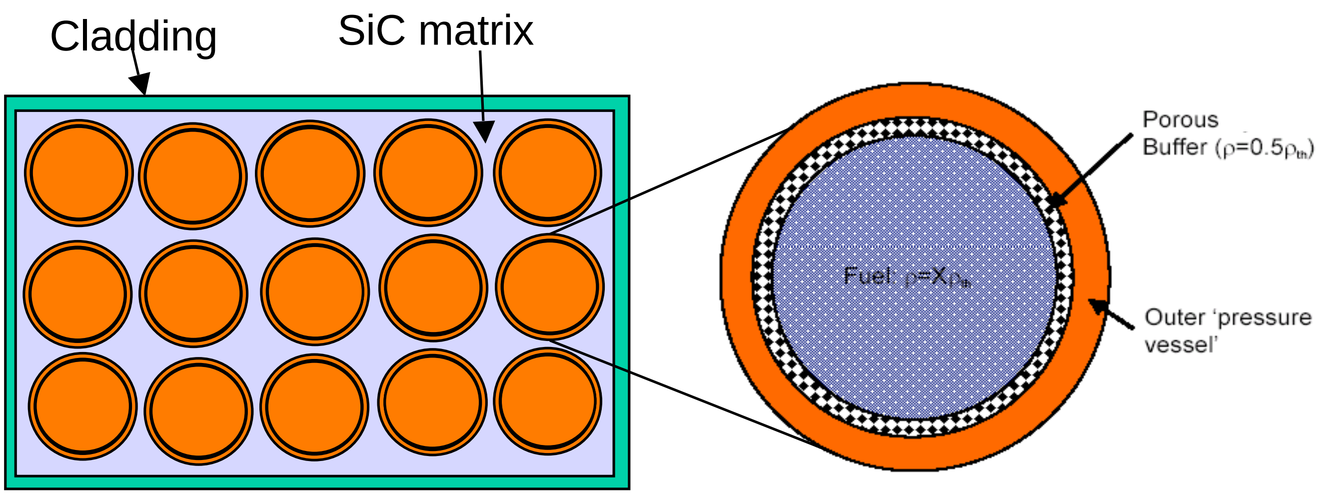


GFR Dispersion Fuel Development

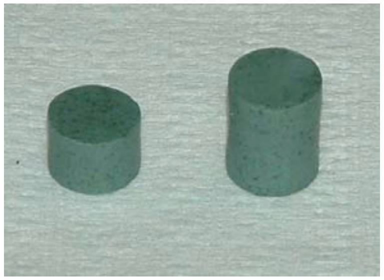
Matrix and particle fabrication studies continue

FUTURIX-MI sample fabrication is on schedule

U bearing dispersion fuel fabrication and characterization continue

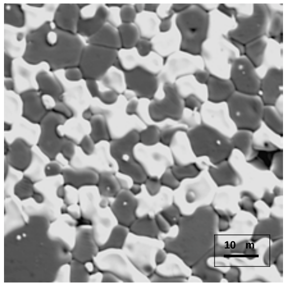


Inert Matrix Fuel Development



Zirconia-magnesia mixture for matrix is characterized

Pu bearing micro-dispersion fuel fabricated

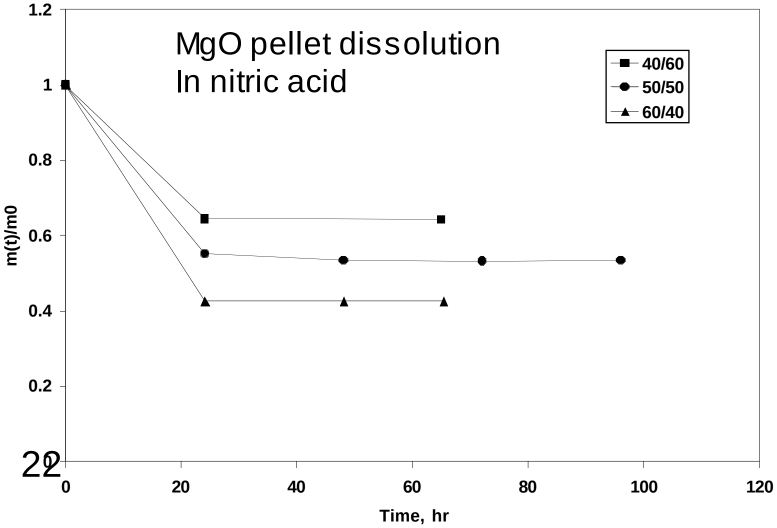


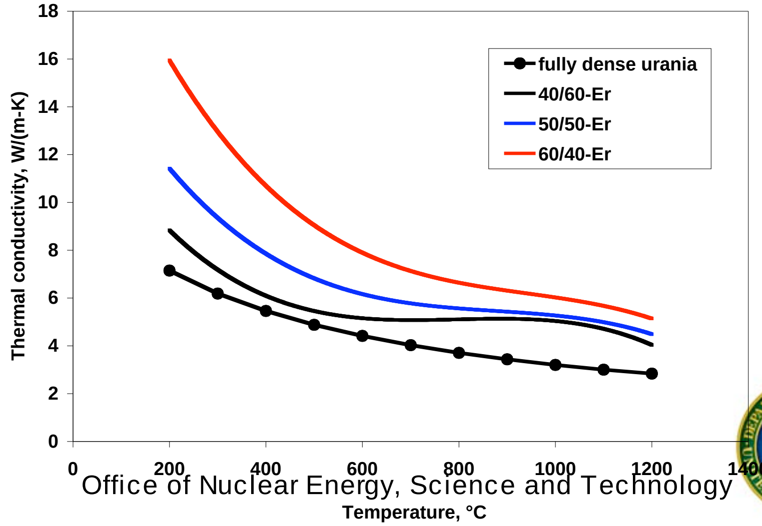
50/50

- White phase: $Mg_{0.160}Zr_{0.840}O_{1.840}$
- Dark phase: MgO
- Grain size 10-20 μm

MgO pellet
after 3 hr in boiling water

MgO-ZrO_2 pellet
after 700 hours in de-ionized water at 300ºC, saturation pressure



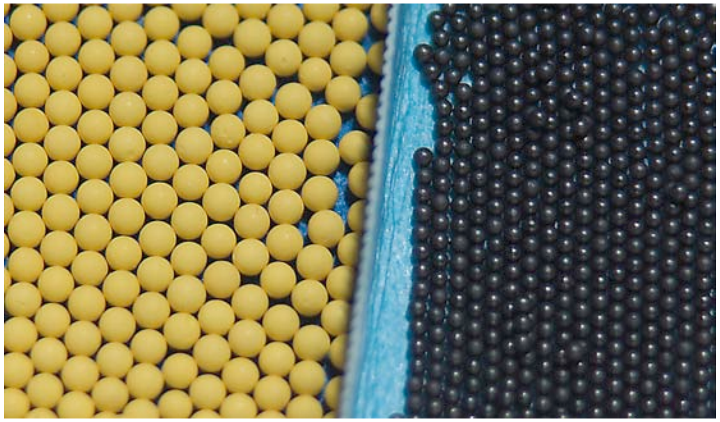


Sphere-Pac Fuel Development

Cold process work is ongoing for sphere preparation and pin loading

Fuel fabrication with surrogate

Completion of dry-run sphere-pac rodlets (9/1/05)



Advanced Fuel Performance Code

- Fission products kinetics and concentration
- Heat transfer simulations
- Diffusion of species (gas and fission products) simulations
- Chemical reactions simulations

Incorporation of the Advanced Models

Advanced Models and Simulations

- Continuum-scale: Thermo-mechanical properties
- Meso-scale: Microstructural evolution, Species mobility
- Atomic-scale: Defect formation free energy, Irradiation effects,
- Electronic Structure: Structural stability, elastic constants

FY05: MODELS AND SIMULATIONS OF OXIDE FUELS

Perform Simulation of Heat Transfer in Oxide Fuel Rods Using Atomistic Models.

a) Electronic structure:
- Calculate energies of formation of defects in UOx.
- Calculate the phonons of U and UOx.
- Calculate migration energies of key defects in UOx.

b) Atomistic:
- Calculate thermal conductivity in UOx system as a function of x and T, using literature pair potentials.
- Calculate diffusivity of O in UOx using literature pair potentials.

c) Integration:
- Develop kinetic lattice Monte Carlo (KLMC) model of the long term ageing evolution of point defects in UOx.
- Calculate non-stoichiometry as function of temperature and O partial pressure.
- Develop integrated thermal conductivity model of UOx, including effects of temperature, non-stoichiometry, and porosity.

d) Simulations:
- Update thermal conductivity model in FRAPCON and TRUCHAS.
- Run FRAPCON and TRUCHAS simulations of UOx fuel rods.

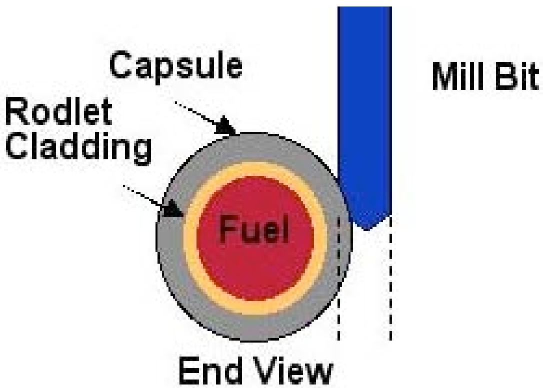
Transmutation Fuel PIE

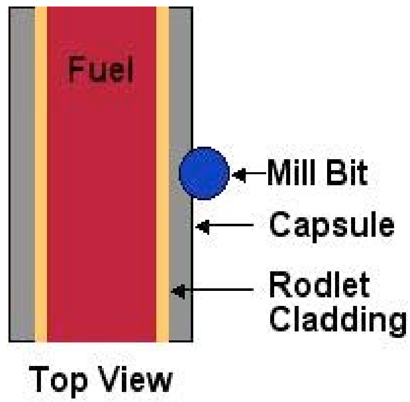
Capsules radiographed

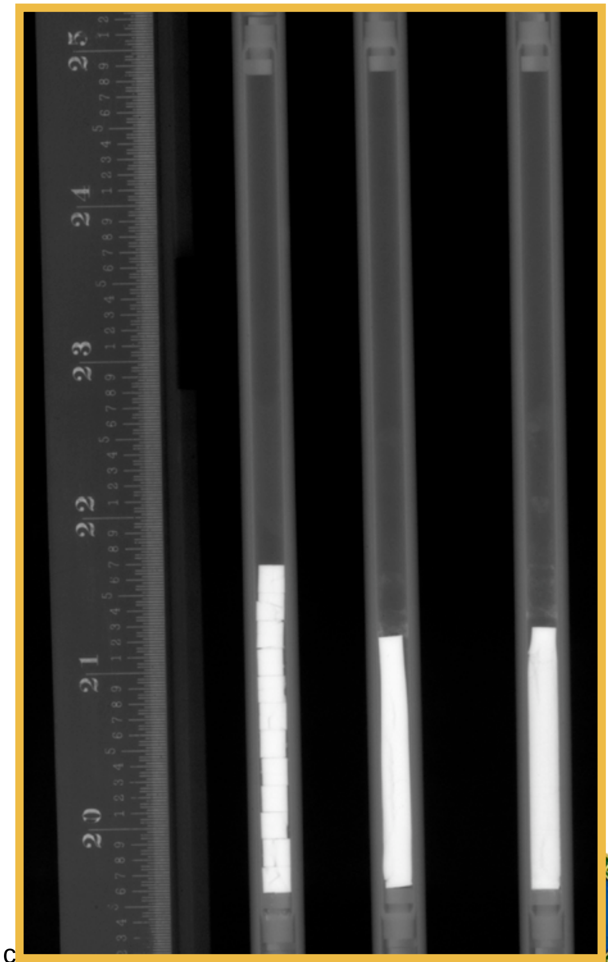
All rodlets removed from capsules

Gamma scans finished. Proliferometry ongoing.

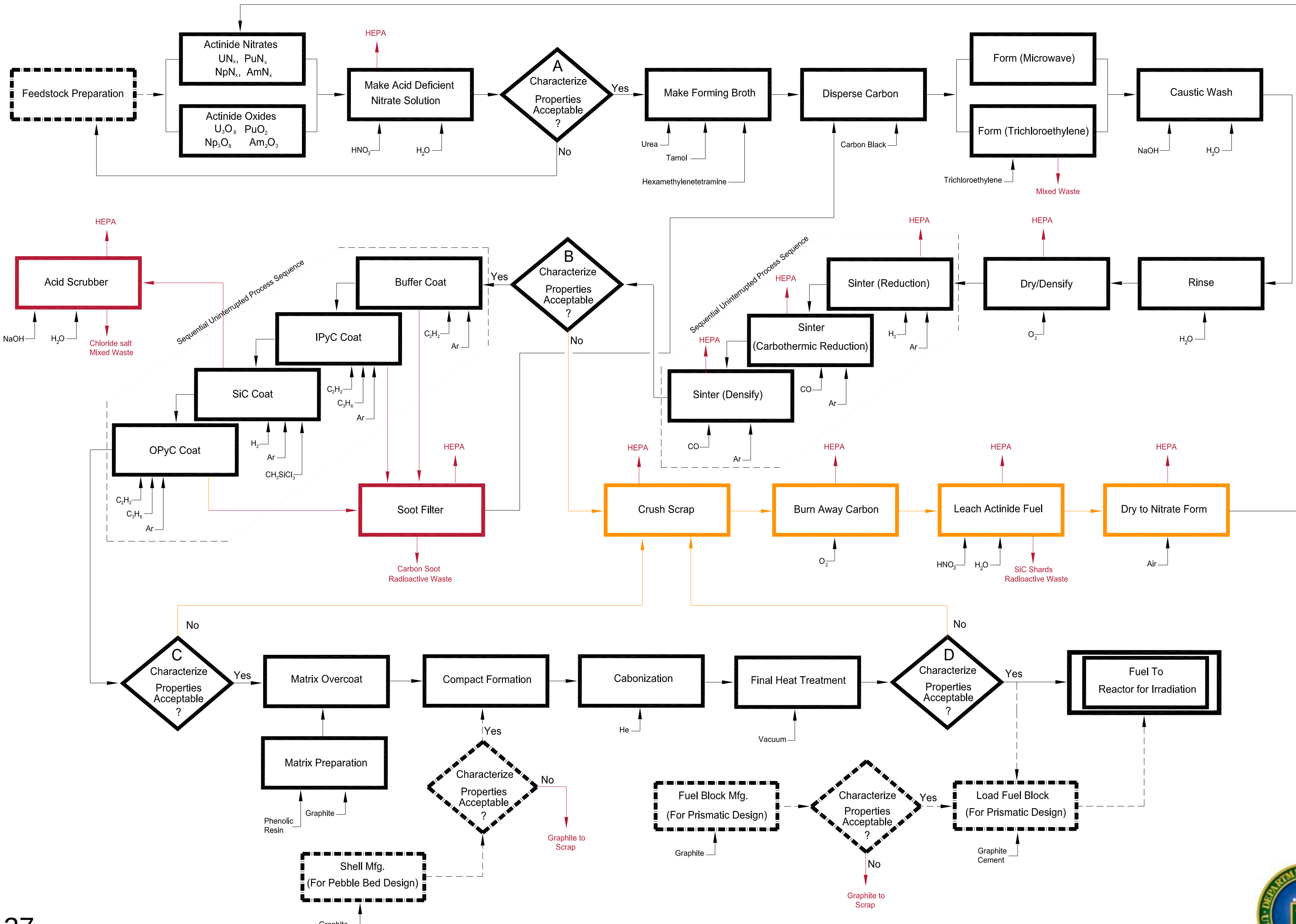
Destructive exam will be completed by Sept. 05







TRISO MOX Process Flow Sheet



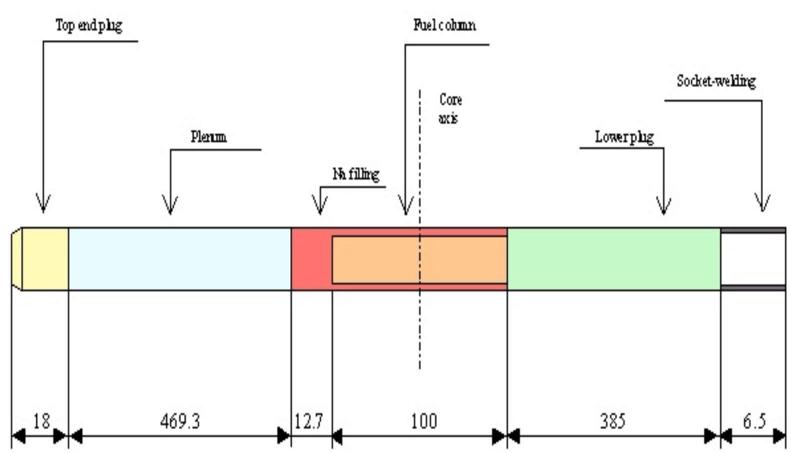
International Collaboration

FUTURIX-FTA

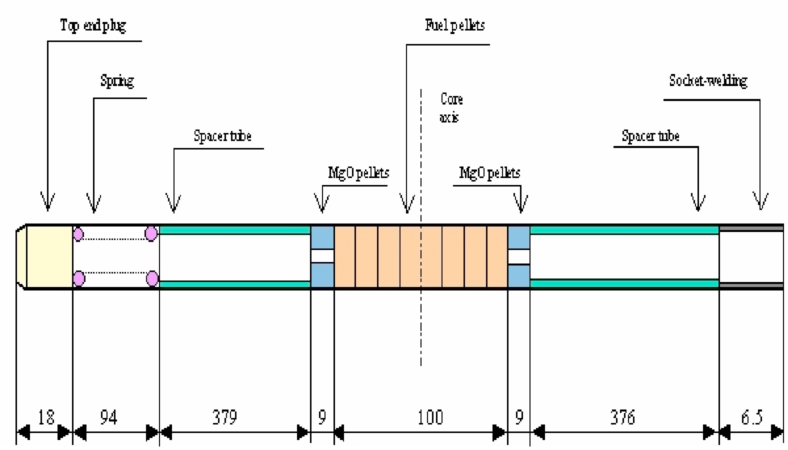
Non-Fertile Fuels	Bond	Fabricator
(48)Pu-12Am-40Zr	Na	ANL
$(Pu_{0.50}, Am_{0.50})N$ + 36-wt%ZrN	Na	LANL
$(Pu_{0.20}, Am_{0.80})O_2$ + 65-vol%MgO	He	CEA
$(Pu_{0.50}, Am_{0.50})O_2$ + 70-vol%MgO	He	CEA
$(Pu_{0.23}, Am_{0.25}, Zr_{0.52})O_2$ + 60-vol%Mo^{92} He	ITU	
$(Pu_{0.50}, Am_{0.50})O_2$ + 60-vol% Mo^{92}	He	ITU
Low-Fertile Fuels		
(35)U-29Pu-4Am-2Np-30Zr	Na	ANL
$(U_{0.50}, Pu_{0.25}, Am_{0.15}, Np_{0.10})N$	Na	LANL

19-pin experimental subassembly to be used
- 8 experimental pins
- 11 standard Phénix driver pins or dummies

Irradiation in Phénix ring 3 or 4
- Satisfies LHGR limit of 350 W/cm



Na-bonded fuels



He-bonded fuels

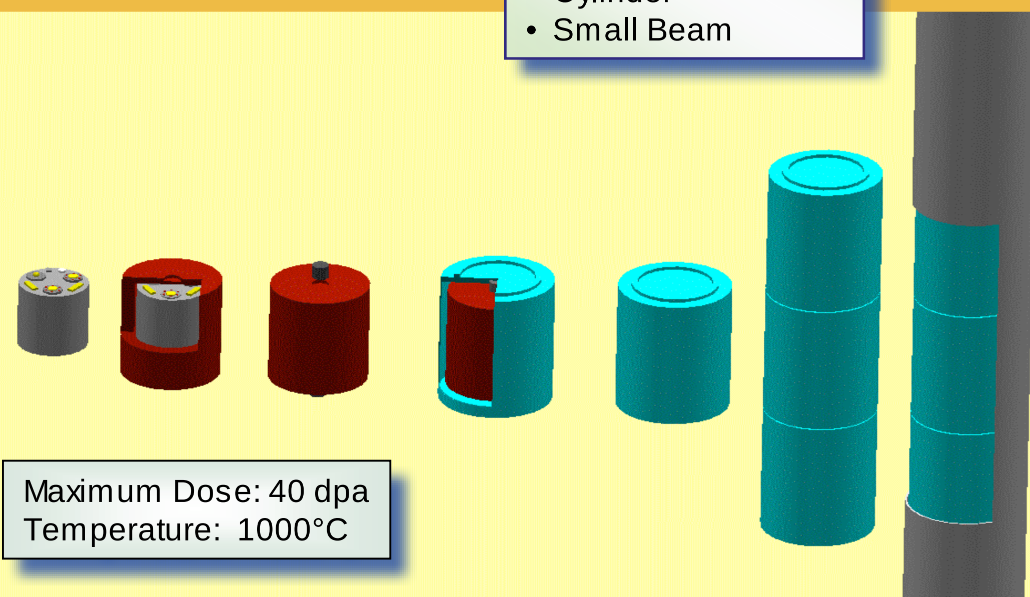
FUTURIX-MI

EFFECT OF IRRADIATION ON PHYSICAL AND CHEMICAL PROPERTIES

•DENSITY (SWELLING)
•MICROSTRUCTURE
•COMPOSITION - CRYSTALLOGRAPHIC PHASES
•ACTIVATION
•THERMAL PROPERTIES : Thermal Conductivity,Thermal Diffusivity, Heat Capacity, Linear Expansion
•MECHANICAL PROPERTIES : Young Modulus, Poisson ration, Hardness, Strength
• ELECTRICAL RESISTIVITY

SAMPLES

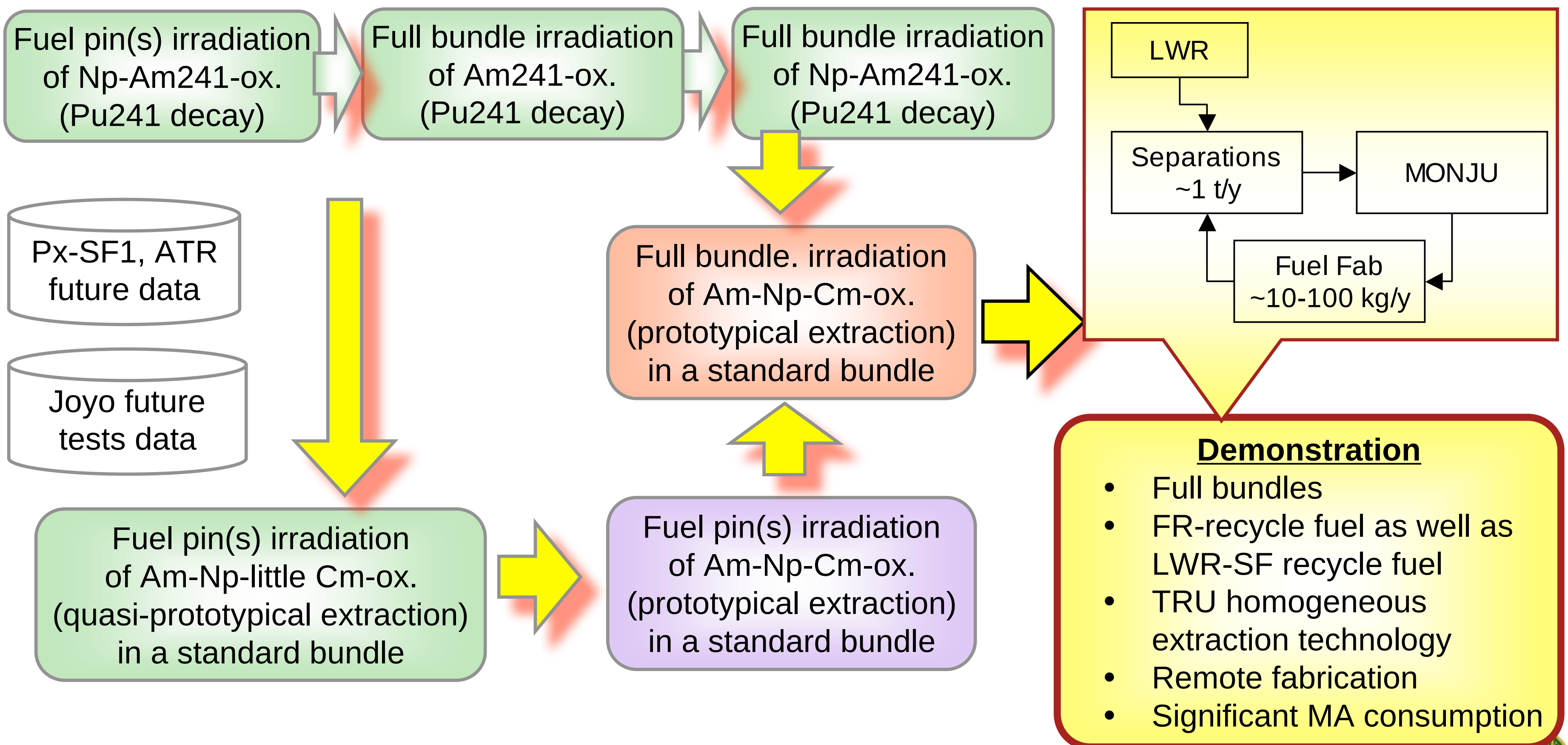
- Small Disk
- TEM Specimen
- Cylinder
- Small Beam



Maximum Dose: 40 dpa
Temperature: 1000°C

International Collaboration

MONJU Global MA Management



Joint Irradiation in JOYO

- **Negotiations with JNC started in February**
- **Proposed test matrix by DOE & CEA**
  - *(1) oxide for support of MONJU MA fuel irradiation*
  - *(2) oxide for transmutation feasibility*
  - *(3) nitride for transmutation feasibility*
  - *(4) metal for transmutation feasibility*
  - *(5) dispersion carbide fuel including GFR application*
  - *(6) dispersion nitride fuel including GFR application*
  - *(7) inert matrices and materials ranging from 500℃ to 1000℃*

I-NEERIs and U-NEERIs

I-NERIs

- **LWR- Inert Matrix Fuels**
  - AECL/Canada
  - ITU/EC
- **GFR-Dispersion Fuels**
  - CEA/France (FUTURIX-MI)
  - ITU/EC
- **Nitride fuels**
  - ITU/EC
- **Transmutation Fuels**
  - CEA (FUTURIX-FTA)/ France
  - JNC-JEARI/ Japan
    - Joint JOYO irradiation
  - Tri-lateral collaboration among DOE-CEA-JNC on “Global Minor Actinide Management using MONJU”
  - AFTRA under EUROTRANS (EC)
- **Fuel Performance Modeling**
  - ITU/EC
  - JNC-JEARI/Japan
  - CEA/France

Value to the US programs
~$75-100 M over 5 years

U-NERIs
(FY04 & FY05 Awards)

- **LWR-Inert Matrix Fuel**
  - University of Florida
- **Nitride & Oxide Structures**
  - Arizona State University
- **Fuel Modeling**
  - University of Florida
- **Nitride Fuels**
  - University of Florida
- **Am-bearing Fuels**
  - Colorado School of Mines
- **TRISO Fuel Coating Technology**
  - University of Tennessee
  - Iowa State University
- **On-line FP monitoring (AGR)**
  - North Carolina State

~$4 M over 3 years

Some specific University research areas of interest for the transmutation fuel development are

- Fabrication process development for TRU bearing fuels
  - Process optimization & correlation of process parametres with irradiation performance parameters
  - Process modeling
  - Low temperature or low heat fabrication techniques for Am bearing fuels
- Fuel characterization techniques
- Design of separate effect irradiation experiments
  - Analyses of the irradiation experiments and resulting data
- Development of advanced PIE methods
- Development and assessment of advanced cladding concepts
- Development and assessment of advanced matrix materials for dispersion fuels/inert matrix fuels for LWRs and fast reactors
- Fuel concepts for advanced LWRs/High-burnup fuels
- Fuel concepts for GFR, design analyses and testing
- Design and assessment of remote fuel fabrication facility
- Advanced fuel modeling (atomistic scale to continuum scale to performance codes)

D&EM PLANS FOR FY06-08

- Improve VHTR analysis capabilities and their validation status
  - Deterministic lattice physics and whole-core diffusion/depletion tools
  - Benchmarks based on measurements in critical facilities and operating reactors
  - System code for modeling coupled phenomena in transient and accident scenarios (coordinated with NGNP system design and evaluation activities)
  - Assessment of CFD simulation for modeling complex flows (e.g., in outlet plenum)
- Define fast reactor modeling needs as GFR and LFR systems are further specified
  - Available tools generally adequate for viability phase scoping analyses
  - Perform selected assessments to further define needs for modeling closed FR fuel cycle, e.g.,
    - Use of integral measurements to test nuclear data for minor actinides, GFR fuel matrix materials, Pb, Bi
    - Preparation of covariance data for actinides to support assessment of data related uncertainties in predicted reactor and fuel cycle parameters

D&EM PLANS FOR FY06-08

- **Advance evaluation methodologies for application to Gen IV systems and selection of preferred system options**
  - **Economics: Models for non-electricity energy products and for comparing economics of small vs. large plant units**
  - **PR&PP: Methods for estimating PR&PP measures**
  - **Risk and safety: Detail and carry out program of work based on GIF charter; define the framework for evaluating safety of Generation IV systems considering defense-in-depth principles and relevant safety standards**

D&EM PLANS FOR FY06-08

FY 2006 Milestones:

- **Document integral measurements applicable to validating nuclear data and analysis methods**
- **Report on improved capabilities for neutronic and depletion analysis of Gen IV systems**
- **Identify and prioritize phenomena to be represented in T-H and safety analysis codes; compile relevant data, correlations and integral measurements**
- **Provide assessment report and best practice guidelines for CFD code application to Generation IV systems**
- **Implement models for economic evaluation of non-electricity energy products**
- **Complete PR&PP methodology and verify its application to Generation IV systems**

D&EM PLANS FOR FY06-08

Milestones for FY2007 - FY2008:

- **Implement and qualify improvements to**
  - Monte Carlo and deterministic capabilities for neutronic, fuel depletion, and material damage analyses
  - T-H and coupled codes for system design and safety evaluations
- **Implement integrated economic evaluation models and issue for user testing**
- **Complete PR&PP evaluation methodology and implementation guide for users**

D&EM BUDGET POJECTIONS

Gen IV D&EM Funding by Fiscal Year ($K)

	FY-05 Approp.	FY-06 Request	FY-07 Plan	FY-08 Plan	FY-09 Plan	FY-10 Plan	FY-11 Plan
Program Coordination	150	150	175	201	231	266	306
Modeling Improvement	650	950	1,100	1,265	1,455	1,673	1,924
Evaluation Methodology	733	750	2,000	2,000	2,000	2,000	1,500

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Bill Corwin

National Technical Director

Gen IV Materials Technology Program

Oak Ridge National Laboratory

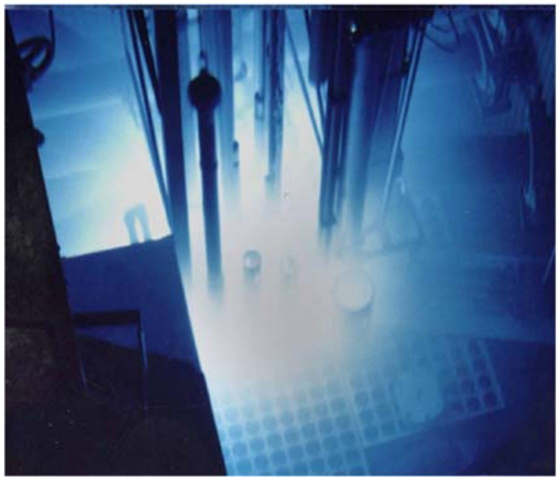
Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Selection, development, and qualification of structural materials needed to design and build the advanced reactors being developed within the Gen IV Reactor Program**
- **These activities are part of the Gen IV Reactor Program and are closely coordinated with similar structural materials research for the AFCI and NHI Programs**
- **Materials needs will be addressed for the NGNP, GFR, SCWR, and LFR reactor systems, as well as for their energy conversion systems, through R&D on their specific issues combined with crosscutting tasks**

Advanced Materials Development and Qualification Essential for All Gen IV Reactors

- **Materials Will Be Exposed to High Temperatures, Neutron Exposures, and Corrosive Environments**
- **60-Year Operating Lives for Gen IV Reactors Will Require Very Long-Term Materials Stability**
- **Process-Heat Use for Large-Scale Hydrogen Generation Will Also Require Materials Compatibility with Heat-Transfer Media and Reactants**
- **Research Will Build upon Extensive Previous Materials Development for Other Reactor Systems and Related Domestic and Foreign Programs**



FY03 and FY04 ACCOMPLISHMENTS

- **Several Assessments of Materials Needs, R&D Plans, and Technology Status Completed**
  - **NGNP, SCWR, LFR, GFR, and NHI Materials**
  - **Crosscutting R&D Plans for Radiation Effects & High-Temperature Materials Experiments and for Development of High-Temperature Structural Design Technology and the Gen IV Materials Handbook**
  - **Modeling and Microstructural Analysis: Needs and Requirements for Generation-IV Fission Reactors**
  - **Impact of High-Performance Computing on Irradiation Modeling**
  - **Generation IV Reactors Integrated Materials Program Plan**

FY03 and FY04 ACCOMPLISHMENTS

Experimental Materials Studies and Codes and Standards Activities Were Initiated

- Irradiation assessment of available nuclear graphites begun for NGNP service
- Initial corrosion testing for SCWR applications was begun and control strategies identified
- Materials were selected and corrosion exposure performed in Pb-Bi for LFR applications
- Joining studies of advanced ODS alloys were begun for GFR applications
- ASTM Standards on graphite testing and ASME Subcommittee on Graphite for Core Support Applications
- ASME Subgroup on Elevated Temperature Design (NH)

GEN IV MATLS WORK IN PROGRESS FOR FY05

$11,945K, 23 Work Packages, 11 Organizations

- **Crosscutting Materials (ORNL)**
  - **Materials for Radiation Service, High-Temperature Service, and Energy Conversion**
  - **Microstructural Analysis and Modeling**
  - **High-Temperature Design Methodology**
  - **National Materials Program Management**
- **Reactor-Specific Materials Technologies (ORNL, INL, LANL, LLNL, U. of Wisc., U. of Mich., Auburn, MIT)**
- **I-NERIs (ANL, INL, Penn. State)**
  - **Materials for Electrolytic Reduction**
  - **Advanced Corrosion Resistant Zirconium Alloys**
  - **Development and Evaluation of SCWR Materials**

Additional Materials Studies for NHI and AFCI Are in Progress

WORK IN PROGRESS FOR FY05

Crosscutting Materials Work Packages

- **Complete Irradiated Materials Survey,Low-Flux RPV Irradiation Site Selection, and Initiate Scoping Irradiations of High-Temperature Alloys in HFIR and Phenix**
- **Initiate Development of *Gen IV Materials Handbook***
- **Update Materials Needs Surveys for Microstructural Modeling and Energy Conversion Systems and Assess Models for ODS Performance and Stability**
- **Develop Preliminary High-Temperature Simplified Design Methods and Initiate Constitutive Equation Development**
- **Upgrade High-Temperature Creep and Creep-Fatigue Testing Facilities**

WORK IN PROGRESS FOR FY05

Reactor-Specific Materials Work Packages

- **NGNP Materials–Selecting & qualifying graphite, high-temperature metallic materials, and structural composites; improving HTDM; and assessing environmental and thermal aging effects**
- **GFR Materials–ODS materials joining, ion irradiation of ceramics, specimen preparation for high-temperature irradiations in Phenix of ceramics, composites, & refractory alloys**
- **SCWR Materials–Corrosion and SCC testing in SCW**
- **LFR Materials–Corrosion in Pb and Pb-Bi of austenitic & F-M steels, ODS alloys, and BMGs**

REACTOR-SPECIFIC MATLS PLANS FOR FY06-08

- **Continue irradiations, high-temperature testing, and environmental and thermal aging effects studies of graphite, metallic materials & structural composites for NGNP**
- **Develop initial simplified high-temperature design rules and constitutive equations for NGNP materials**
- **Initiate high-temperature, high-dose irradiations for GFR core support materials**
- **Continue mechanical properties and corrosion screening experiments for SCWR and LFR**

MATERIALS CROSSCUTTING PLANS FOR FY06-08

Materials for Irradiation Service

- Perform low-dose scoping irradiations of materials and PIE of RPV and reactor internals candidate materials in HFIR
  - 600 to 900ºC and up to ~10 dpa
  - 9Cr-1MoV, three nickel-base alloys (617, 800H, & Nimonic PE16), and two ODS F-M alloys (12 & 14YWT)
- Perform high-dose scoping irradiations of advanced F-M and ODS alloys in Phenix
  - 390 to 515ºC and up to ~70 dpa
  - EP-823, HT-9, Mod 9Cr-1Mo, HCM12A, NF 616, Inconel 800H, advanced 3Cr-W (w/ and w/o V) and 12YWT, 14 WT, & 14YWT ODS F-M steels

MATERIALS CROSSCUTTING PLANS FOR FY06-08

Materials for High-Temperature Service

- Establishment of *Gen IV Materials Handbook*
- Complete initial high-temperature materials scoping studies and codification actions
- Initiate joining and combined-effects high-temperature screening studies on commercial and near-commercial alloys and advanced high-temperature materials
- Initiate preparation of documents of alloy 617 and other commercial high-temperature alloys for ASME codification

MATERIALS CROSSCUTTING PLANS FOR FY06-08

Microstructural Analysis and Modeling

- Update integrated report prioritizing microstructural modeling needs for Gen-IV reactor program, and identifying needed special-purpose experiments
- Evaluate models for nucleation phase of the significant extended defects produced under irradiation
- Assess relevance of existing radiation and high-temperature models for ODS materials and initiate extension of atomistic, Monte Carlo models for ODS alloy lattice stability
- Initiate microstructural model development and special-purpose irradiations in critical areas

MATERIALS CROSSCUTTING PLANS FOR FY06-08

High-Temperature Design Methodology

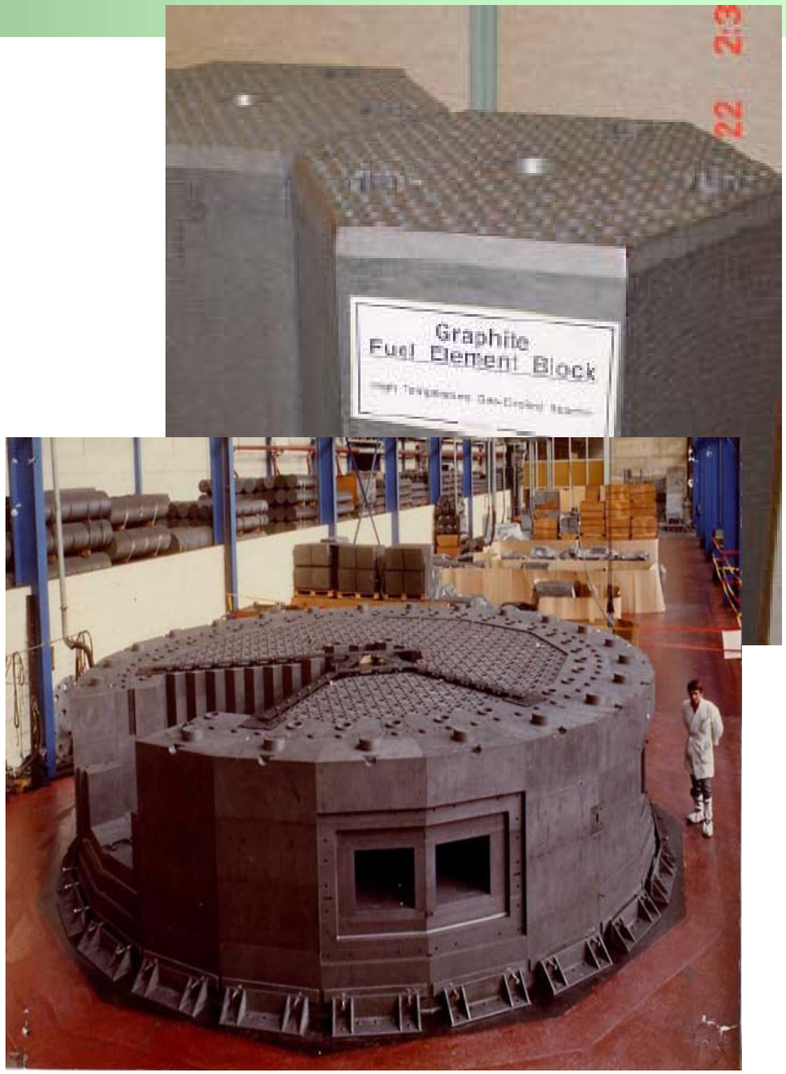
- Complete simplified methods development for primary high-temperature Gen IV materials
- Develop updated constitutive equations for modified 9Cr-1Mo steel (Grade 92) and Alloy 617
- Initiate development of rules to allow use of low-temperature design criteria for vessels subjected to limited high-temperature service
- Develop initial data and rules for very high-temperature usage of leading Gen IV materials

High-Priority NGNP Materials R&D Has Been Identified for FY06-FY08

- **Selection and qualification of graphite**
- **Selection and qualification of high-temperature metallic materials and development of improved high-temperature design methodology**
- **Assessment of irradiation effects and fabrication methods of reactor pressure vessels**
- **Assessment of environmental and thermal aging effects**
- **Development and qualification of structural composites**
- **Development of supporting ASME and ASTM codes and standards**
- **Development and population of materials database**

Currently Available Nuclear Graphites Must Be Qualified

- **Operating temps from 750° to 1250°C, off-normal to 1500°C**
- **Coordinated procurement of candidate materials**
- **Irradiation effects and irradiation-creep assessments**
- **Development and validation of graphite behavior models**
- **Development of required ASME Code and ASTM standards**

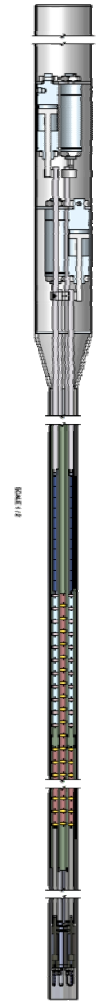


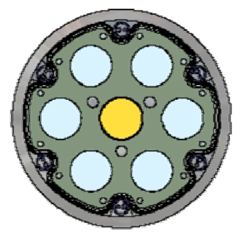
International NGNP Graphite Selection & Procurement Strategy Has Been Developed

- **Strategic Issues:**
  - Inputs from potential reactor vendors and graphite suppliers
  - Domestic (vs.) non-domestic supply
  - Diversity of suppliers
  - Longevity of supply
  - Cost
- **Four primary candidate grades have been selected :**
  - NBG-17 and NGB-18 (vibration molding, SGL, Germany)
  - PCEA (extruded, Graftech, USA)
  - IG-110 (iso-molded, Toyo-Tanso, Japan)
  - H-451 (extruded, Great Lakes, USA, historical reference graphite)
- **Selection and procurement has been coordinated with GIF partners**
- **Procurement has begun and will be completed in FY06**

Graphite Irradiation Creep Capsule Design Has Begun

- **Exposure in south flux trap in ATR**
- **600, 900, and 1200°C from 1 to 30 dpa**
- **Gas bellows for 13 and 20 MPa constant load**
- **Determine both creep coefficients and other property data required for constitutive equation constants for graphite materials**
  - **90 creep specimen (stressed/unstressed) pairs of 2 historical grades & 3 prospective new nuclear grade-graphites**
  - **Nine new prospective nuclear-grade graphites will make up over 300 unstressed piggyback specimens**





High-Temperature Graphite Irradiation Experiments Will Supplement Very Sparse Data above 1000°C

- **A very high temperature graphite irradiation capsule will be designed for irradiation of graphite in HFIR at temperatures up to 1200°C**
  - **NBG-17, NBG-18 (SGL Carbon)**
  - **PCEA, PCIB (GrafTech International)**
  - **IG-110, IG-430 (Toyo-Tanso)**
- **Preliminary capsule design and experimental plan will be completed in FY05**
- **Irradiation data obtained on candidate graphites in FY06 and beyond would include dimensional changes, elastic constants, strength data and coefficients of thermal expansion**
- **Scoping irradiations of NGB-10, PCEA, and IG-110 at multiple temperatures from 300-1200°C have begun**

Adequate Models Do Not Exist to Describe and Predict Graphite Behavior as F(Irradiation)

- **Early models developed for historically available graphites need to improved and validated for current nuclear graphites**
- **Such models based on sound physical principles and reflect known structural and microstructural changes that occur as a function of neutron irradiation**
- **Models that describe and predict irradiation-induced dimensional changes and creep behavior are being developed and will be validated using experiment data from graphite irradiation program**
  - **Microstructural models**
  - **Macrostructural models**

An experts group met in January at Univ. of Manchester to coordinate international modeling activities

Graphite Codes and Standards Are Being Developed under NGNP Materials Programs

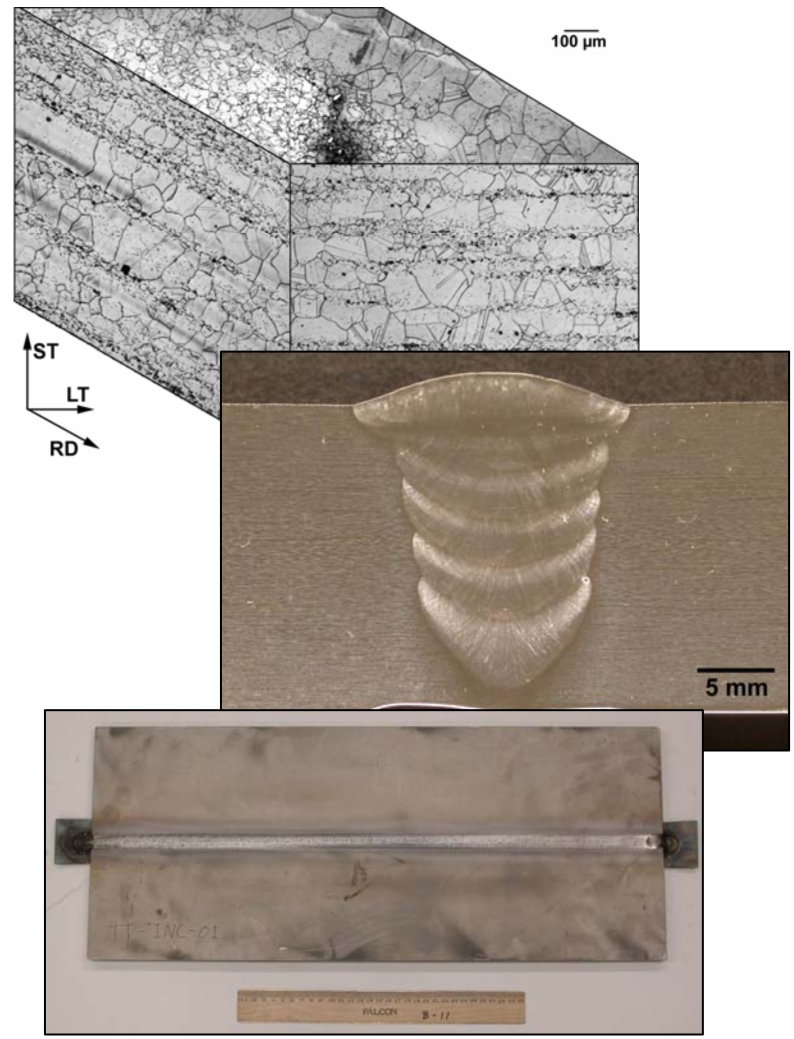
- **ASME design code development, required for use of graphite and C/C core support structures, addressed by newly formed ASME Section III Subcommittee CE**
- **ASTM committee DO2-F on Manufactured Carbons and Graphites is developing test method for determining K_{Ic} based on existing standard C-1421 (for ceramics)**
- **Other ASTM Standards for graphite being developed**
  - **Crystallinity by XRD**
  - **Surface area**
  - **Thermal Expansion**
  - **Graphite oxidation**

Development of Improved High Temperature Design Methodology and Supporting Data Critical to NGNP

- **Provide the data and models required by the ASME Code to formulate time-dependent failure criteria**
- **Provide material-dependent, experimentally-based constitutive models for inelastic design analyses required by ASME Sec III Subsection NH**
- **Update current high-temperature design rules based on separation of time-dependent and time-independent responses and strain-hardening idealizations for higher temperatures where such distinctions are inadequate**
- **Develop materials data and methods needed to support design and regulatory acceptance**
- **NGNP HTDM development primarily focused on RPV and IHX**

Alloy 617 Is Being Evaluated Initially and 230 Will Be Added

- **Collect historical 617 data for *Gen IV Materials Handbook***
- **Develop controlled chemistry spec for Alloy 617 (CCA) and procure material**
- **Initiate creep and creep-fatigue testing on standard Alloy 617 basemetal and joints samples in air, inert environment, and controlled-He atmospheres at 800-1000°C**
- **Initiate aging on base metal and welded specimens for 10,000 hour at 1000°C in inert atmospheres**
- **Develop initial simplified methods for Alloy 617 for use in ASME Code Case**



HTDM Activities Include Simplified Methods Development

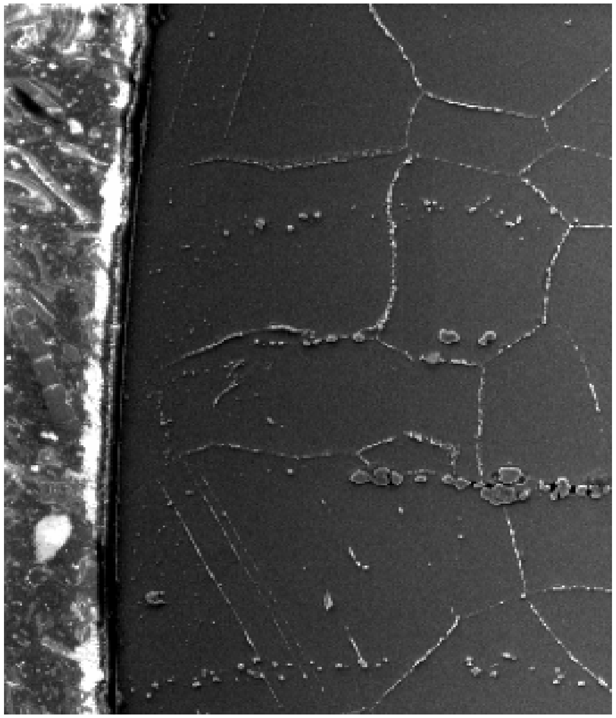
- Review technical development of current elevated temperature design rules developed for LMFBRs
- Assess Code adequacy with various operating conditions for various 1-D, 2-D, & 3-D geometries / loadings.
- Evaluate…
  - Load Control Design Rules: resistance to constant loads (creep) with Reference Stress Approach
  - Deformation Control Design Rules: resistance to cyclic loads (fatigue, creep-fatigue) with Cyclic Reference Stress Approach

HTDM Activities Also Include ASME Code Interactions

- **Overall goal to foster addition of advanced materials coverage into ASME Code Section III Subsection NH needed by NGNP**
  - **Longer times and higher allowable temperatures (e.g. extend usage of 2-1/4 Cr-1 Mo or 9 Cr-1 Mo (Grade 91) RPV steels**
  - **Acceptance of modified materials specifications (e.g. Mo 9 Cr-1 Mo Grade T91 vs T92)**
  - **New materials chemistries (e.g. Alloy 617 CCA)**
  - **Additional materials**
- **Initial thrust is reactivating ASME activities to adopt Draft Code Case on Alloy 617**
  - **Current ASME Alloy 617 draft code case must address a number of gaps and shortcomings before it can be accepted and applied**
  - **NGNP is requesting evaluation of code case by ASME Subgroup on Elevated Temperature Design, Subsection NH, Division I**
  - **Will serve as “ice breaker” for other high-temperature materials**

Environmental and Aging Effects Must Be Addressed for NGNP

- **Oxidation, carburization, and decarburization of metallic components in impure helium**
- **Microstructural stability during long-term aging**
- **High-velocity erosion/corrosion**
- **Rapid oxidation of graphite and C-C composites during air ingress accidents**
- **Compatibility with heat-transfer media and reactants for hydrogen generation**
- **Emissivity of RPV exterior**
- **Initial NGNP focus will be on high-temperature alloys, particularly 617**



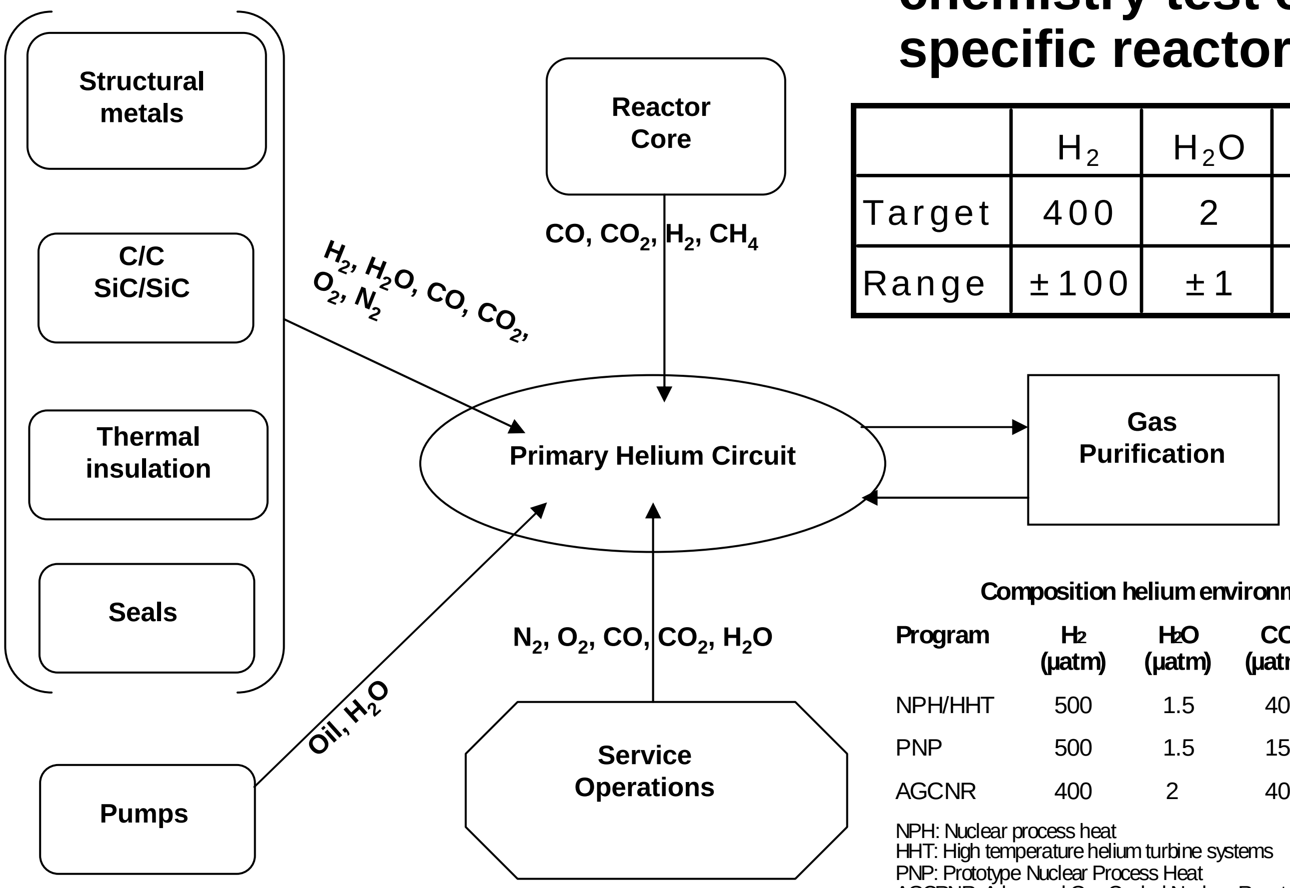
Alloy 617 Aged 100 Hours at 1000°C in air

Establish dynamic stability ranges of He impurities is 1st step

Helium Gas Chemistry Used in Past Helium Gas Cooled Reactor Programs Reviewed

- **Needed to establish a helium chemistry test envelope without a specific reactor design (μ atm)**

	H_2	H_2O	CO	CO_2	CH_4	N_2
Target	400	2	40	0.2	20	<10
Range	±100	±1	±10	±0.15	±5	



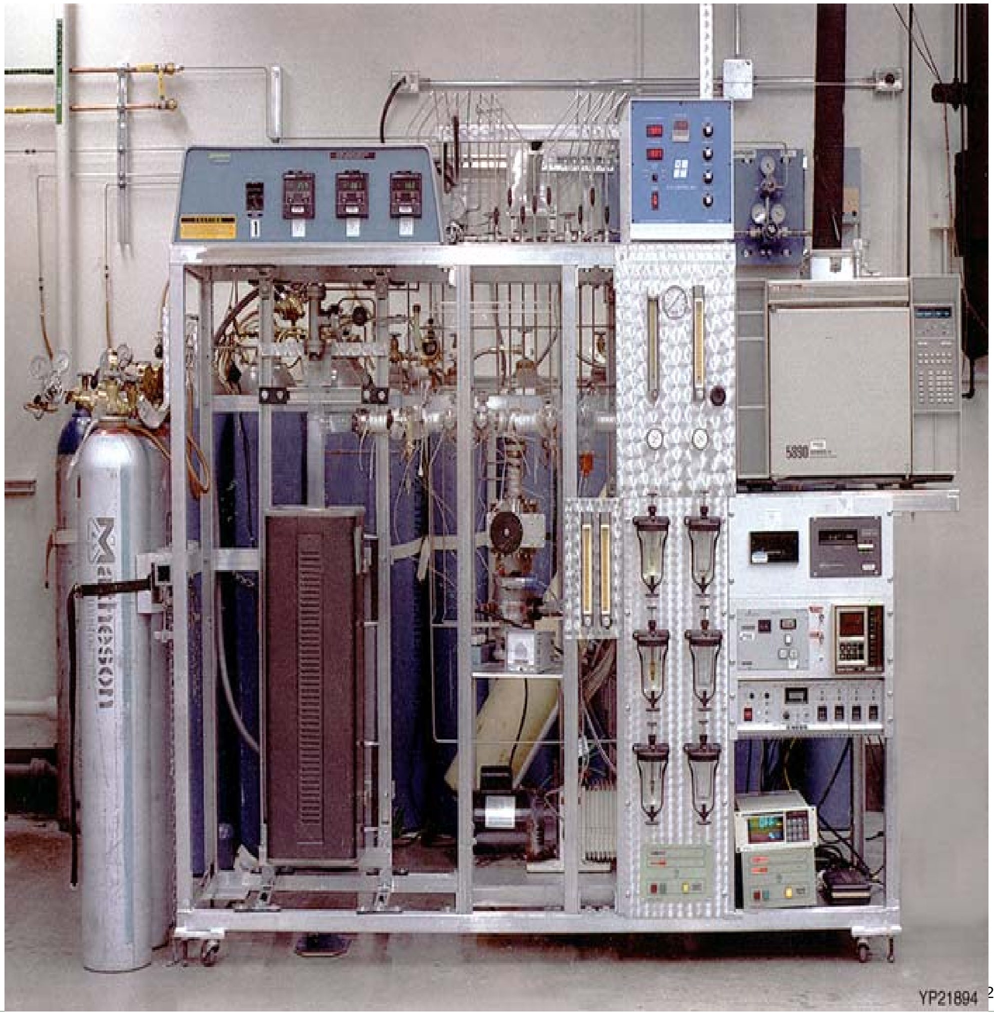
Composition helium environments (advanced HTGR) used in past tests

Program	H_2 (μatm)	H_2O (μatm)	CO (μatm)	CO_2 (μatm)	CH_4 (μatm)	N_2 (μatm)	He (atm absolute)
NPH/HHT	500	1.5	40		50	5–10	2
PNP	500	1.5	15		20	<5	2
AGCNR	400	2	40	0.2	20	<20	2

NPH: Nuclear process heat
HHT: High temperature helium turbine systems
PNP: Prototype Nuclear Process Heat
AGCRNR: Advanced Gas Cooled Nuclear Reactor

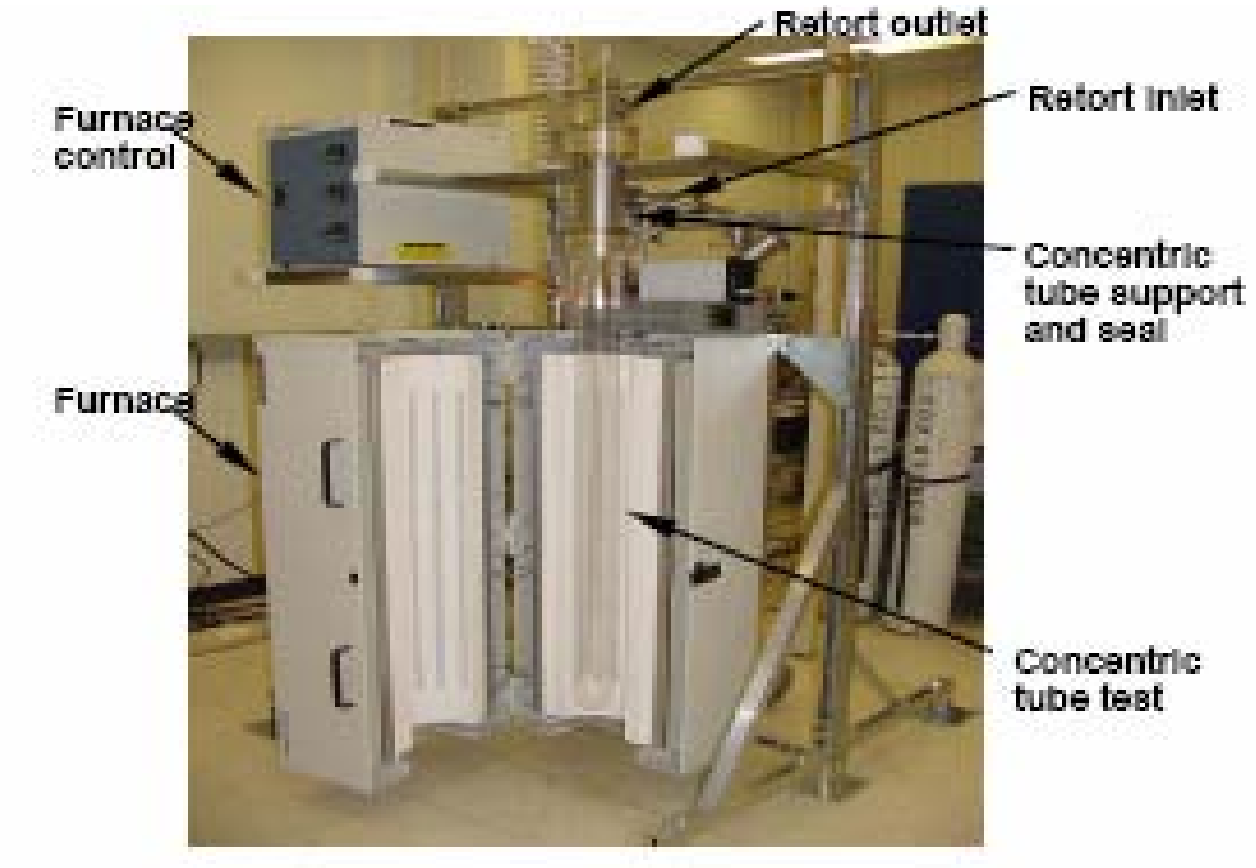
Assess He Chemistry in Existing Low-Flow Rate Recirculating Helium Loop

- **Will enable evaluation of gas/gas reactions to test dynamic stability of proposed test environments**
- **Allow for exposures of selected materials at in controlled helium chemistry at high temperatures**



Larger Low-Velocity He Loop Being Developed Will Increase Materials Exposure Capabilities

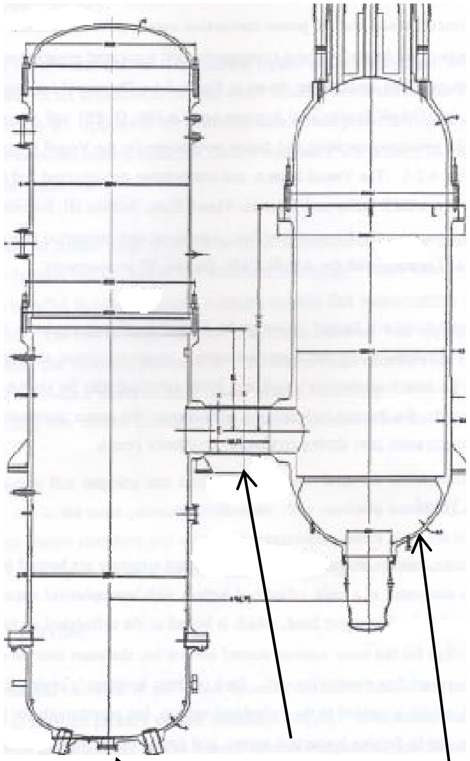
- All components have been ordered or received
- Retort and furnace section has been assembled
- Retort has been vacuum tested to $2X10^{-6}$ torr
- Testing to begin in FY06



9Cr-1MoV and 2 1/4Cr-1Mo Steels Are Top Candidates for NGNP RPV

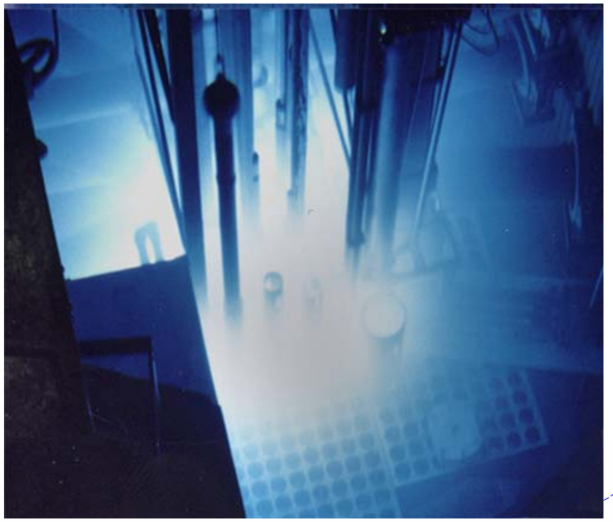
- **Temperature of service and vessel size dominate materials requirements**
  - **Up to 550°C**
  - **Up to 3 x 10^{19} n/cm^2 fluence**
- **Issues include irradiation effects, long-term strength, & high-temperature excursions**
- **High-temperature design methodology needs updating for nuclear service**
- **Very large vessel sizes will require scale-up of ring forging & joining technologies and ensuring thick-section properties**

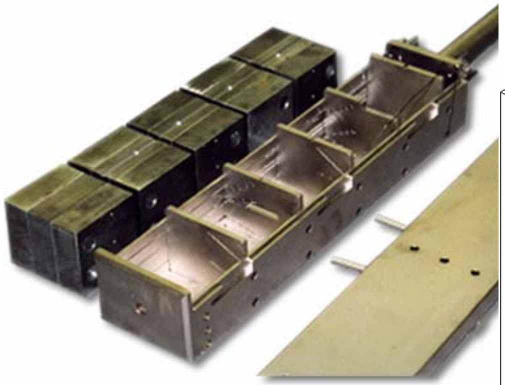
Correctly scaled size of typical PWR RPV

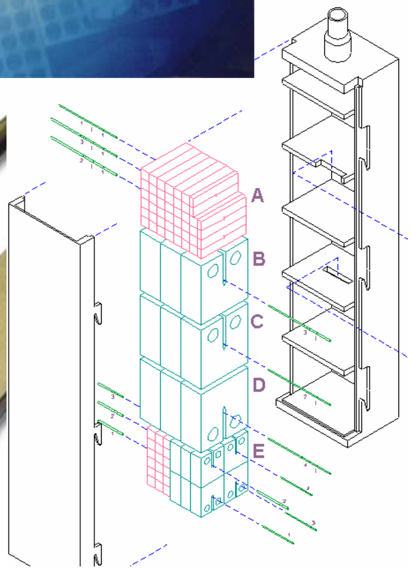


NGNP RPV Materials Irradiation Testing and Qualification Is a High Priority Task

- **Low-flux irradiation facility is being developed in cooperation with NRC's HSSI Program**
- **This facility would replace NRC irradiation facility that was shutdown at Ford Test Reactor at University of Michigan**
- **The facility will be versatile, multi-purpose, reusable, low-flux irradiation facility**
- **Initial irradiations should start in FY06 directed at candidate RPV alloys needed for design and regulatory acceptance**





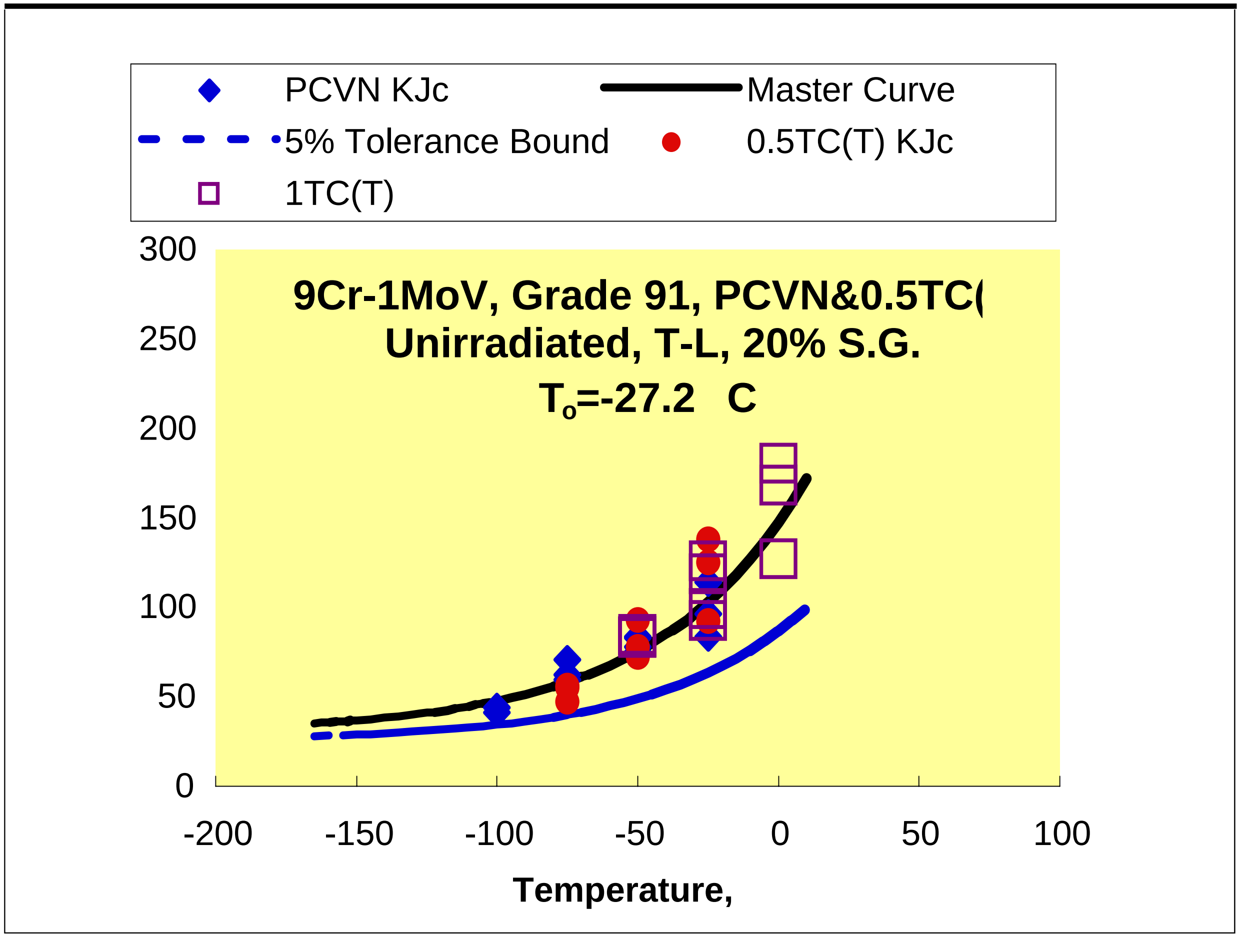


High-flux scoping irradiations will begin in HFIR in FY-05

Use of Master Curve Evaluated for Fracture Behavior of Gen IV RPV Steels

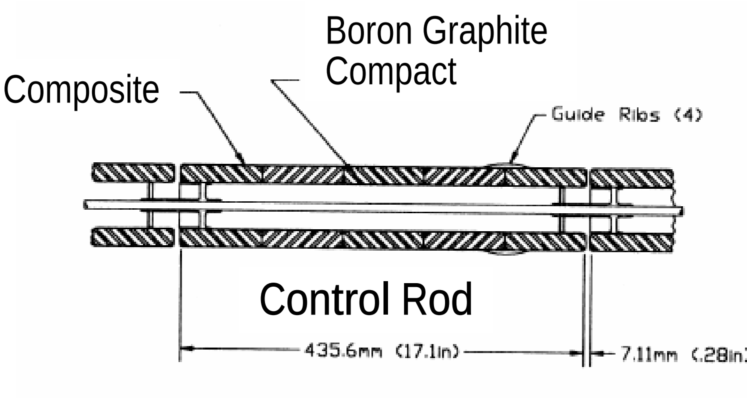
Weakest Link Size Effects and Fracture Toughness Master Curve Show Excellent Relationships for a 25-mm Thick Plate of 9Cr-1MoV Steel (Grade 91)

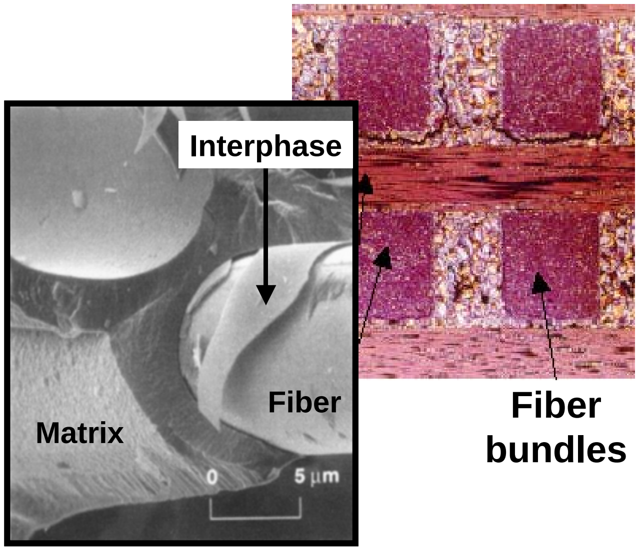
Confirmation for Other RPV Steels and in Irradiated Conditions Will Be Required



Structural Composites Needed for Very High-Temperature Reactor Components

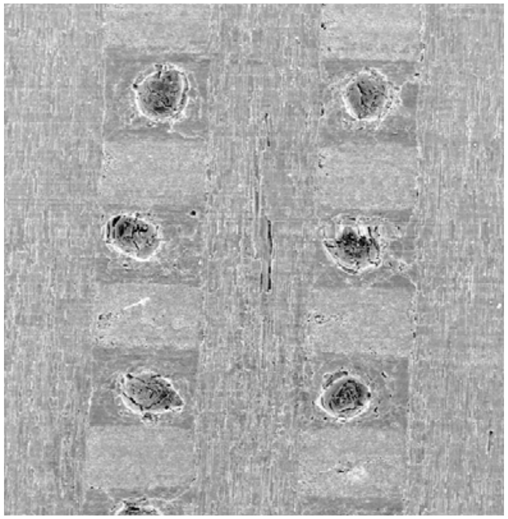
- **Control rods and high-temperature reactor internals will likely require structural composites**
- **Initial focus will be on SiC-SiC for control rods**
  - **NGNP will need to extend 200°C higher than current data and up to 30 dpa dose**
  - **Fabrication, irradiation-effects & environmental “creep” issues**
  - **Experimental comparison with C-C performance will be made**
- **C-C composites will also be evaluated for larger structural applications**





High-Dose Irradiations Will Determine Economic Viability for SiC-SiC Control Rods

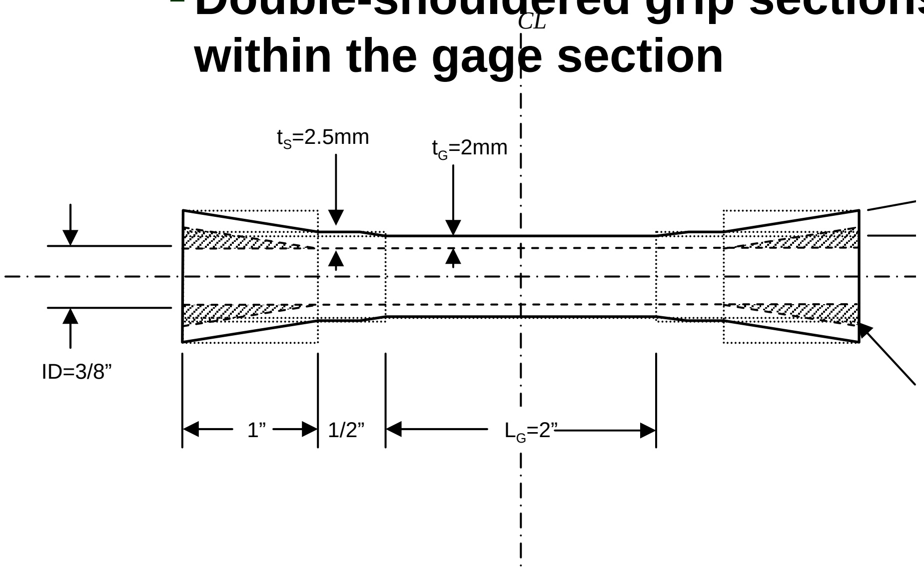
- **Irradiation tests in HFIR of bend bar capsules will determine of C-C versus SiC-SiC lifetimes**
  - **10, 20 & 30 dpa levels at 300, 500 & 800°C (C-C to 15 dpa)**
  - **Compliment higher temperature ORNL fusion and Futurix-MI irradiations of SiC-SiC**
  - **10 dpa achieved in FY05**
- **Begin definition of creep tube geometry for out-of-pile and in-pile testing**
- **Develop design for creep capsule located in-pile in ATR**

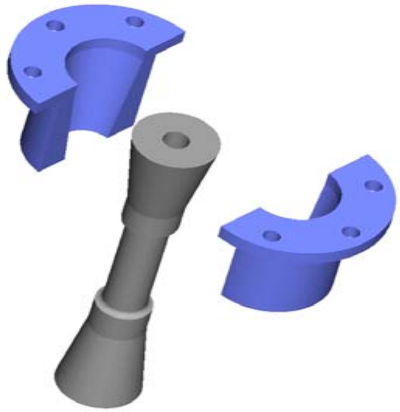


3-D SiC-SiC Architecture

Specimen & Fiber Architectures Determined

- Tube/flat plate mechanical property correlations are being developed
  - Tubular specimen with constant ID and integrated ceramic end inserts designed for evaluation of tensile strength
  - Novel fabrication method for the tapered composite tubes, involving SiC matrix infiltration to SiC fiber preforms braided over mandrel and end inserts, was developed
  - Double-shouldered grip sections employed in order to force fracture within the gage section

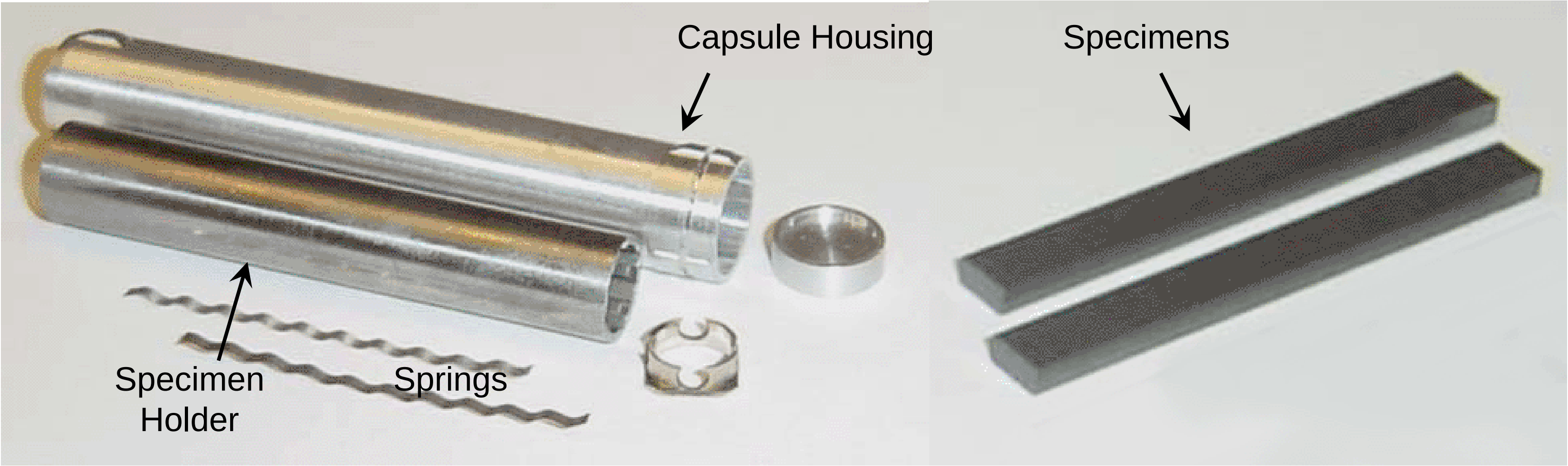




Bi-axially Braided Tube

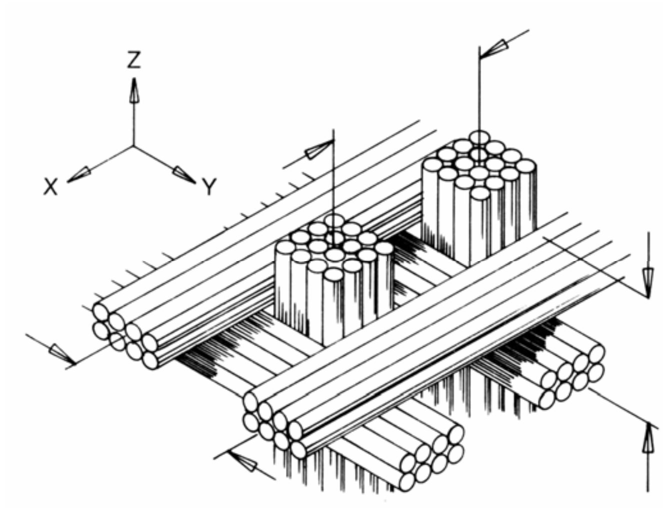
High-Dose SiC-SiC & C-C Irradiations Begun

- **NGNP bend bar rabbit capsules for SiC/SiC and graphite composites have been inserted in HFIR**
- **Some of the 10 dpa irradiation capsules have completed irradiation.**

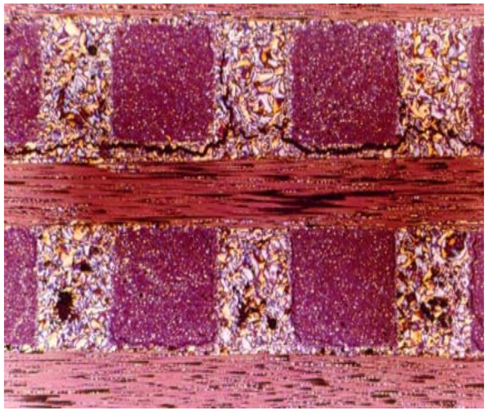


C-C Composites Will Be Evaluated

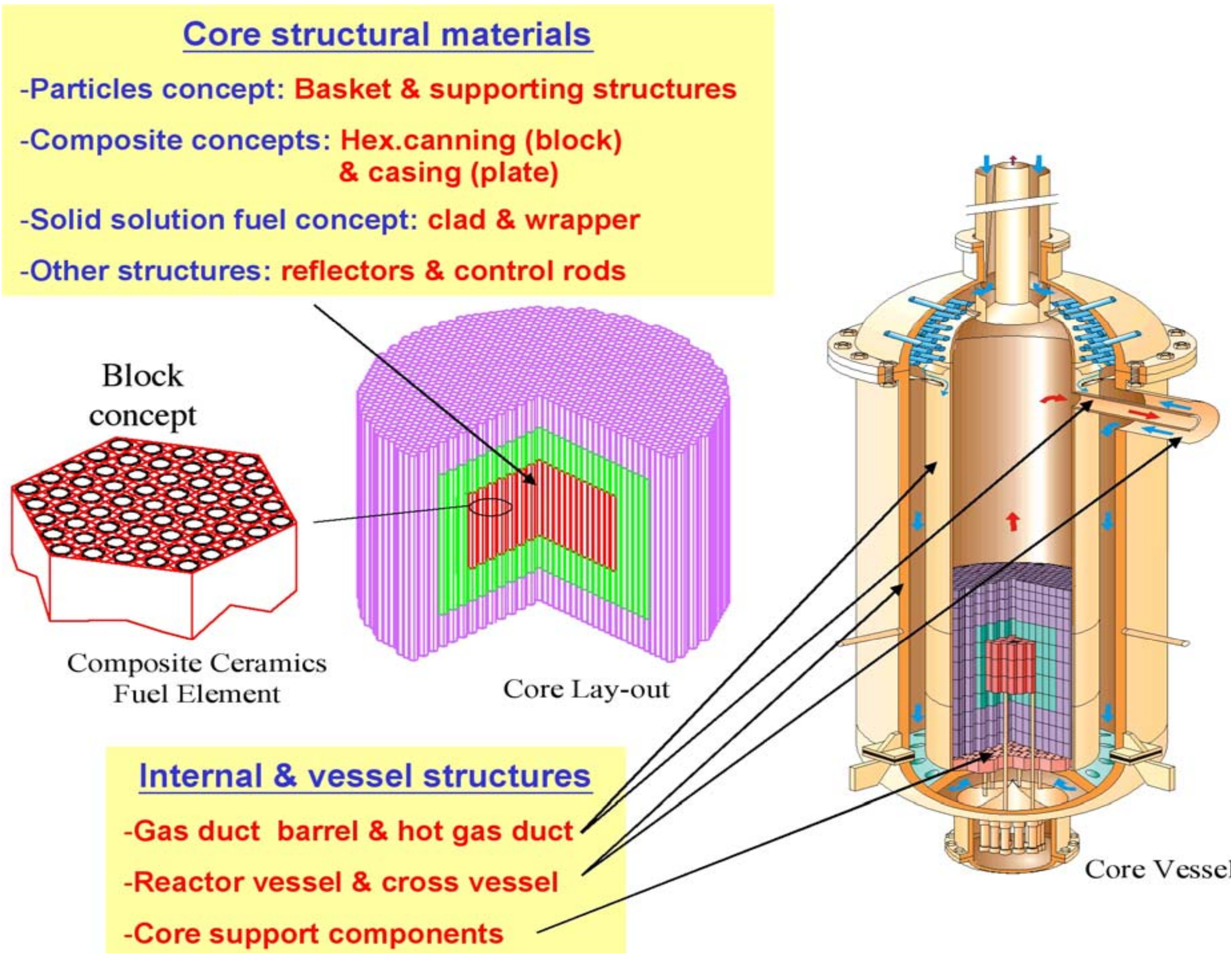
- **Assess potential vendors for fabrication of complex components**
- **Candidate C-C composite tubes will be procured and characterized for control rod applications**
- **Architectures and designs for larger structures will be evaluated for subsequent procurement and evaluation of scale-models and prototypes**



3-D Composite



To Maintain a Fast Spectrum, GFR Core Cannot Contain Graphite



High-Dose, Fast-Spectrum, Irradiation-Resistant, Low-Carbon Core Materials Are Key

Most Other GFR Materials Needs Will Be Enveloped by NGNP

GFR Fuel Matrix and Structural Material Reference Requirements Are Demanding

Requirement	Reference Value
Melting/decomposition temperature	**>2000°C**
Radiation induced swelling	**< 2% over service life of 15 to 200 dpa**
Fracture toughness	**> 12 MPa $m^{1/2}$**
Thermal conductivity	**> 10 W/mK**
Neutronic properties	**Materials must allow low heavy metal core inventory and maintain good safety parameters**

Candidate structural materials: SiC, NbZrC, ZrC, TiC, ZrN, TiN, AlN, Al_2O_3, SiC-SiC, TZM, Nb1Zr, Mo-Zr-B, Mo-Si-B, and NiAl

REACTOR-SPECIFIC MATLS PLANS FOR FY06-08

GFR Materials

- **Evaluate irradiation resistance of structural materials for core support applications**
- **Perform scoping materials compatibility studies with super-critical CO_2 in the temperature range of 400 to 650°C**

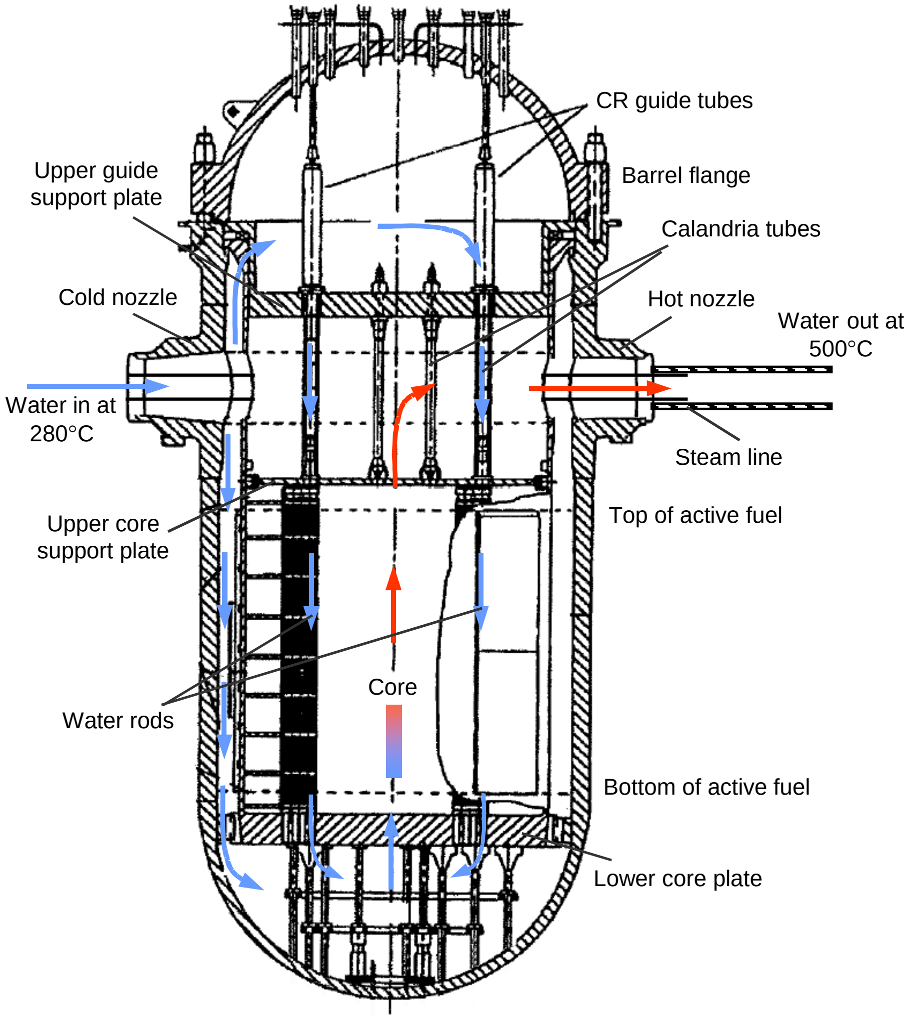
Samples Have Been Sent to CEA for Inclusion in Phenix Futurix MI Irradiations

Can Nr	1	2	3	4	5	6	7	8
Material	SiCf/SiC 2D	Nb-1Zr	αSiC sngl xtl	α SiC $^{nat}B_4C$	β SiC CVD	TiN	ZrN	TZM[f]
Manufacturer	MAN technol.	INL[b]	LETI 6H	HEXOLOY SA	ROHM&HASS	LTMEX	LANL[e]	PLANSEE
Disc 8mm h=2mm	10	10	10	10	10	10	10	10
TEM spec	20[j]	20	15	0	0	20	20	20[g]
Beam 25x2,5x2 mm^3	6	6	0	0	0	0	6	6[g]
Cyl. 3mm h=5mm	0	0	0	1	1	0	0	0
Cylinder 3mm h=10mm	1	1	1[k]	1	1	1	1	1
Material	SiCf/SiC 2D	Mo Alloys	SiCf/SiC 2D	α SiC $^{11}B_4C$	TiC	TiN	ZrN	ZrC
Manufacturer	NITE[a]	ORNL[i, g]	ORNL[c]	BOOSTEC[k]	LTMEX	CERCOM[d]	LTMEX	LTMEX
Disc 8mm h=2mm	10	5[i] + 5[g]	10	10	10	10	10	10
TEM spec	20	10[i] + 10[g]	20	20	20	20	20	20
Beam 25x2,5x2 mm^3	6	3[i] + 3[g]	12	12	12	12	6[h]	6
Cyl. 3mm h=5mm	0	0	0	0	0	0	0	0
Cylinder 3mm h=10mm	1	1[i]	1	1	1	1	1	1

- **Irradiations to approximately 40 dpa at 1000°C**
- **Insert in reactor early in 2007**

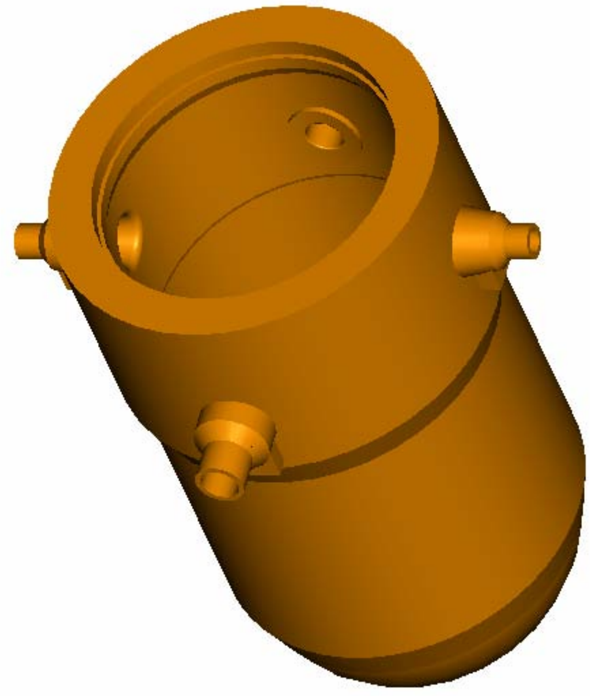
SCW Corrosion on Internals Is Greatest SCWR Materials Challenge

- **Effect of radiolysis on coolant chemistry**
- **Effect of radiation and coolant on corrosion, SCC, and IASCC**
- **Temperatures from 280 to 500°C**
- **Radiation exposure will further limit internals materials**
  - **microstructural stability**
  - **mechanical properties**
  - **fracture resistance**
- **Internals candidate materials**
  - **Low-swelling stainless and F-M steels**



Manufacturing Requirements for SCWR Vessel Ring Forgings Stretch Infrastructure

- **Maintaining through-thickness mechanical and chemical properties during fabrication is primary challenge**
- **Inspectability for very heavy sections must be ensured**
- **Primary candidate material**
  - **A508 Grade 3 Class 1**
- **Alternate high-strength materials**
  - **A508 Grade 4N Class 1**
  - **3Cr-3WV**



- **280°C wall temperature**
- **<5x10^{19} n/cm^{2} (E>1 MeV)**
- **27.5 MPa nominal pressure**
- **Thickness *46 cm (18") in the beltline region, ~61 cm (24") in the nozzle region***

REACTOR-SPECIFIC MATLS PLANS FOR FY05-07

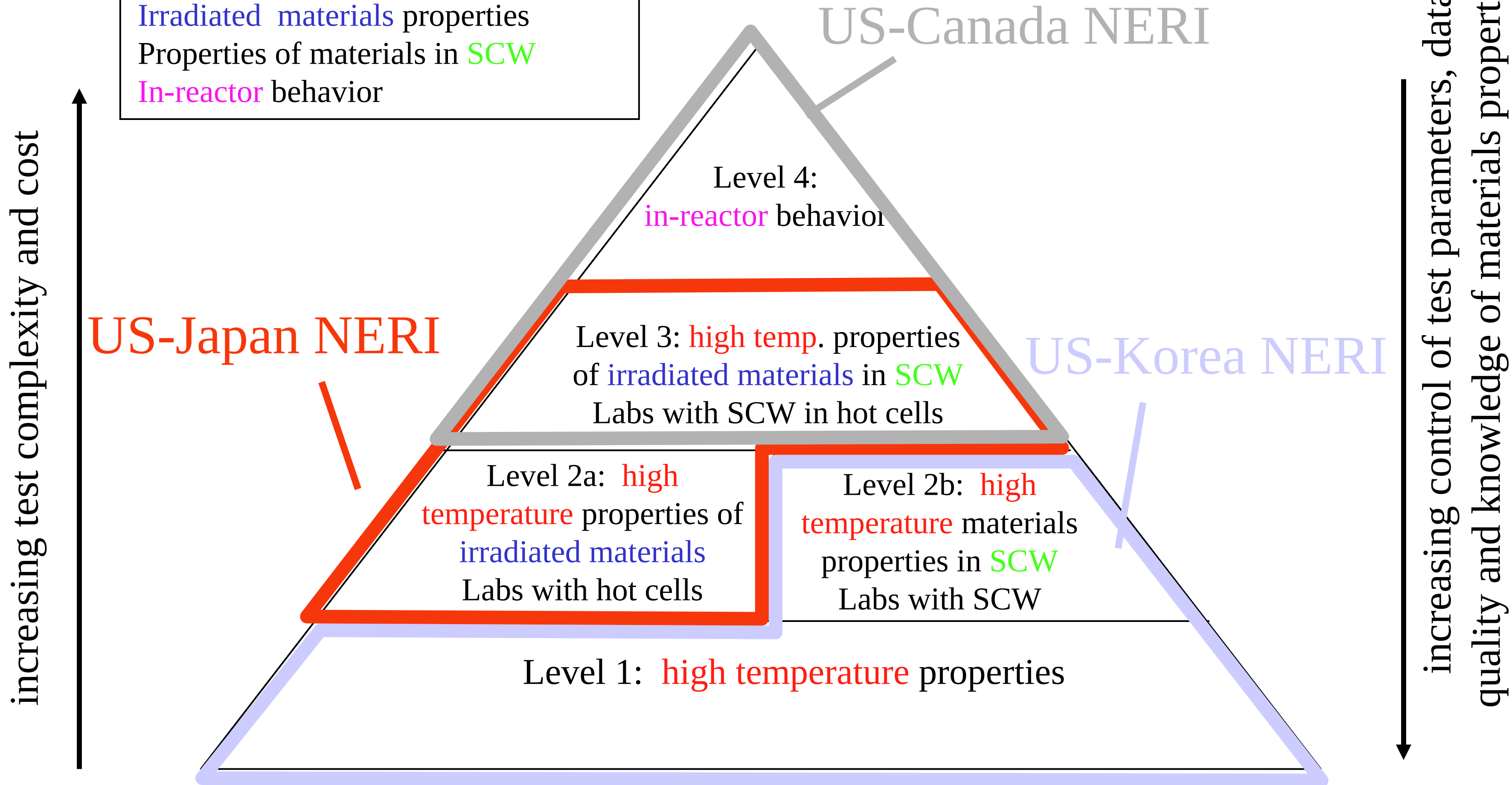
SCWR Materials R&D Will Focus on Compatibility

- **Perform initial corrosion and SCC screening tests for internals in supercritical water for SCWR**
- ***Irradiation issues will be addressed through GIF collaborations***

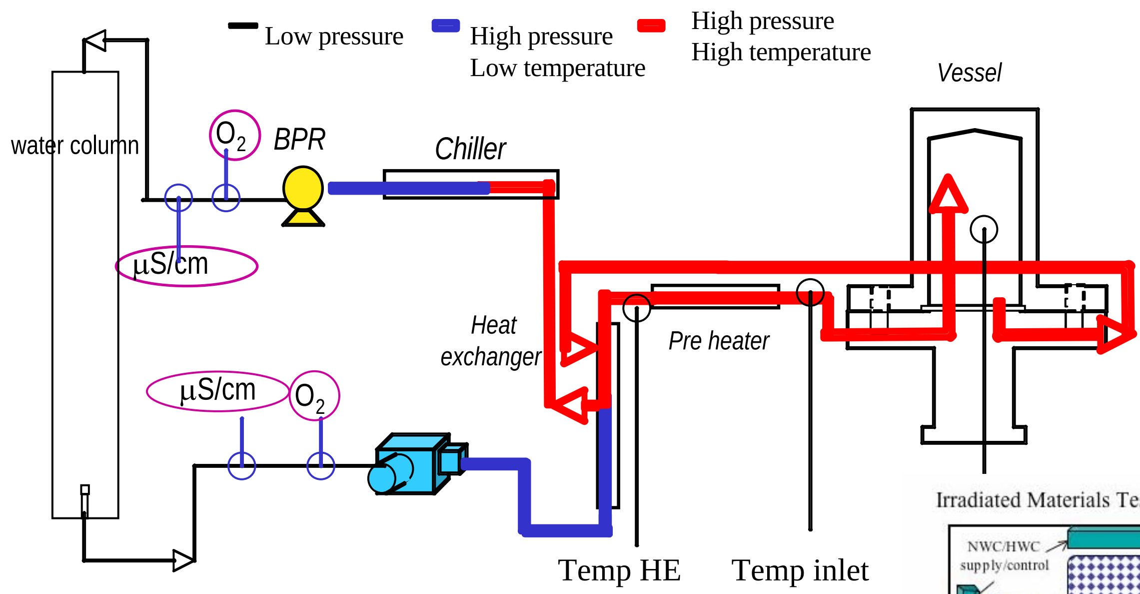
U.S. SCWR Materials Program Well Integrated with GIF Partners

<u>Types of Materials Studies Required</u>

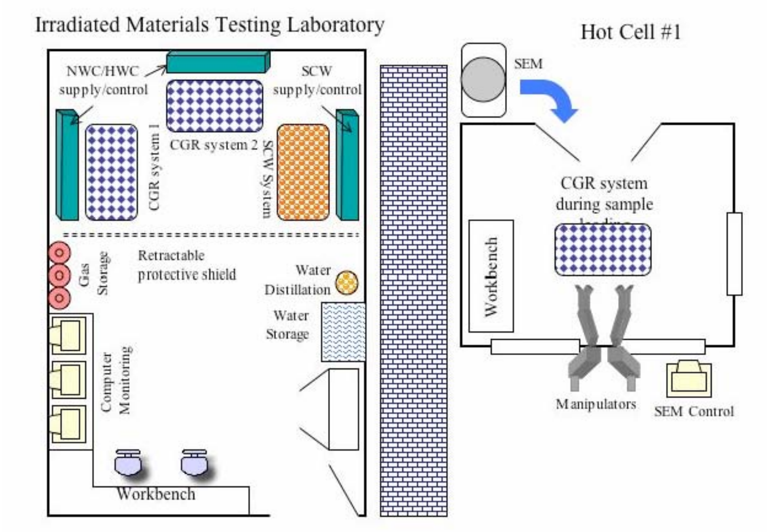
High temperature materials properties
Irradiated materials properties
Properties of materials in SCW
In-reactor behavior



4-Sample CERT Facility Developed at U Mich Will Be Used to Evaluate SCC and IASCC



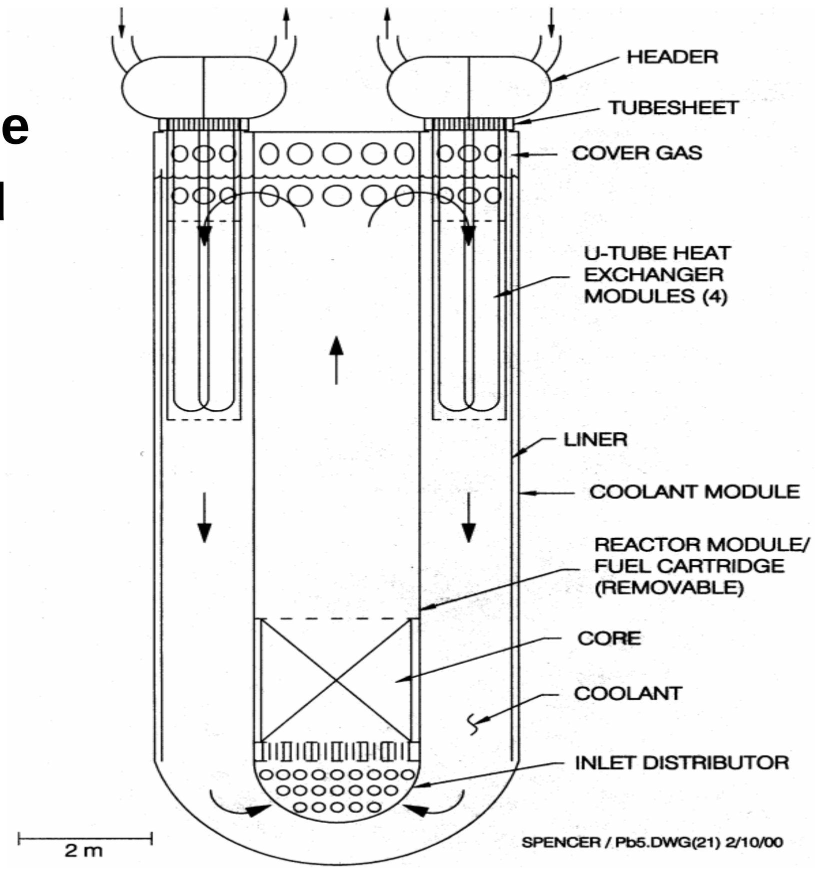
- ***Test specimens can be loaded in hot cell and moved to controlled laboratory for testing irradiated materials***



- ***Four samples can be strained simultaneously***
- ***The stress-strain curves developed correlate very well with the data obtained in the single sample SCW system at 500°C***

A 20 MWe (45 MWt) Small Secure Transportable Autonomous Reactor (SSTAR) Is Being Developed

- **Long refueling-interval, transportable system, proliferation resistant, and passively safe**
- **Range of Operating Conditions**
  - **650°C peak cladding temperature**
  - **20 year core lifetime**
  - **Pb or Pb-Bi coolant @ 1 atm**
  - **150 dpa peak dose**
  - **Supercritical CO_2 Brayton cycle**
  - **Cu-I or Ca-Br thermochemical H_2 production**



ODS Materials Are Prime Candidates for Cladding and Core Supports

REACTOR-SPECIFIC MATLS PLANS FOR FY06-08

Initial LFR Materials Studies Will Address Compatibility

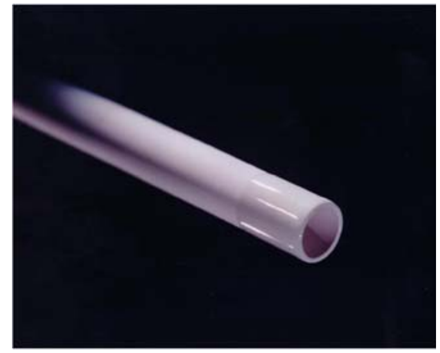
- **Perform scoping studies of preliminary LFR candidate materials for corrosion resistance in Pb and Pb-Bi**
- **Initiate scoping studies of surface treatments for controlling corrosion in LFR environments**
- **Initiate assessment of surface protection mechanisms in LFR materials**

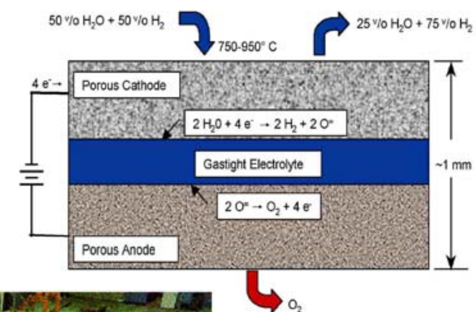
Pb and Pb-Bi Eutectic Exposures Being Performed in Delta Loop and Quartz Thermal Convection Loops

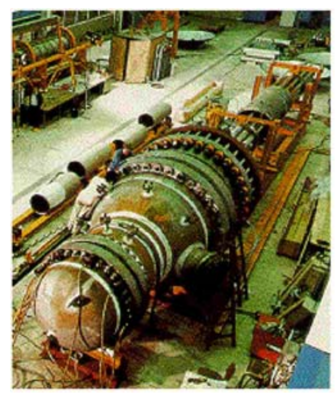
- **Initial Corrosion Tests Confirmed:**
  - **Efficacy of oxygen control for corrosion reduction and effects of Si and Cr on enhancing corrosion resistance**
  - **Al surface coating (pack cementation) is effective in protecting 316L in short to medium term at 520°C**
  - **Shot-peening of 316L may enhance short-term corrosion resistance**
  - **Significant intrusion of Pb-Bi into the MA 957 alloy at 650°C but little to no intrusion of Pb into alloy**
- **ODS steel specimens under preparation for DELTA testing (MA956, MA957, 14WT, 14YWT, 12YWT, PM2000)**
- **Collaborating with LLNL to prepare laser-peening of specimens for DELTA testing (316L, HT-9, T91)**
- **Bulk amorphous metal samples are being included Delta Loop and lead exposures at ANL**

Nuclear H_2 Generation Materials Must Withstand Harsh Environments

- **Thermo-Chemical processes**
  - **S-I, 950°C vapor to 500°C boiling sulfuric acid**
    - **Ceramic/noble coatings, sandwich structures**
  - **Inorganic membranes may dramatically reduce separation temperature**
- **High Temperature Electrolysis**
  - **Electrode cost, performance, stability, & fabrication**
  - **Catalysts**
- **IHXs for H_2 plant and nuclear/H_2 interface**
  - **High temperatures for operation and off-normal events**
  - **Secondary loop(s) coolant type(s)**
  - **H_2 plant reactants**
  - **Pressure drops (across IHX and to ambient)**
- **Intermediate loop piping**
  - **Temperatures, pressures, and coolant(s)**







A *Materials Handbook* Is Being Developed for the U.S. Gen IV Program

- **Stakeholders meeting held in July 2004 defined consensus needs and overall handbook approach**
- **Contain materials data and materials-related information that are used in all phases of design and analysis**
- **Internally consistent, validated, and highly qualified**
- **Authoritative single source of materials data**
- **Ensure that materials data are available at earliest possible time as input to preliminary designs and comparisons prior to final approval or codification of design values**
- **Usable for codification and regulatory analysis**

GIF Partner Participation Desired at Earliest Opportunity

Content of the *Materials Handbook* Will Meet Needs of Gen IV Reactors

- **Prioritized according to the needs of Gen IV Reactor SIMs, systems designers and vendors, codes/standards, and regulatory bodies**
- **Only metallic and ceramic structural materials included**
- **Initially priority given to materials needed for NGNP**
- **Structural materials needed for the Nuclear Hydrogen Initiative and the Advanced Fuel Cycle Initiative will be included**
- ***Gen IV Materials Handbook Implementation Plan* has been completed and web-based *Handbook* site development begun**
- **Initial compilation of historically available data underway**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

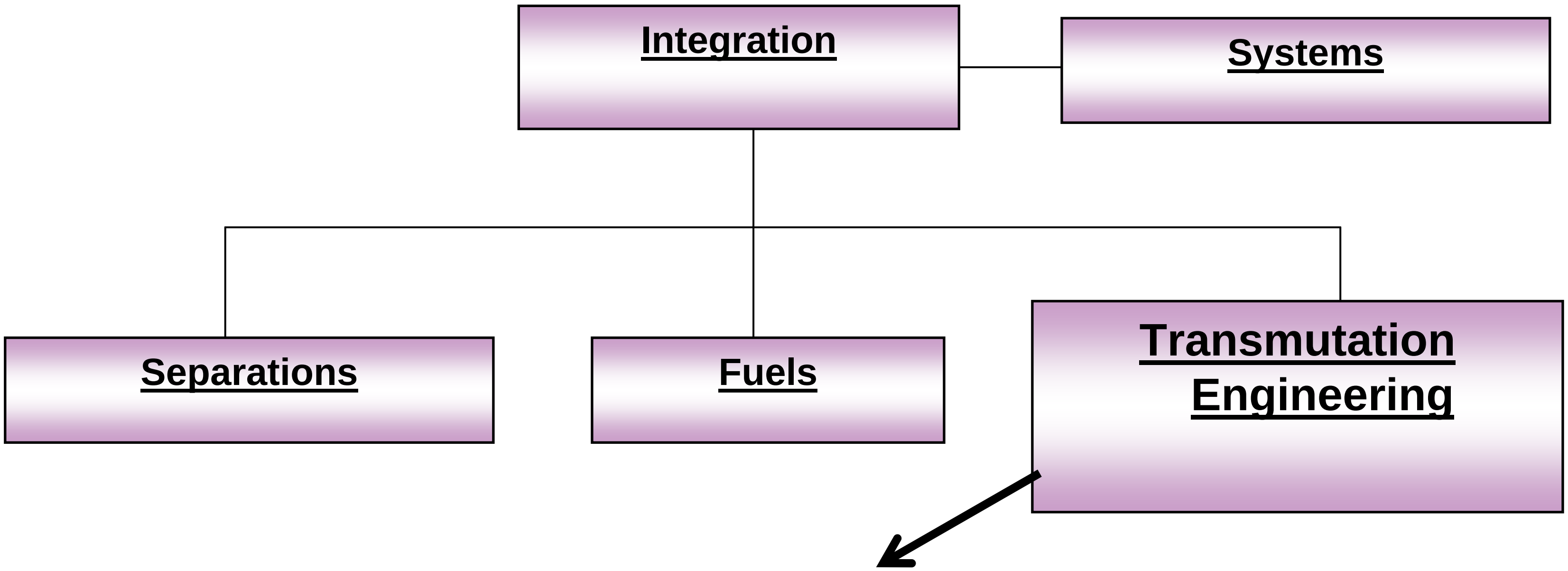
Transmutation Engineering

Michael Cappiello

National Technical Director

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Transmutation engineering provides critical R&D to support AFCI transition fuel cycle and the AFCI/GenIV equilibrium fuel cycle



- **Physics,**
- **Materials,**
- **Coolant technology, and**
- **ADS**
- **Fast spectrum irradiation capability (MTS).**

Transmutation Engineering is Organized into Three Research Areas

Physics

- Nuclear data in thermal, epi-thermal and fast spectra
- Nuclear Safety data
- Codes and Models

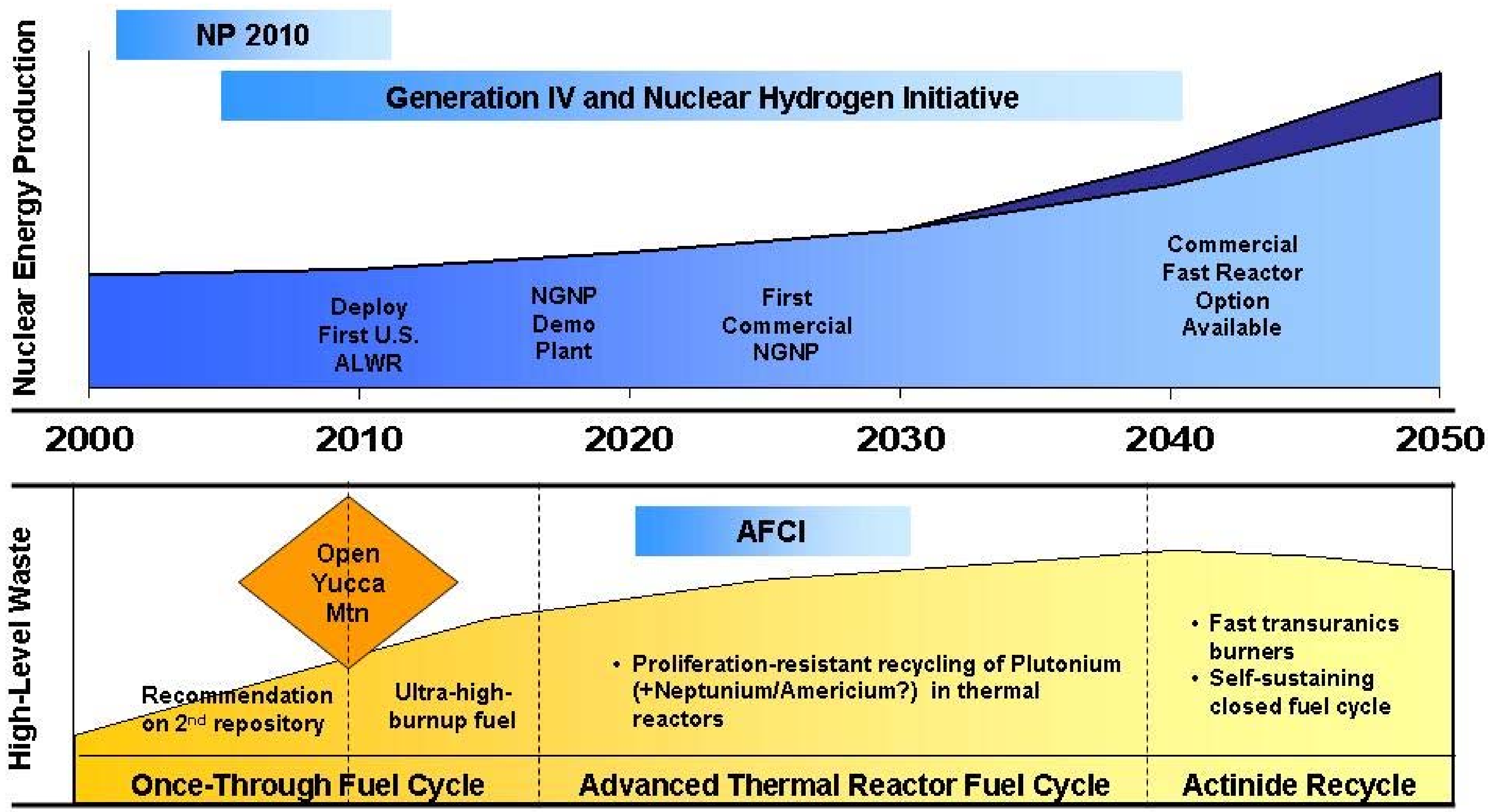
Materials

- Structural material degradation during irradiation: material limits
- Lead-Bismuth Coolant, sensor technology and corrosion mitigation
- Materials Test Station

ADS

- Coupling of accelerator to sub-critical reactor
- Operation and safety of ADS
- Target technology
- Accelerator Reliability

And Supports the Top Level AFCI/GenIV Vision



FY04 Summary

Physics:

- **Iron and Chromium gas production measurements**
- **NP237 fission and capture measurements**
- **^{237}Np, ^{241}Am ,^{242m}Am, ^{242g}Am data evaluations**

Structural Materials:

- **Examination of FFTF irradiated low activation ferritic/martensitic steels at 400C to doses up to 60 dpa**
- **Ion Irradiations on HT-9 at U. Mich to a dose of 3 dpa**
- **Effects of Shot Peening on Corrosion and Radiation Resistance of Austenitic and Ferritic/Martensitic Steels**
- **Atomistic Modeling of He in Body-centered Cubic (BCC)-Fe**
- **MTS conceptual design**

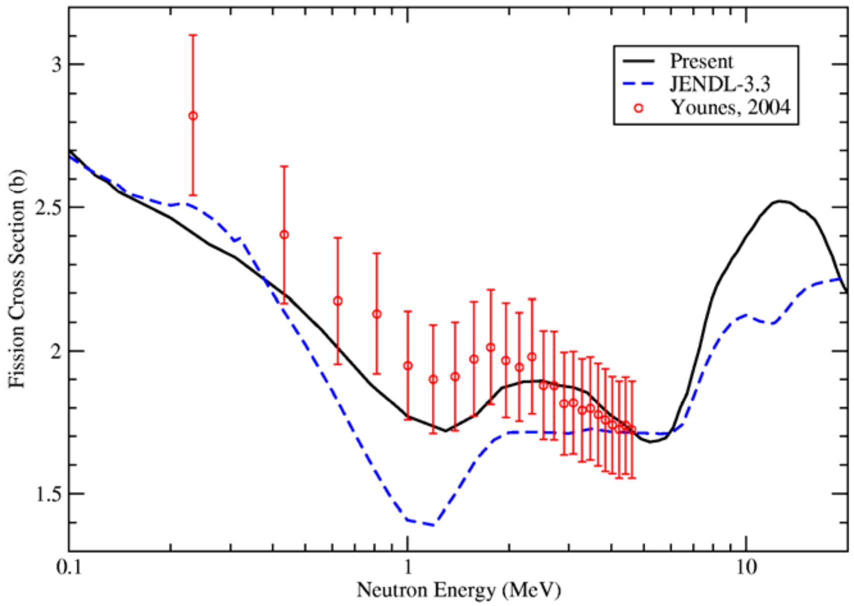
ADS:

- **Continued collaborations with TRADE and MEGAPIE**
- **Development of RACE experiment: Coupled electron accelerator (Idaho State) to TRIGA (U-Texas and Texas A&M)**

Transmutation Physics is Addressing High Priority Nuclear Data Needs

Highlights:

- 237Np data reduction
- Target fabrication capability established
- Am242g evaluation
- Working group established with Gen-IV



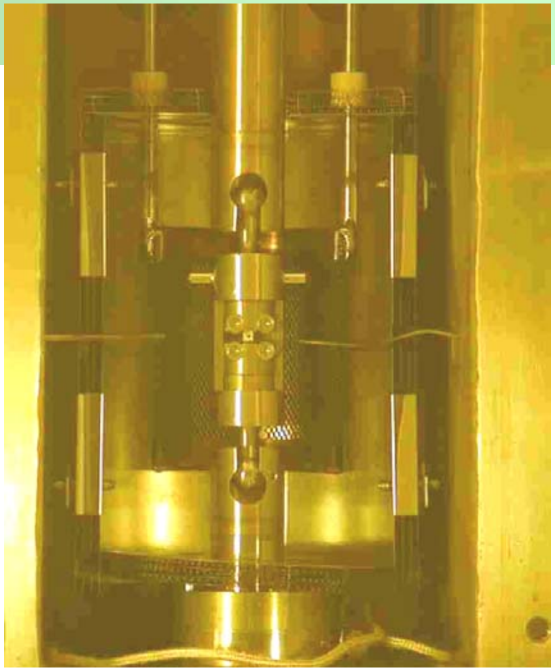
Collaborations:

- CEA/CNRS: Model development
- NC State: displacement library
- EURATOM: Fission cross section measurements

Plans:

- Actinide fission and capture measurements
- Uncertainties for evaluated nuclear data files

Structural Materials Research Continues with Archived Irradiated Samples



Tensile specimen testing apparatus in the CMR hot cell. Capable of testing at variable strain rates and temperatures up to 700˚C

Highlights

- Mechanical Testing completed on STIP II irradiated HT-9 and EP-823. Total dose up to 20 dpa and 2,000 appm helium. Tested at RT and 400˚C
- Ion irradiations performed at U. of Michigan on single crystal Fe with and without helium to validate modeling calculations

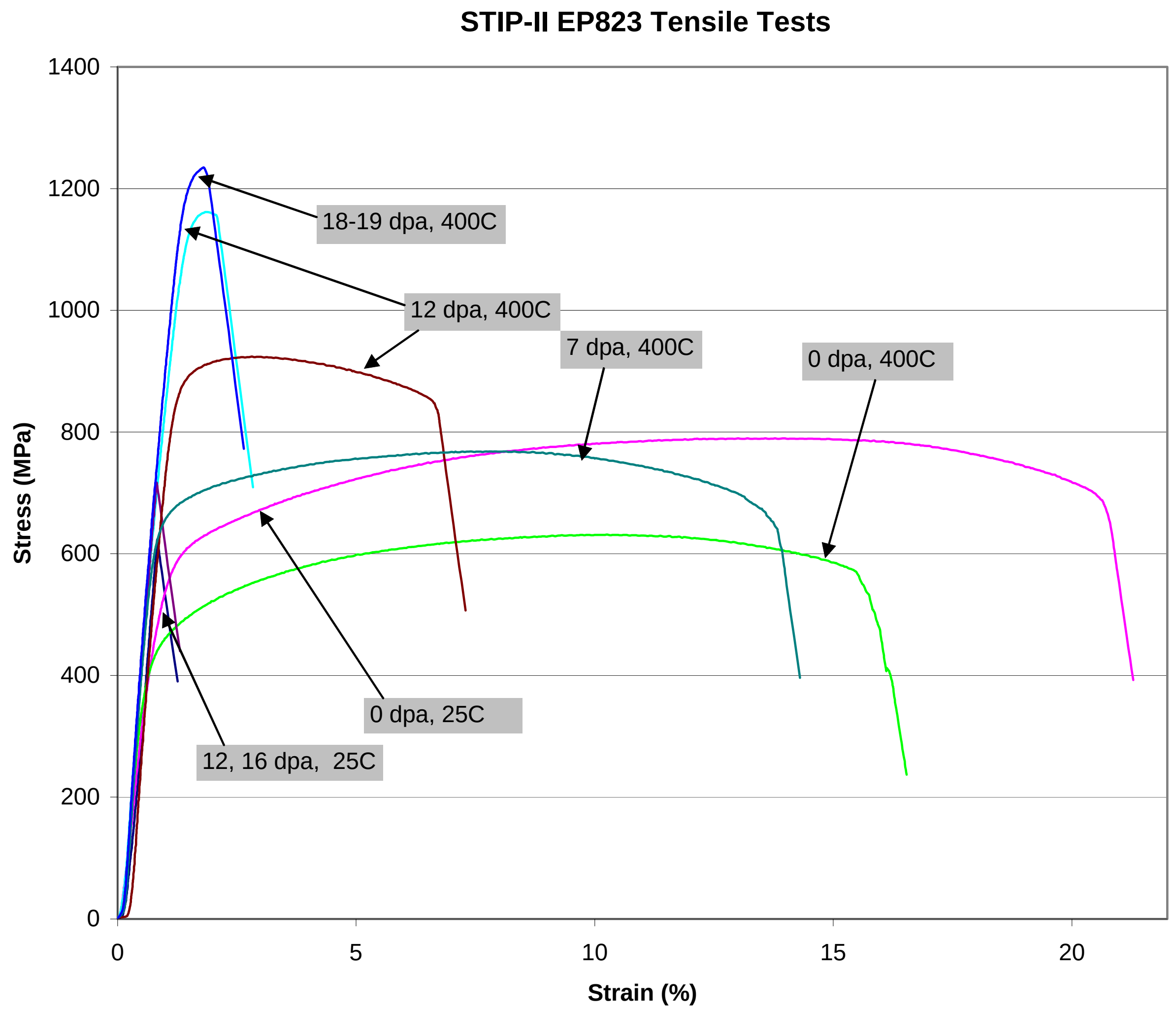
Collaborations

- DOE-CEA (WP-3), PSI, MEGAPIE
- Joint working group with Gen IV

Future Work

- Test FFTF irradiated HT-9 in tension at 400˚C at the LANL hot.
- Ship specimens to CEA for the MATRIX-SMI irradiation.

Highlight: Tensile Tests on STIP II irradiated EP-823 show Reduced Ductility at 400 and 25C



- Zero ductility observed for irradiated specimens tested at 25°C
- Uniform elongation reduced to less than 1% when tested at 400°C after irradiation to greater than 12 dpa

Materials Modeling

Small Sample Material Irradiations

Material Property Changes, Micro Structure Changes

Material Simulations, Development and Validation at the Atomistic Scale

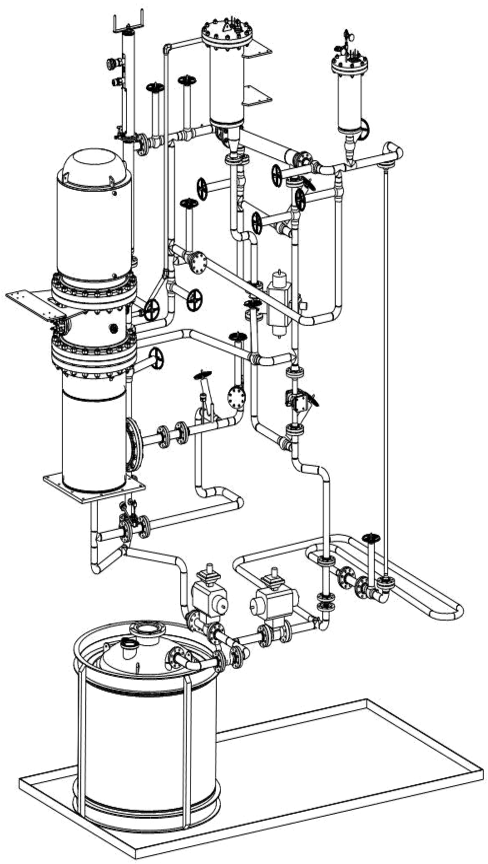
Improvements in Alloys, Definition of Data Needs

Issue: Need Fast Neutron Facility to Perform Needed Experiments

FY04 Highlight:

Initiated MEAM model development

LBE Technology Development is Centered at the DELTA Loop



DELTA Features:
- Test Bed for Component Development
- Removable Test Sections
- Natural Circulation

Highlights

- Structural alloy corrosion tests
- Oxygen sensor development and testing

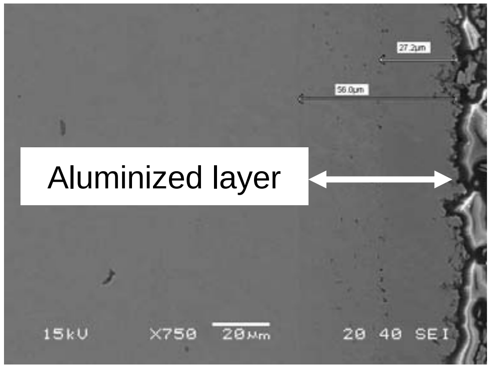
Collaborations

- University: UNLV, MIT, ISU
- OECD: International PbBi Hand book

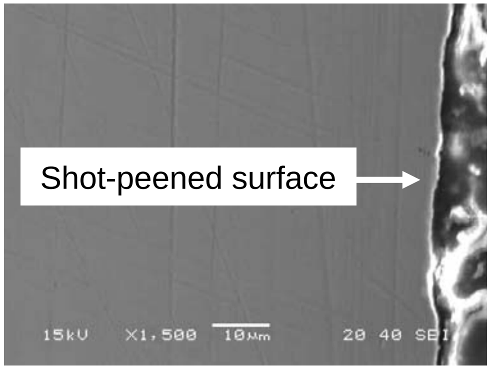
Plans

- Surface treatments development
- High temperature lead corrosion
- Engineering scale loop design for INL

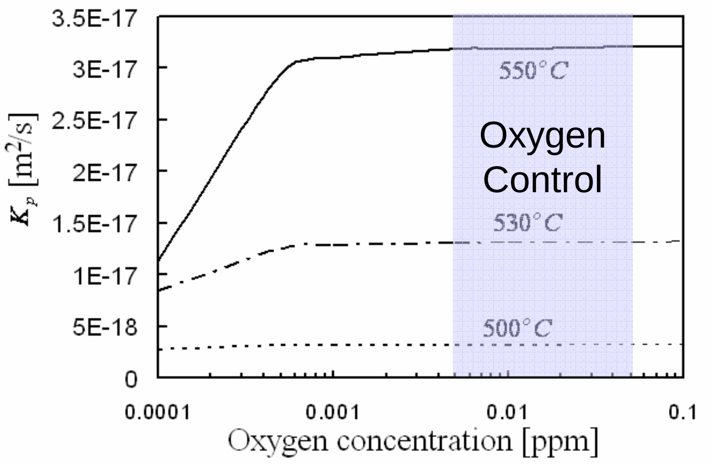
DELTA Corrosion Test at 520C Shows Enhanced Corrosion Resistance for Surface Treated Materials



Aluminized 316L, 267hrs



Shot-peened and smoothed 316L, 400hrs



Modeling Oxidation:
- Oxide structures
- Computation of oxide growth rate based on Wagner's theory

Future Plans

- LBE technology and materials TRL evaluation
- More surface treatments
- Alloys and functionally-graded materials development and testing
- High temperature lead corrosion
- Oxygen control through solid mass-exchanger
- Engineering Scale Test Loop at INL

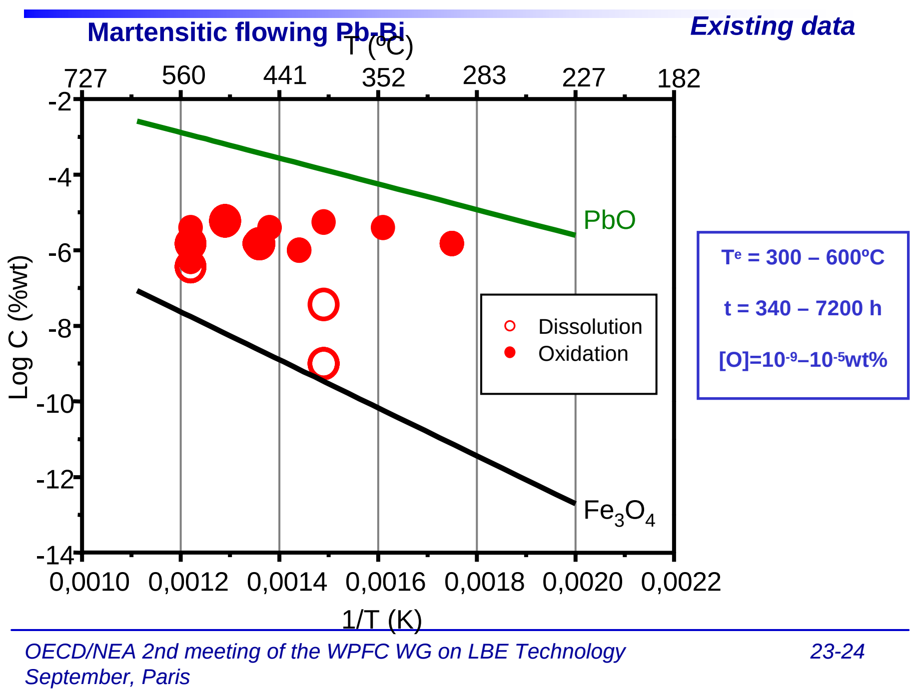
LBE Technology University and International Collaborations are Providing Useful Information

University Collaborations

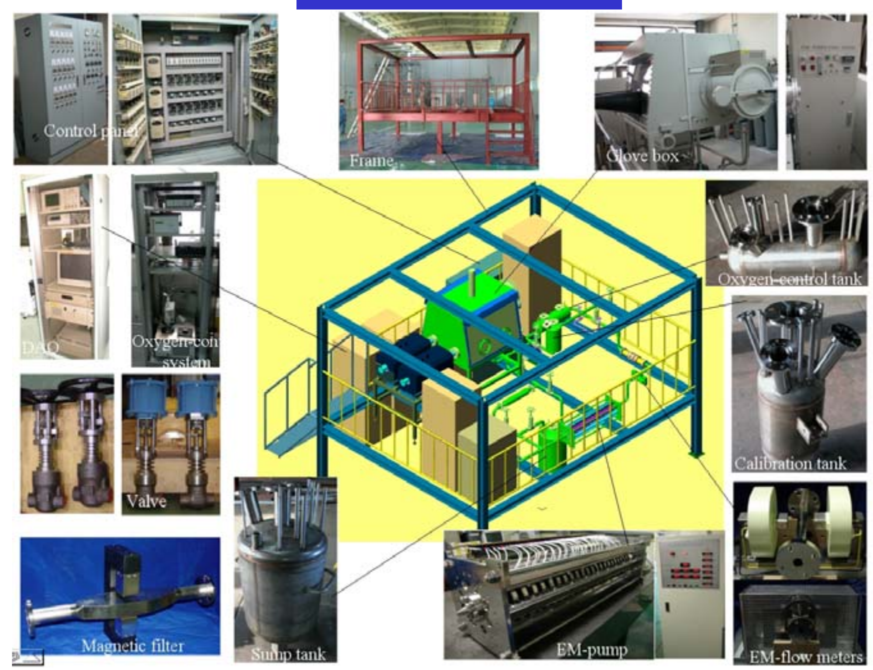
- UNLV TRP projects and LBE target loop installation and testing (with visiting IPPE specialists)
- MIT, ISU - oxygen sensor improvement and radiation effects testing
- UIUC - online corrosion measurement (transition to U-NERI)
- TAMU - nano-engineered surface coating for enhanced corrosion and radiation resistance (U-NERI)

International Collaborations

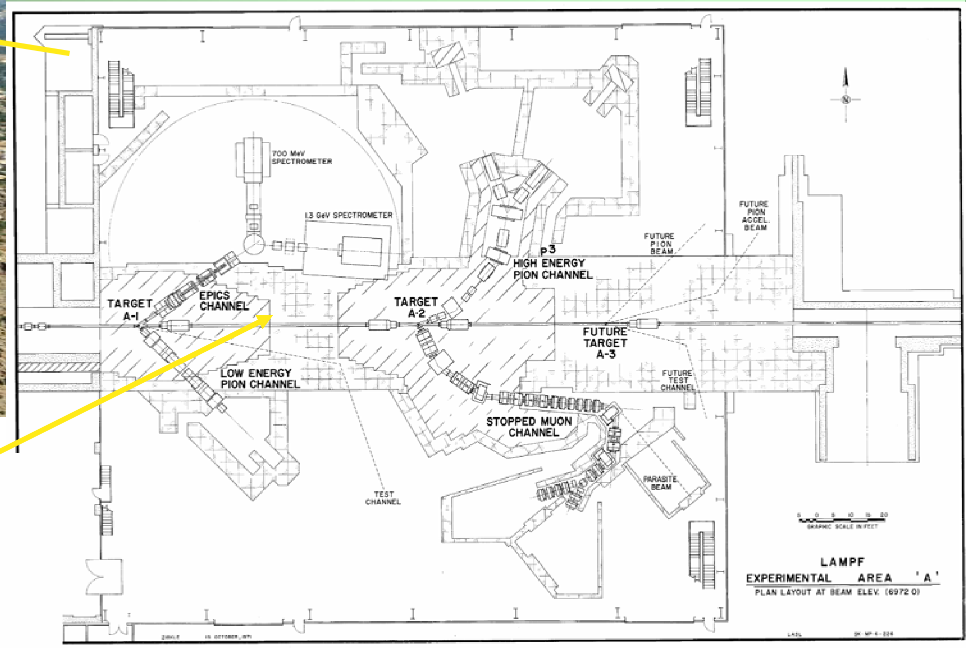
- OECD/NEA LBE handbook: over 70% completed for first draft
- I-NERI: KAERI/SNU LBE loop design and construction, oxygen sensors improvement

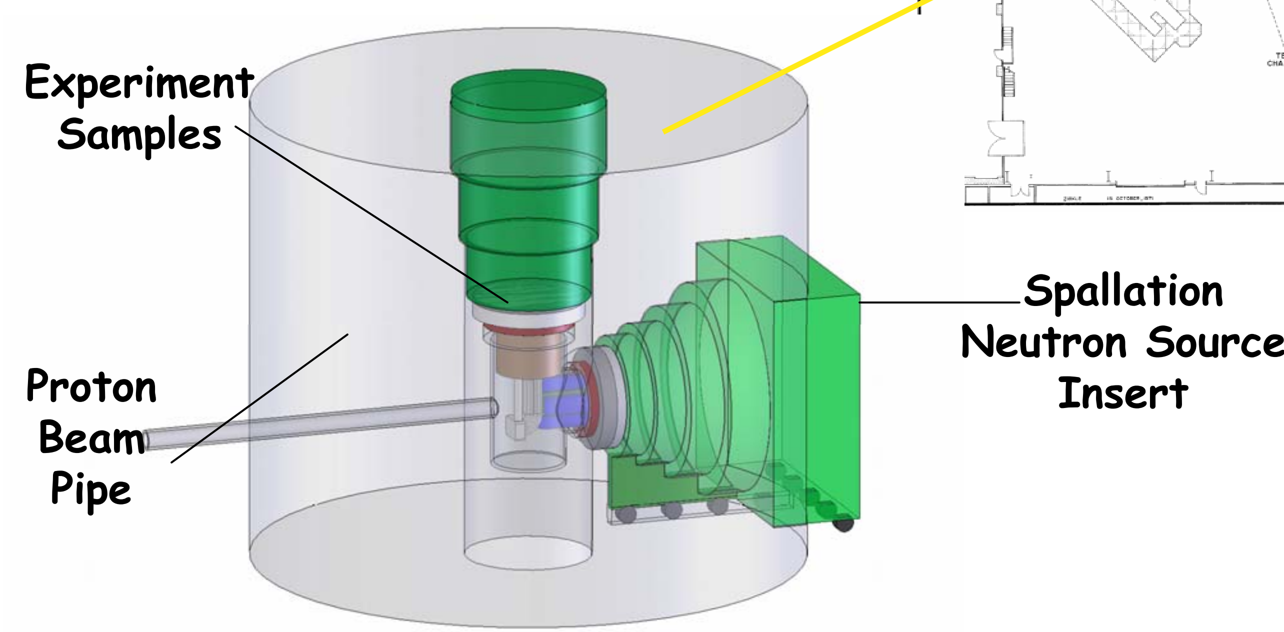


KAERI LBE Loop

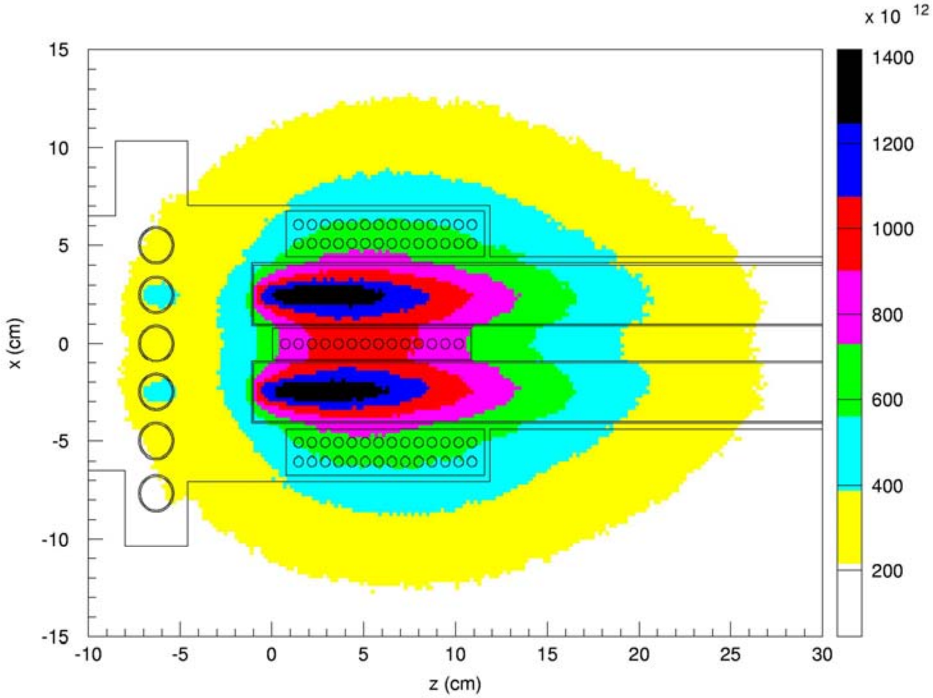


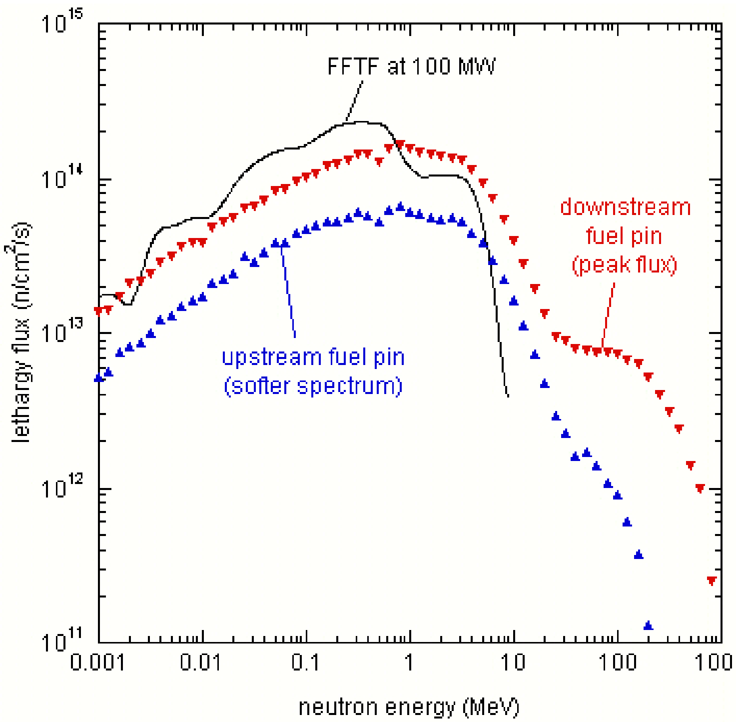
The Materials Test Station will be used for future irradiations





MTS Neutron flux intensity reaches 10^{15} n/cm^{2}s and is very similar to a fast spectrum reactor





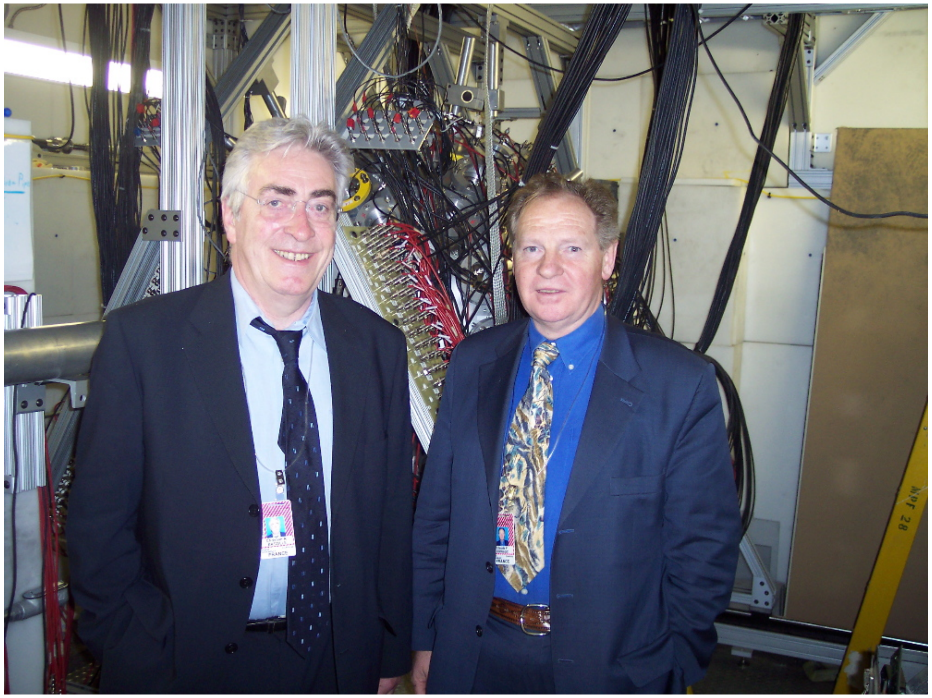
Transmutation ADS Continues with Foreign Collaborations

Highlights:

- OECD ADS Report
- TRADE Target and experiment support
- MUSE 1, 2 and 3 (completed)
- MUSE 4, in progress
- TRADE 1B, in progress

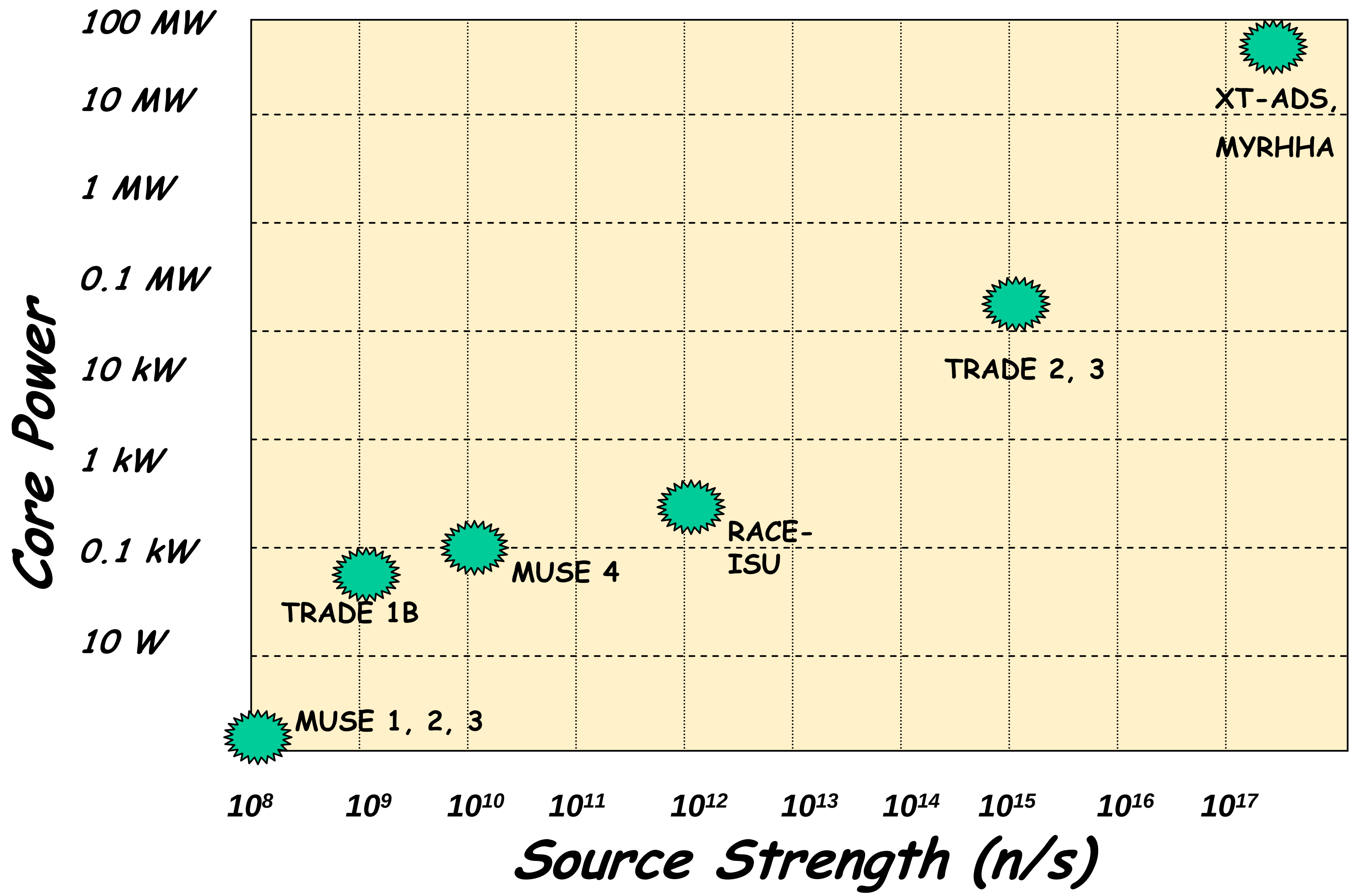
Plans:

- TRADE 1B, 2 and 3 Physics experiment lead (Imel)
- XADS Collaboration (Eurotrans)

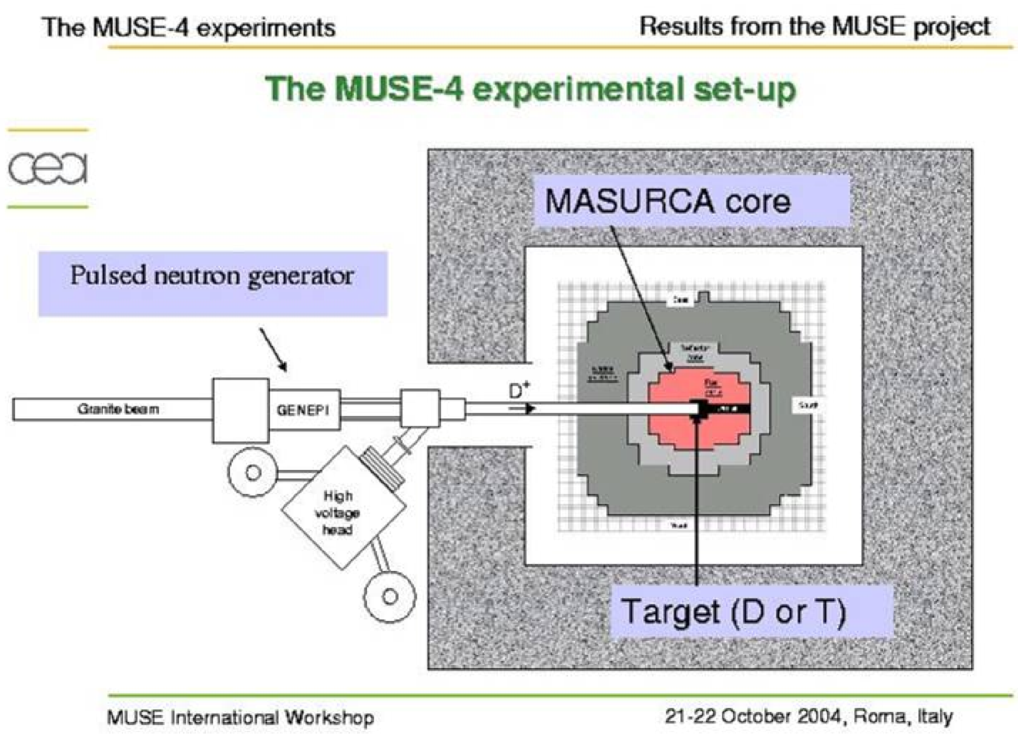


French members of parliament, Mr. Christian Bataille and Mr. Claude Birraux visit LANSCE.

Coupling Experiment Comparison

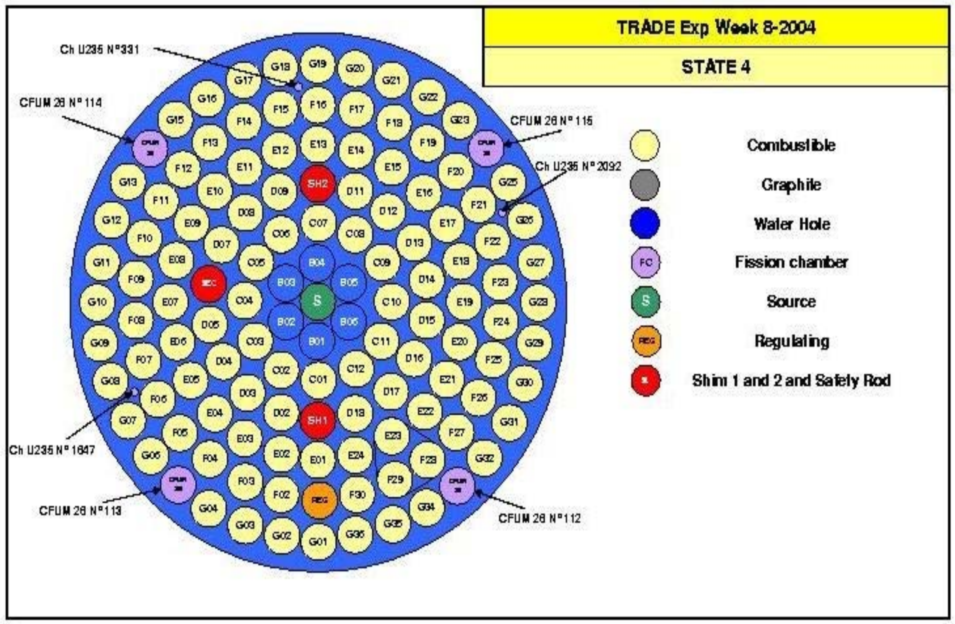


MUSE Accelerator/Reactor Coupling Experiments Completed for Several Sub-Critical Configurations



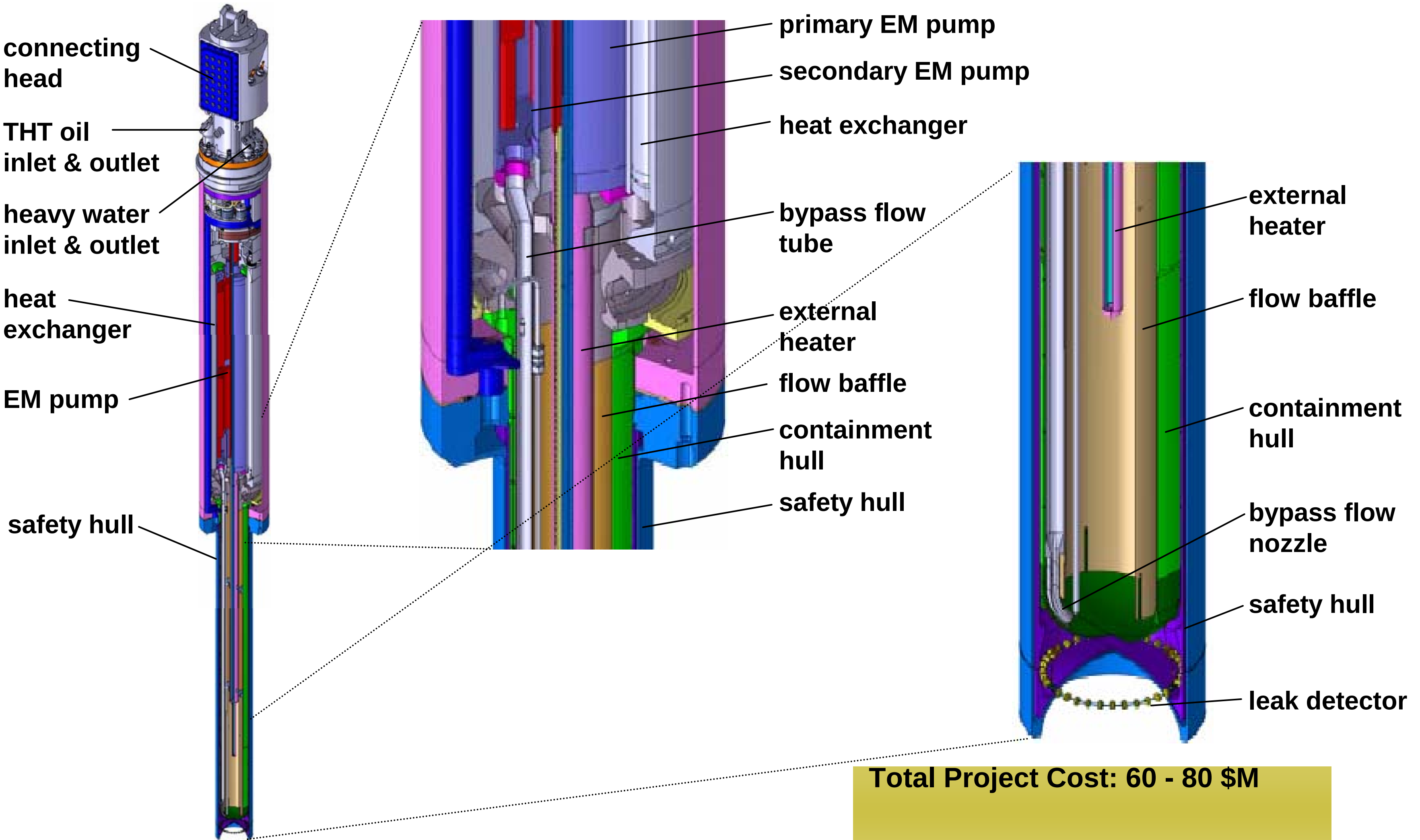
- Demonstrated the difficulties of sub-critical reactivity measurements to the degree of uncertainty we wish
- Demonstrated there is still much work to be done in space-time kinetics
- A medium for very important collaborations
- 9 PhD theses!

TRADE Phase 1 Experiments Show PNS Method Promising. TRADE Plus (with proton spallation source) will not be performed.



- A 3 month feasibility study has been started to find out what objectives of EuroTrans can be satisfied by other experiments.
- The RACE project (Idaho Accelerator Center, UT, TAM) is being seriously considered as a possible solution.

The MEGAPIE project will demonstrate a lead-bismuth spallation target at the Paul Shearer Institute in 2006

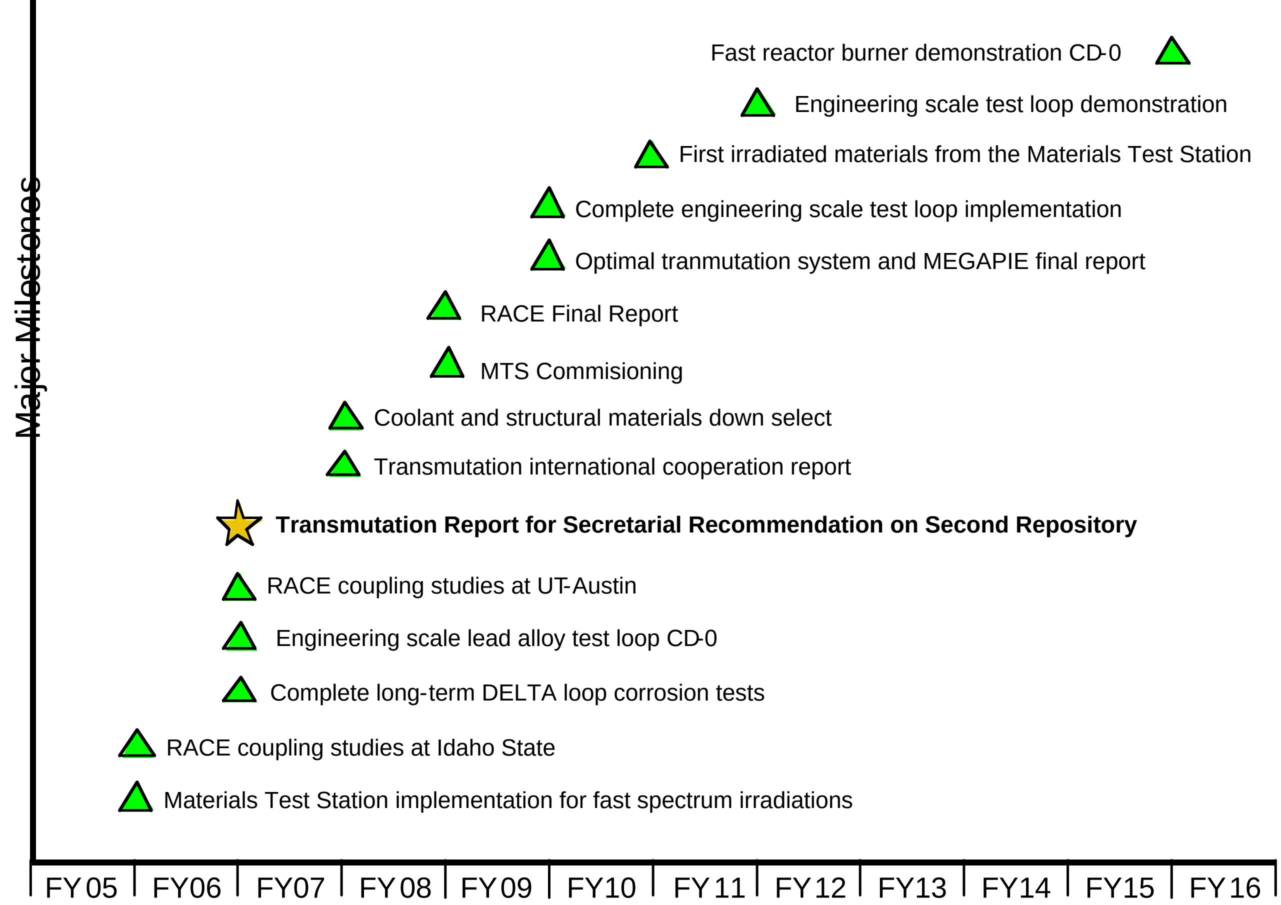


Total Project Cost: 60 - 80 $M

BUDGETS (FY04-05)

TASK AREA	PERFORMERS	FY 2004	FY 2005
Physics	LANL,ANL,BNL,ORNL	2123	1990
Structural Materials	LANL,PNNL	954	1185
Coolant Technology	LANL	1350	970
ADS	LANL,ANL	805	635
University	NCState,UF,UM,UIUC,UCB,MIT	562	562
Integration	LANL	326	
MTS	LANL, ANL, INL, ORNL		6944
Total		6120	11724

Transmutation Engineering Milestones Lead to a "Burner" Reactor Demo



Summary

- **Transmutation Science and Engineering plays an important role in AFCI and provides technical basis for transmutation.**
- **Fast Neutron Spectrum is most efficient for transmutation of actinides. MTS will provide irradiation environment for needed tests.**
- **Small scale experiments and modeling plays a fundamental role in large scale and end of life predictions.**
- **Research needs are extensive and will be accomplished thru National Labs and International and University collaborations.**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Paul Pickard

Gen IV Energy Conversion

Sandia National Labs

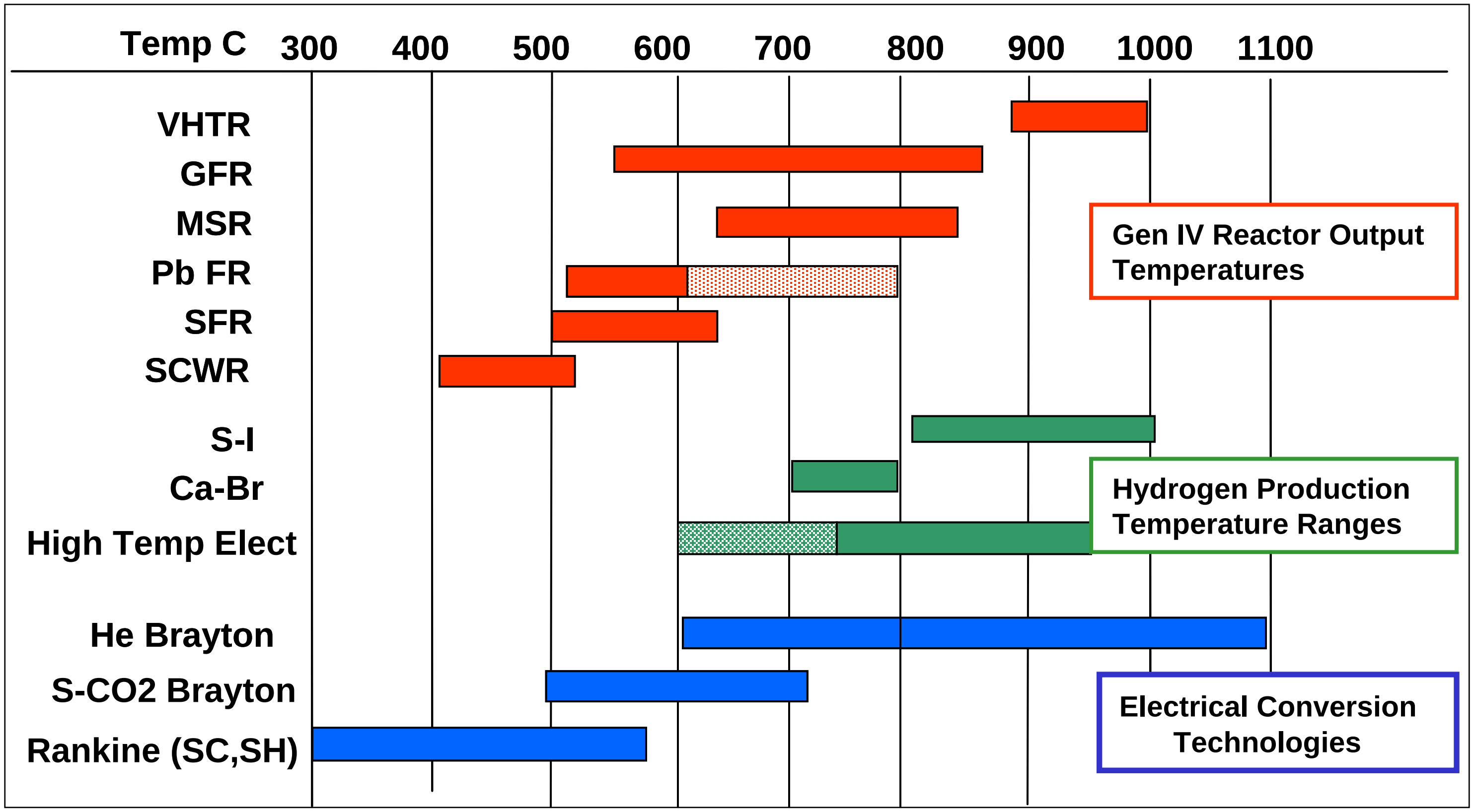
Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

GEN IV -- ENERGY CONVERSION RESEARCH AREA INCLUDES

- **Development of advanced power conversion options for Gen IV reactors**
  - **Higher efficiency, lower cost**
  - **Optimal coupling to Gen IV output characteristics**

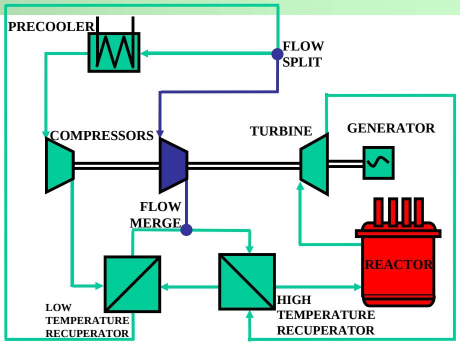
  Cost (\$/kW-hr) = (Capital cost recovery + Operating costs)/Electrical Output
- **Compare increased efficiency potential -capital cost impact**
- **Study Areas**
  - **Intermediate temperature systems -- Supercritical CO2 Cycle**
    - **GFR, LFR, SFR, MSR (550 -700°C)**
  - **Very high temperature systems – He Brayton Cycles**
    - **VHTR (1000°C)**
  - **Advanced Heat Transport**
    - **intermediate loop, indirect cycles, H2 production**

Generation IV Reactors – Higher Outlet Temperatures

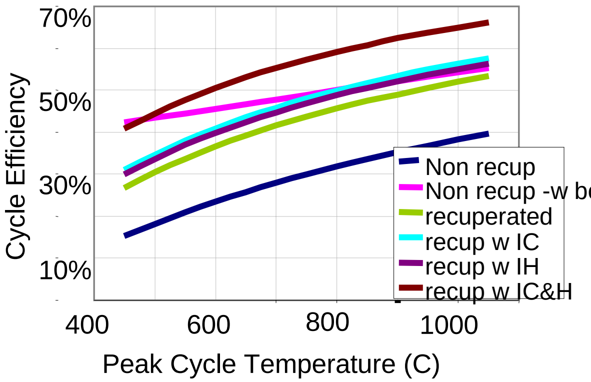


Gen IV Energy Conversion
FY05 Major Task Areas

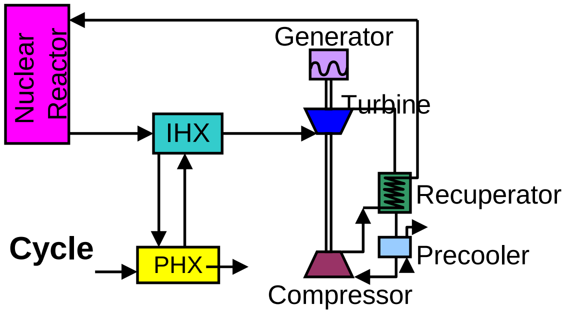
- Supercritical CO_2 Brayton cycles ~ 550-700°C. (GFR, LFR, SFR, MSR)
  - S-CO_2 turbo-machinery (TM) design
  - System design and cost assessment
  - Control strategies
  - Key technology scaling exp.design



- High-temperature He Brayton cycle (~1000°C. (VHTR)
  - Higher efficiency configurations
  - Cost benefit estimate



- Advanced Heat Transport
  - Direct, Indirect Elec. & H2, intermediate loop configurations, heat transfer medium options
  - Efficiency, cost, safety implications



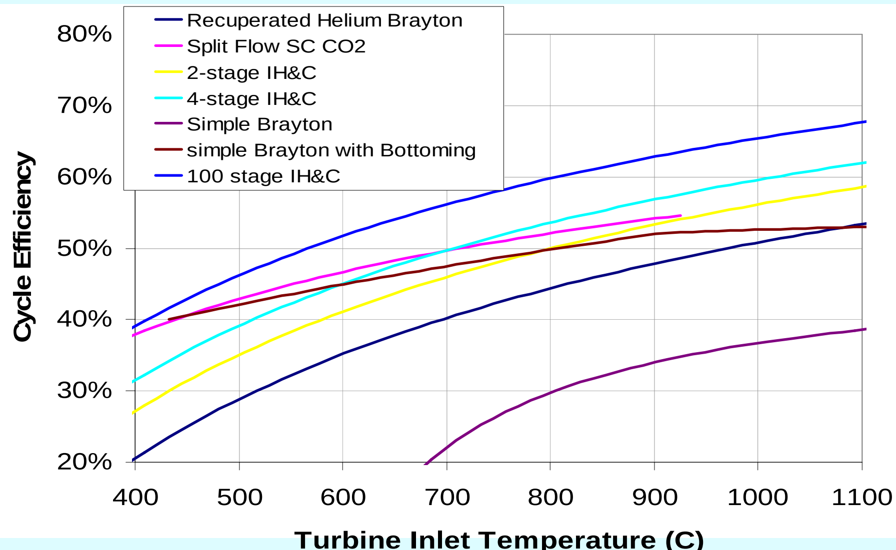
Direct Electrical Cycle and Serial HXs

Energy Conversion
FY-05 – 8 Work Packages

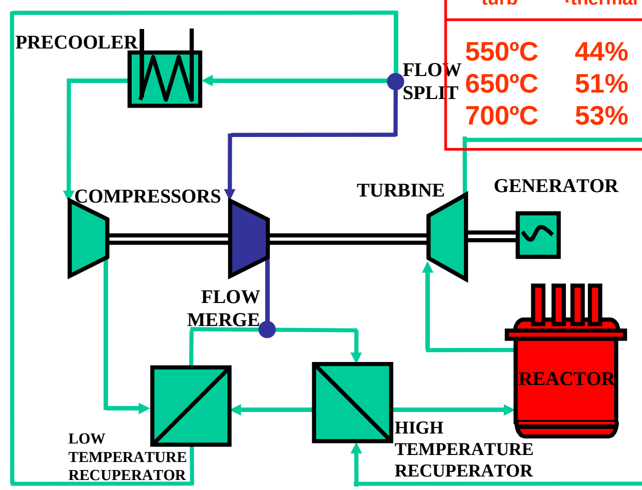
	FY05 Budget	Organization
1.10.01 – Program Coordination	$75 K	SNL
1.10.02 – Brayton Cycle Analysis	$80 K	SNL/UCB
1.10.03 – Supercritical CO2 Cycle	$220 K	MIT
1.10.03 – S-CO2 PCS Controls	$70 K	ANL
1.10.03 – S-CO2 TM Evaluation	$50 K	Industry(SNL)
1.10.04 – Adv Heat Transport Analysis	$55 K	INL
1.10.04 – Heat Transport Configurations	$20 K	SNL
1.10.04 – Adv Heat Transport Materials	$30 K	ORNL
TOTAL	$600K	7 Orgs

Supercritical CO_2 Cycle

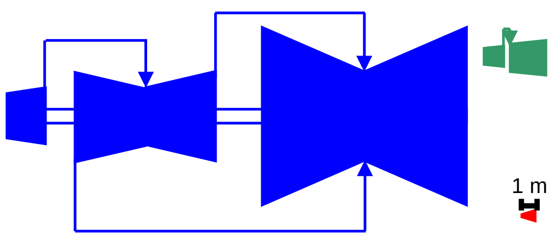
- High thermal efficiency at intermediate outlet temperatures, simple cycle



T_{turb}	$\eta_{thermal}$	η_{net}
550ºC	44%	40%
650ºC	51%	47%
700ºC	53%	49%



Compact turbo-machinery, but less effective heat exchange – still potential for lower capital cost

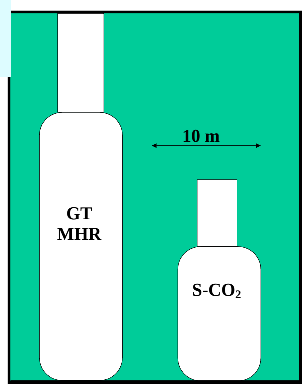


He Turbine (300 MWe)

1 m

S-CO2 (300 MWe)

Steam Turbine (250 MWe)



Key Issues:

- TM -main compressor, critical point operation
- Control strategy
- Heat exchanger eff. (recuperator)

Technical challenges for the S-CO2 cycle

- Higher Pressure (20 MPa) than He Brayton (7-10 MPa) (Less than supercritical steam fossil ~28 MPa)
- Optimizes at modest ΔT across heat source (150°C)
  - **Requires highly efficient recuperation (Heatric™ PCHE Units)**
- Requires attention to precooler & heat sink design because compressor is near CO_2 critical point (≈31°C)
- Recompression cycle requires two compressors, two recuperators
- Conventional control approaches need to be re-evaluated
  - **Two compressors in Parallel**
- Lack of industrial experience with high throughput axial compressors for CO_2 near its critical point (31°C, 7.3 MPa)

S-CO2 Power Conversion Study

- Highlights/Status:
- ***Completed design of turbine, main and recompressing compressors.***
  - ***Topical Report*** *"Supercritical CO2 Turbine and Compressor Design", (Wang, Guenette Hejzlar, Driscoll) March 2005 MIT GFR 015*
  - *Reoptimization of the cycle based on revised turbomachinery efficiency - (90 – 94 %) and very compact (~1m dia)*
- ***Revised cost (scoping) studies show potential for 15-28% savings versus advanced reactors using Rankine cycle and about 10% versus helium Brayton cycle***
- ***Evaluated S-CO2 cycle sensitivity to range of non-ideal operational conditions***
- ***Progress on PCS dynamics code to evaluate control and transient response. Modifying GASSPASS (ANL), (NIST gas property database, upgraded off-normal turbomachinery models)***
- ***Defined a base case 300kW pilot demo scale unit as a first step in planning for small scale testing of S-CO2 concept***
- ***Initiated TM industry contacts (Siemans, Barber Nichols, NREC)***

S-CO2 Study
Turbomachinery Design- Key Results

	Main compressor	Recompressing Compressor	Turbine
Number of stages	7	8	4
Maximum tip diameter (m)	0.5	0.9	1.2
Length (m)	0. 7	0.4	0.7
Total to total efficiency (%)	91.1	90.5	94.2

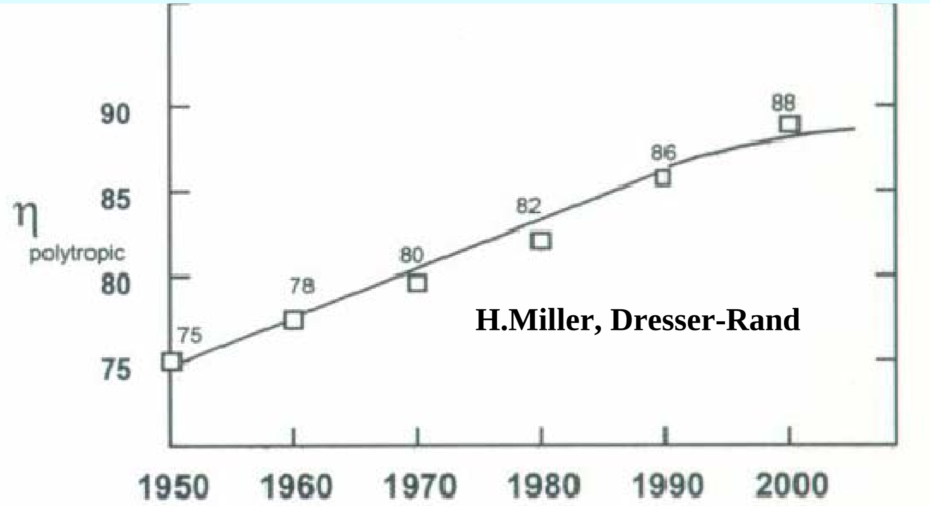
Turbomachinery is compact with higher calculated efficiencies than helium machines

S-CO2 –Small Scale Testing Options -- FY05 Status

- Main compressor – key demonstration issue
  - Issues: tip and profile losses (increase as size decreases)
  - Scaling to 250 kWth – small blade heights (~2.5 mm)
- Radial Compressor Options
  - More tolerant of S-CO2 variations, but lower efficiency
- Possible approaches
  - single stage of larger diameter,
  - radial designs
- Axial vs. radial **-** not clear if we can make
  - Radial big enough for full scale (300MWe)
  - Axial small enough for demo unit (300kWe)

Parameter	Full Scale	Small Scale
Power (kW)	62,000	250
Shaft Speed (RPM)	3600	56,520
Mass Flow (kg/s)	2574	10.3
Tip Radius (mm)	400	40
Blade Height (mm)	40	2.5

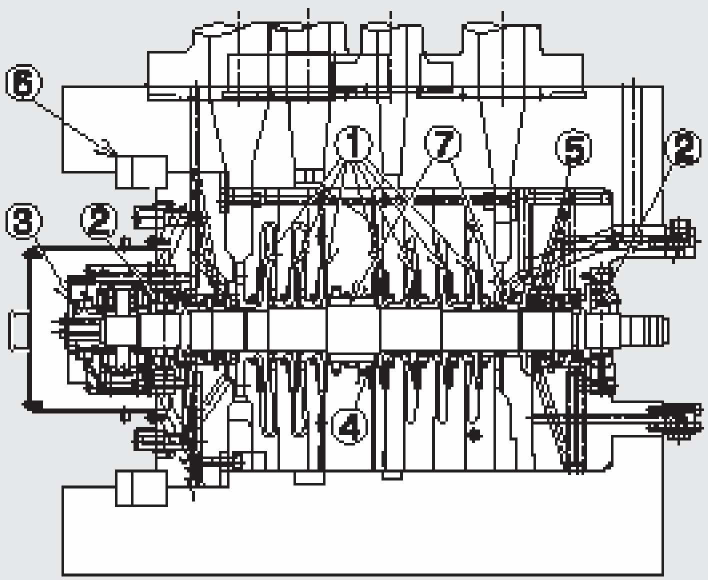
- Radial has larger operating range between stall and choke, but less efficient (~ 4%, - reduces cycle efficiency by ~1%). (axial compressor efficiency = 91%)



Radial vs. axial compressor
Largest CO2 radial compressor (MHI)

	MHI	MIT	MIT/MHI
Mass flow rate (kg/s)	280	3500	12x larger
Volumetric flow rate (m^3/s)	10	5	2 x smaller
Inlet/outlet pressure (MPa)	0.1/20	8/20	80 x smaller p ratio

MHI radial compressor for CO2 injection

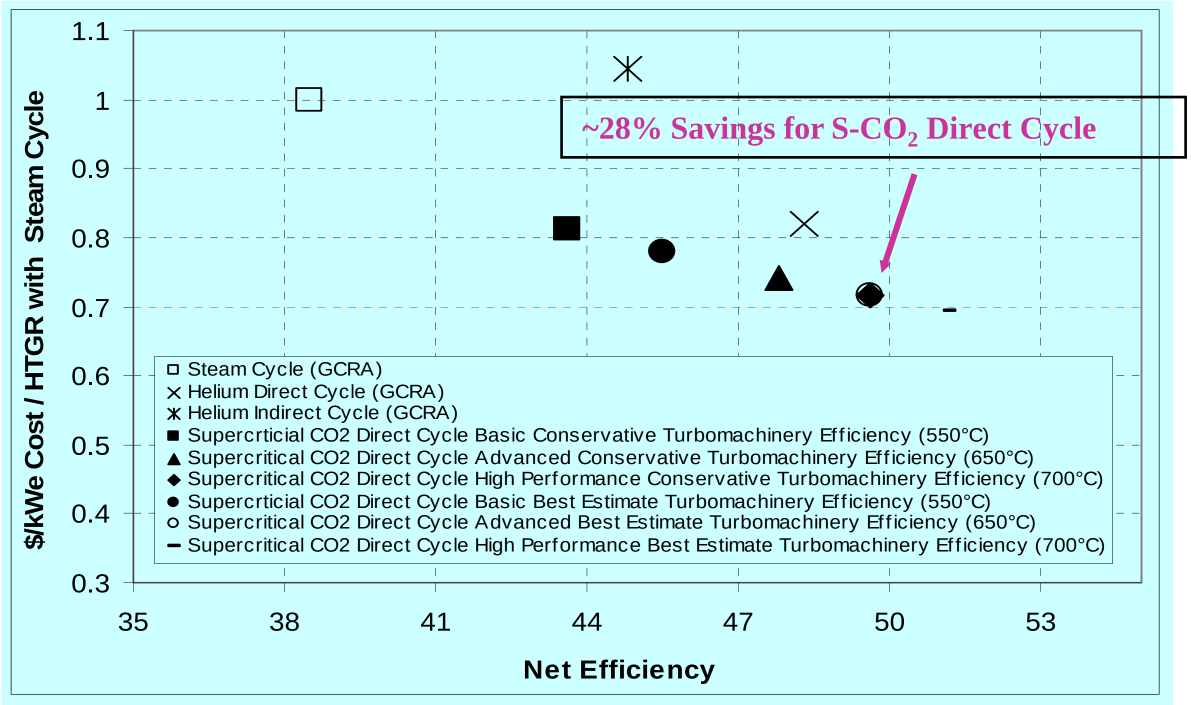


1. Three-dimensional impeller
2. Direct lubricating tilting pad type bearing
3. Overhang damper
4. Swirl preventive labyrinth seal
5. Dry gas seal
6. Shear ring (stop ring)
7. Balance piston

Motor 12MW, Pressure ratio=200
Vol. flow rate = 37100m^3/h

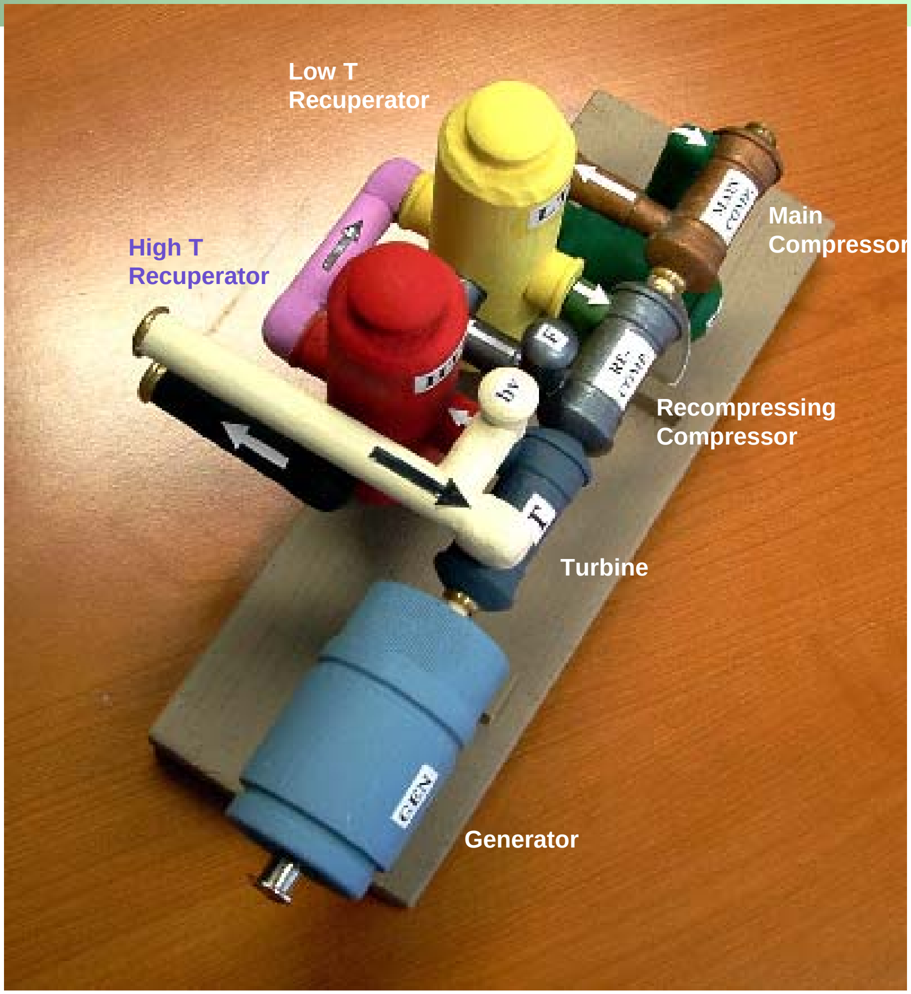
S-CO2 System Design and Cost Assessment

- ***Potential cost reductions 28 % (ref steam cycle)***
- ***Advantage improves with temperature***
- ***Direct vs. Indirect ~ 4 % eff, capital cost penalty, limited cost benefit***
- ***Point design required for reliable comparison***

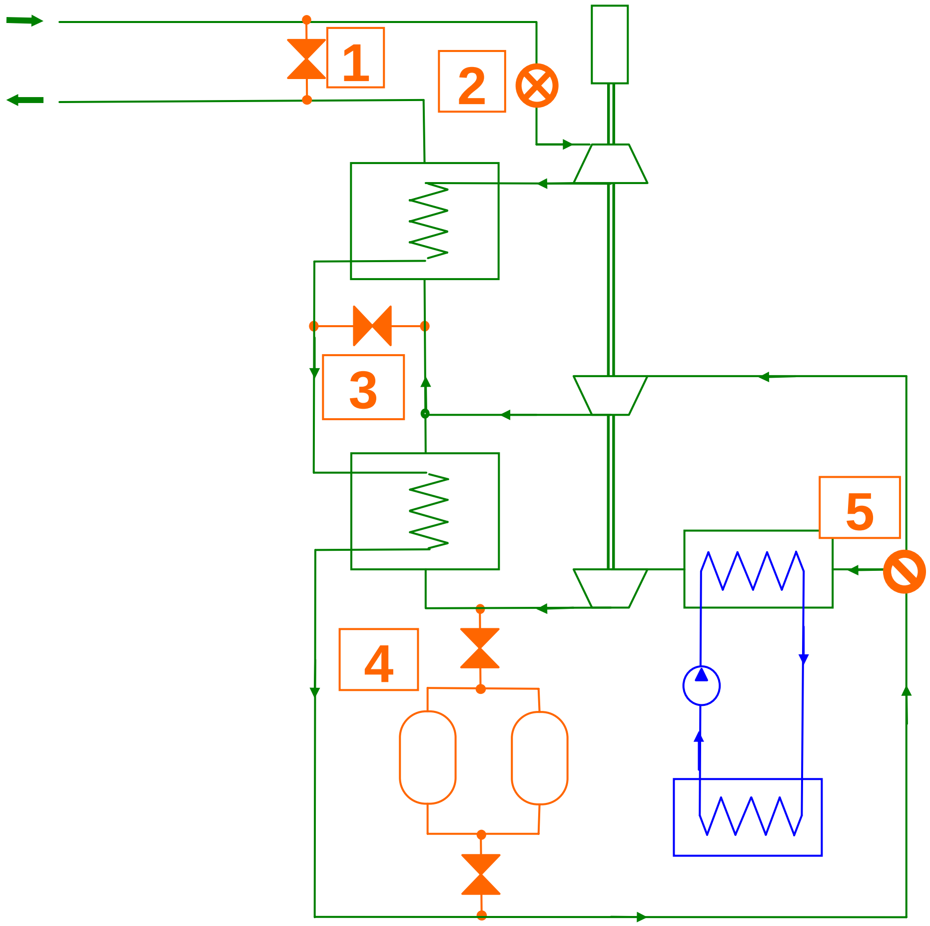


Current S-CO2 PCS Layout

- Completed assessment of four different power cycle component layouts
- (GT-MHR, PBMR, CEA, AREVA, ESCHER WYSS).
- Down-selected to a non-integral but compact single shaft arrangement (modified CEA S-CO2 layout)



Alternative Control Strategies for S-CO_2 Cycle (LFR)



1 – In-reactor heat exchanger bypass valve

2 – Turbine inlet/throttle valve

3 – Turbine bypass valve

4 – Inventory control tanks and valves

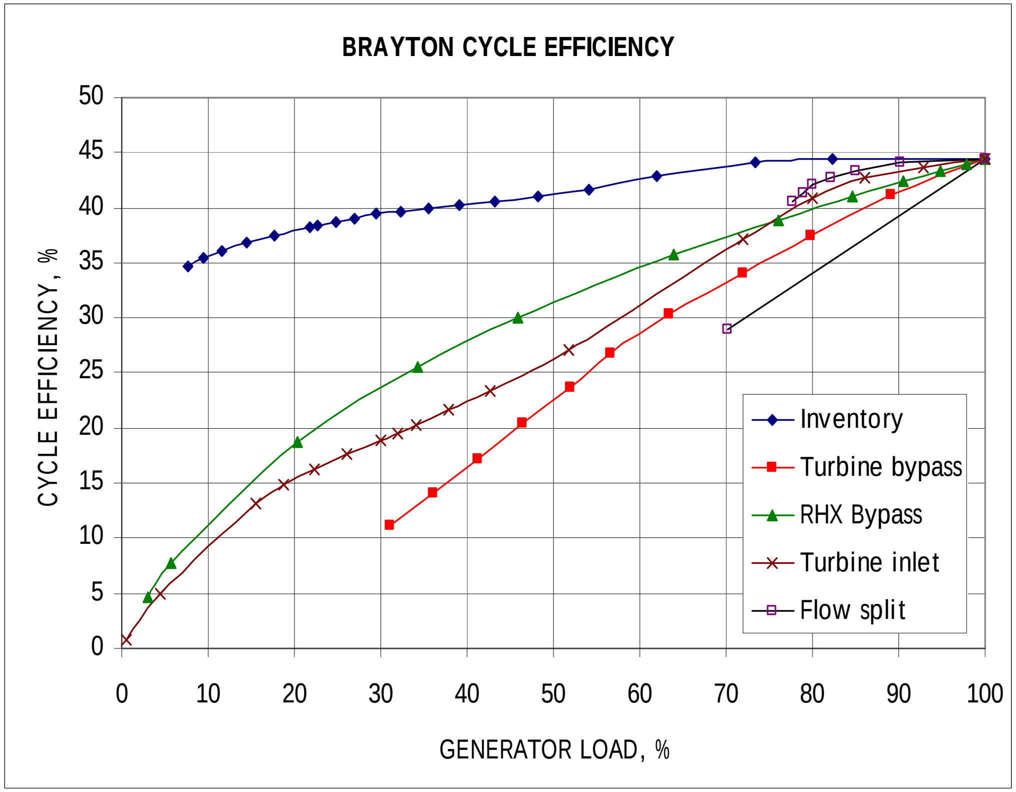
5 – Flow split valve

Alternative Control Strategies for $S\text{-}CO_2$ Cycle (LFR NERI results)

♦ Quasi-steady state analysis for turbine and compressors (common shaft)

•*Inventory control maintains a higher cycle efficiency than other strategies in upper power range, (not appropriate at low powers)*

•*Turbine inlet valve control or in-reactor heat exchanger bypass control effective over the lower range of power levels*



Supercritical CO_2 Cycle Next Steps - FY06 – FY08

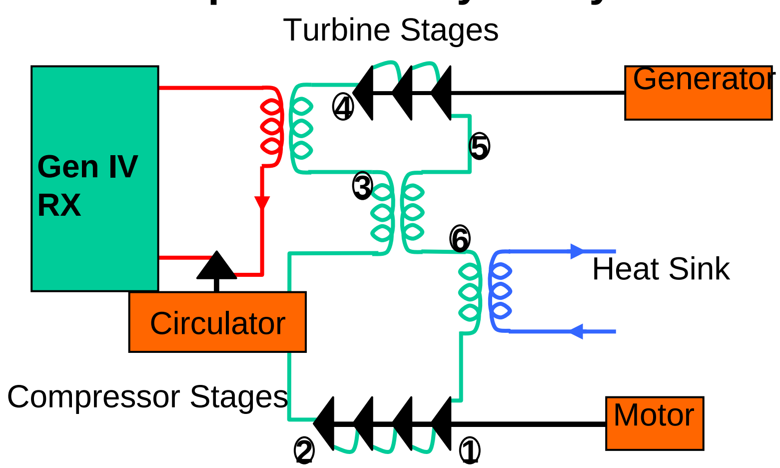
- Control and transient response evaluation, - no prior experience base on recompression cycle
- Complete analysis and recommend confirmatory experiments:
  - Basic few-stage lab tests to confirm operation of main compressor and aerodynamic loss coefficients
  - Conceptual design of small-scale PCS unit to confirm performance and controllability
- Industrial review/critique – R&D involvement (design, fabrication)
- Complete engineering analysis (pressure and thermal stress) of 300MWe PCS layout (engineering practice and ASME code requirements)
- Complete evaluation of radial vs. axial compressors for system and scaling experiments

High Temperature Brayton Cycles

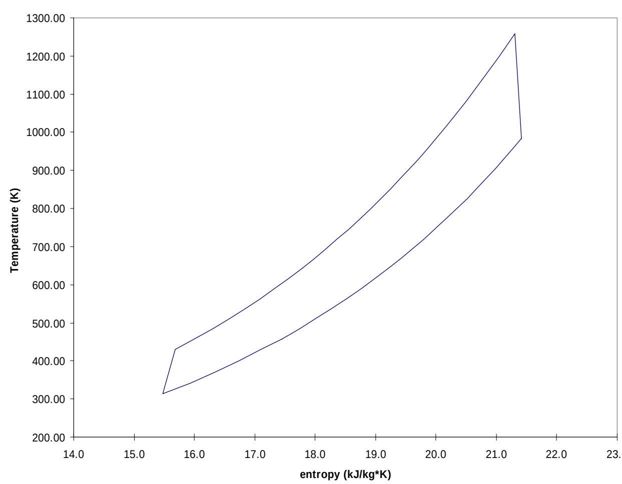
- Evaluation of potential efficiency improvements for high-temperature He Brayton cycles (SNL)
  - Preliminary analysis of cycle performance potential (FY04)
  - Preliminary cost estimates for improved efficiency configurations (FY04)
  - Conceptual and engineering system design for IH/IC systems (FY05-06)
  - Small scale component and system demonstration experiments (FY07- - -)

Interstage Heated/Cooled Brayton Cycle

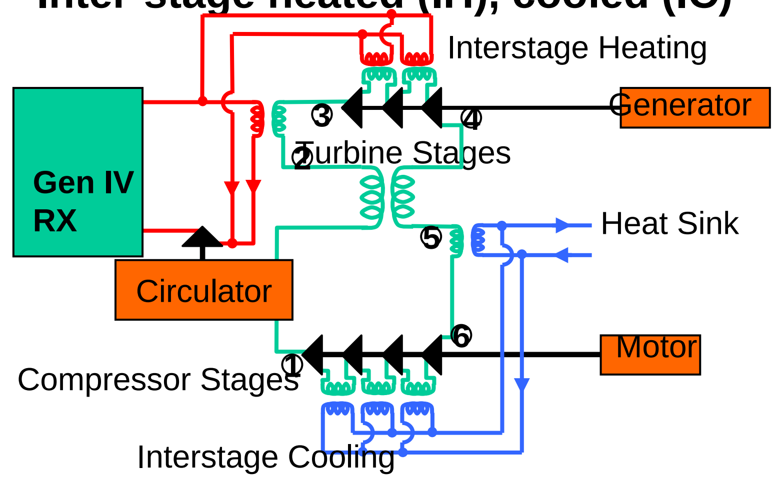
Recuperated Brayton Cycle



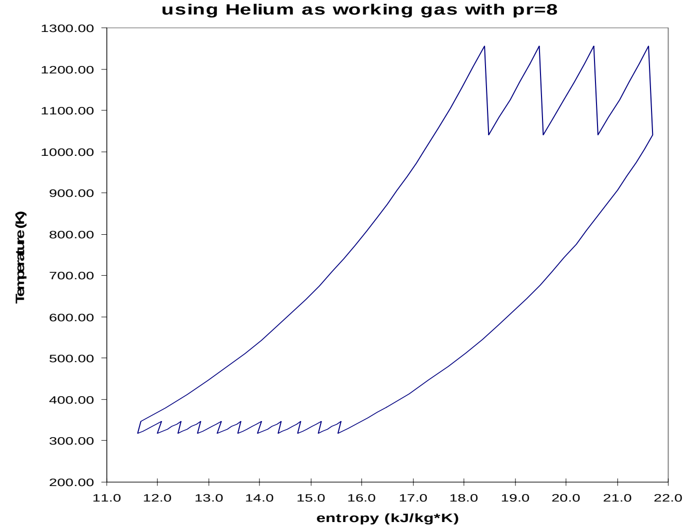
T-S Diagram for Recuperated Brayton cycle using Helium as working gas with pr=2



Inter-stage heated (IH), cooled (IC)



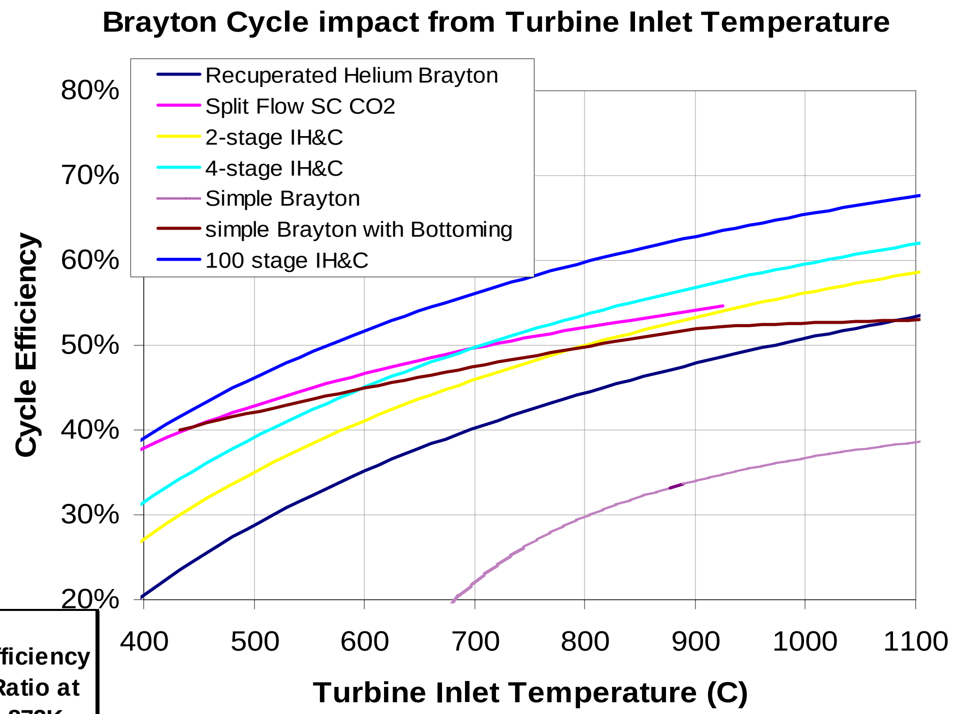
T-S plot for Recuperated Brayton cycle with interstage heating and coc using Helium as working gas with pr=8



High Temperature Brayton Cycle Analysis

IH / IC options effective at higher temperatures

S-CO2 – high efficiency at intermediate temperatures



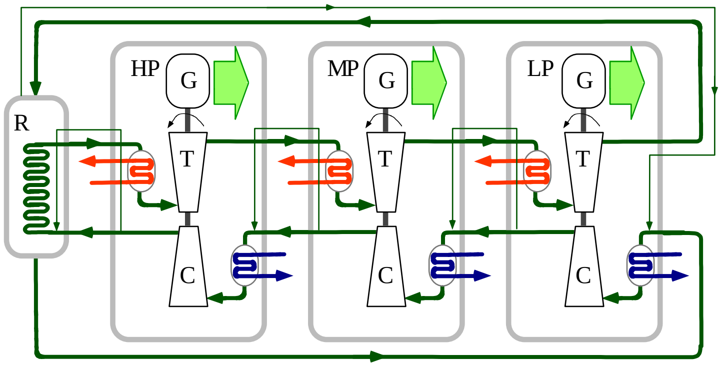
Cycle Modification	Efficiency at 1173K Source Temp	Efficiency Ratio at 1173K	Efficiency at 873K Source Temp	Efficiency Ratio at 873K
Reference (Indirect Recuperated Helium Brayton)	47.8%	1.00	35.2%	1.00
Combined Cycle (He Simple Brayton and SC Steam Rankine)	51.9%	1.09	44.9%	1.28
2 stage IH&C Brayton	53.3%	1.12	41.1%	1.17
4 stage IH&C Brayton	56.8%	1.19	45.6%	1.30
CO2 SC Split Flow Brayton	54.1%	1.13	46.6%	1.32
Recuperated CO2 (Ideal Gas) Brayton	50.0%	1.05	37.6%	1.07
Recuperated Nitrogen Brayton	49.4%	1.03	36.7%	1.04

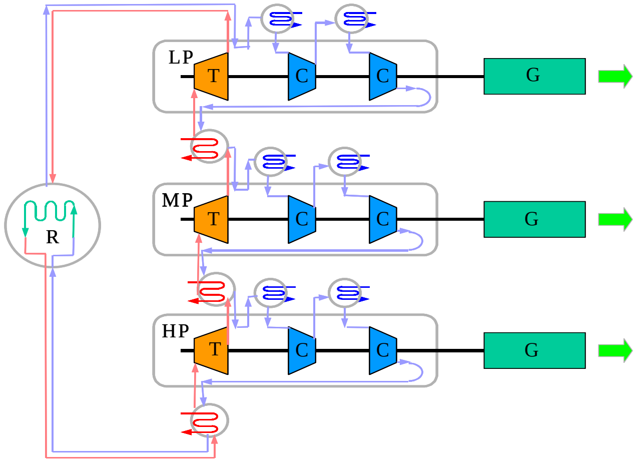
High Temperature Brayton Cycle
Interstage Heating / Cooling Cost Benefit

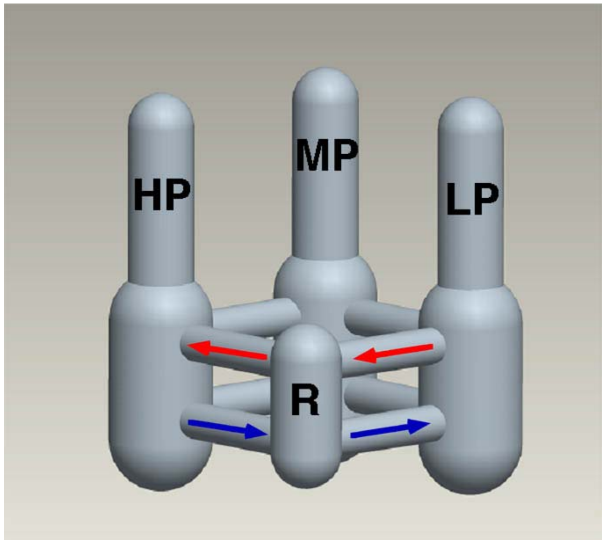
- **Issue is complexity/cost vs. efficiency improvement**
- **Configurations can simplify ducting, heat losses**
  - Single vs. multiple shaft
  - Vertical, horizontal
  - Integral or distributed units
- **Cost minimized by high specific power (small components)**
- **Efficiency maximized by optimized pressure ratio and high effectiveness components (size, cost)**
- **IH or IC - more complex – but cycle less sensitive to component effectiveness -- component total costs may not always be higher**
- **Compare efficiency and estimated costs for specific PCS module configurations**

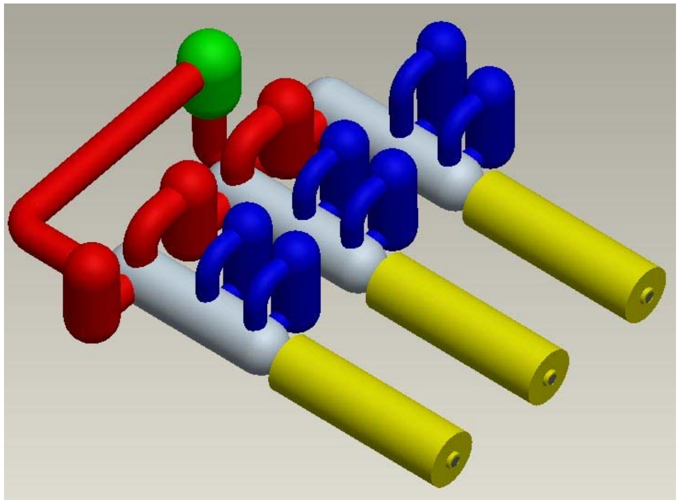
Candidate IH/IC Configurations

Multi Shaft - Modules









Brayton Cycle Options Analysis – Next Steps

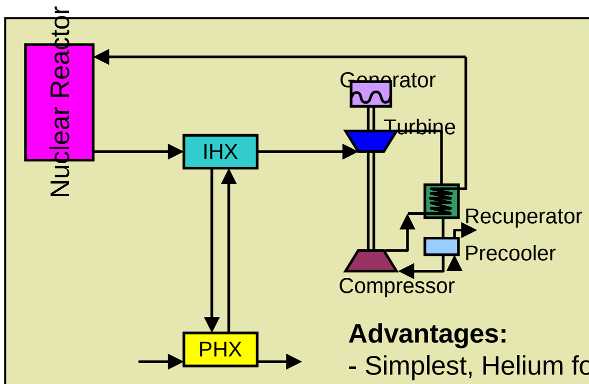
- Scoping performance analysis on improved efficiency designs
- Improved cost assessment models -(HX size, ducting, etc)
- Engineering designs, verify performance and cost ratios
- Evaluate key technology validation requirements for scaled system
- Innovative approaches for scaled experiments (HX, flow, turbomachinery, systems)

ENERGY CONVERSION
FY05 Advanced Heat Transport

- Task: Advanced Heat Transport Study
- INL, ORNL, SNL (NHI Interface task)

- Milestones for FY05
  - Complete assessment of intermediate loop configuration options and identify promising approaches (6/1/2005)
    - Heat transport configurations and intermediate loop medium heat transfer analysis (INL, SNL)
    - Molten salt materials assessment ORNL
- Status:
  - Basic heat transport configurations defined, scoping analysis initiated
  - Initial comparison of molten salt and He heat transfer medium characteristics

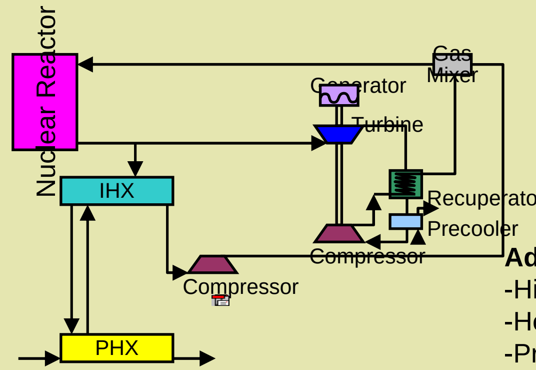
Possible Heat Transport Configurations



Advantages:
- Simplest, Helium for PHX

Disadvantages:
Less nuclear system isolation
Efficiency impact on H2 or PCS

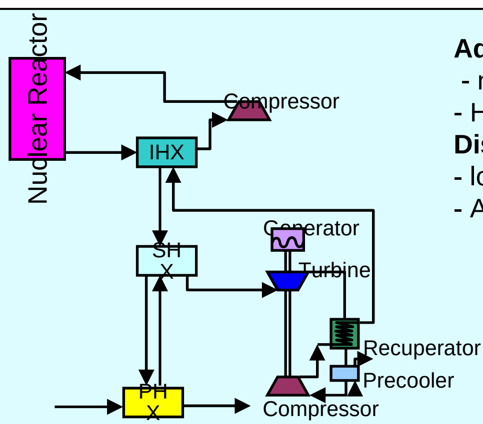
Direct Energy Cycle and Serial IHX



Advantages:
-Higher PCU efficiency
-Helium or MS for PHX
-Process isolation

Disadvantages:
Additional HX

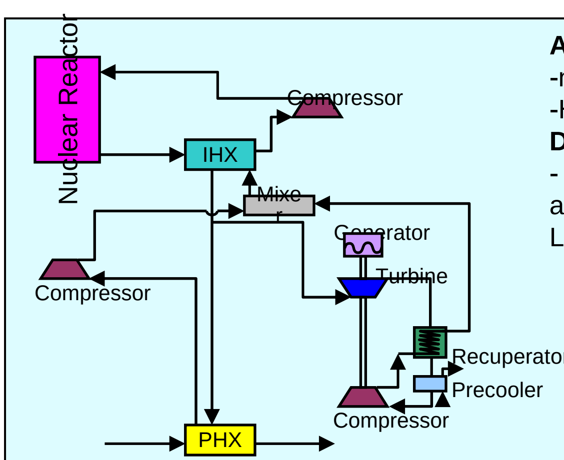
Direct Energy Cycle and Parallel IHX



Advantages:
- nuclear isolation
- He or MS for PHX

Disadvantages:
- lower cycle efficiency
- Additional HXs

Indirect Cycle and Serial IHX



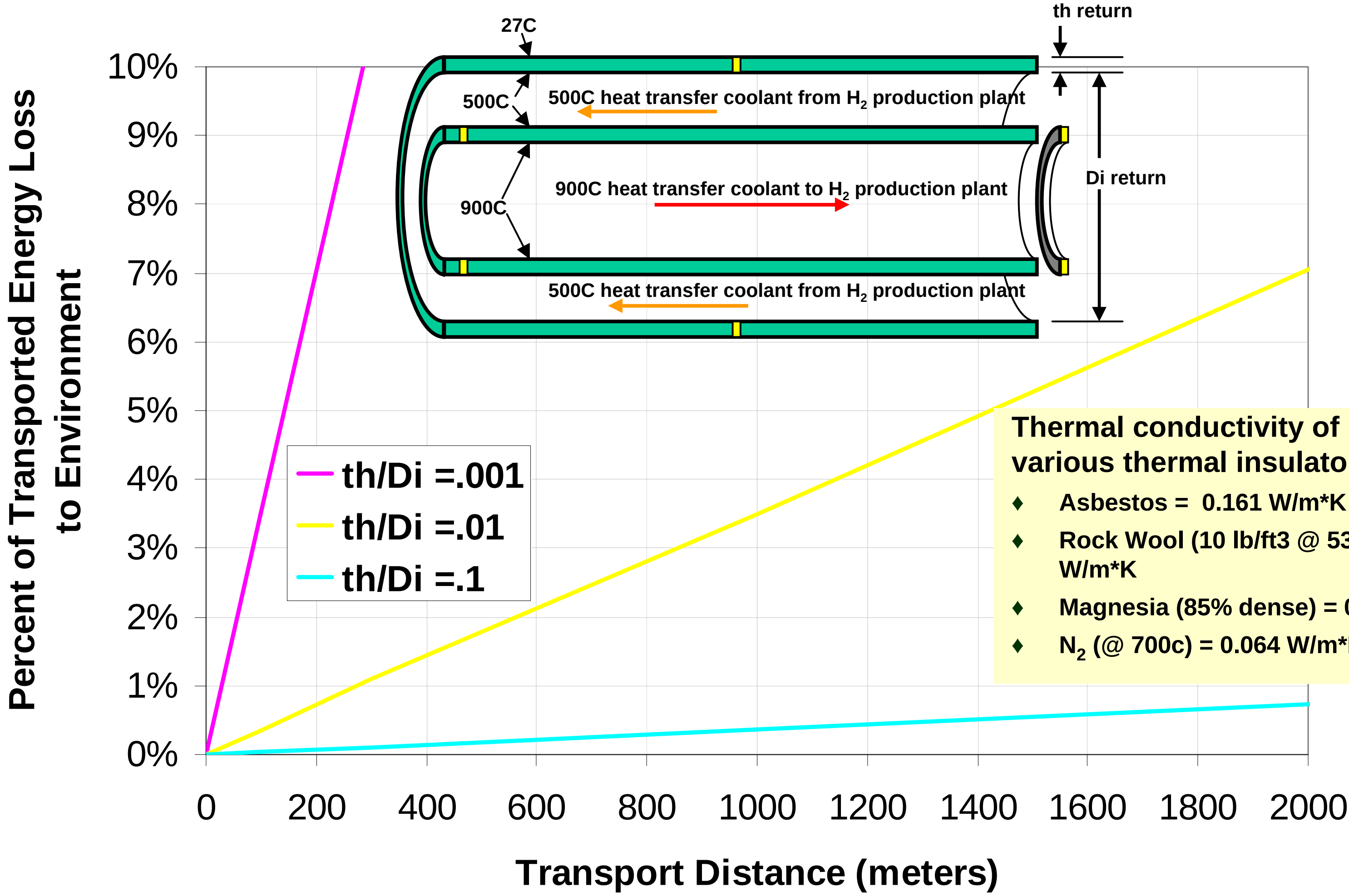
Advantages:
-nuclear isolation
-He for PHX

Disadvantages:
- Only helium between IHX and PHX
Lower cycle efficiency

Indirect Cycle and Parallel PHX

Heat Loss From an Intermediate Heat Transport Loop Appears Manageable

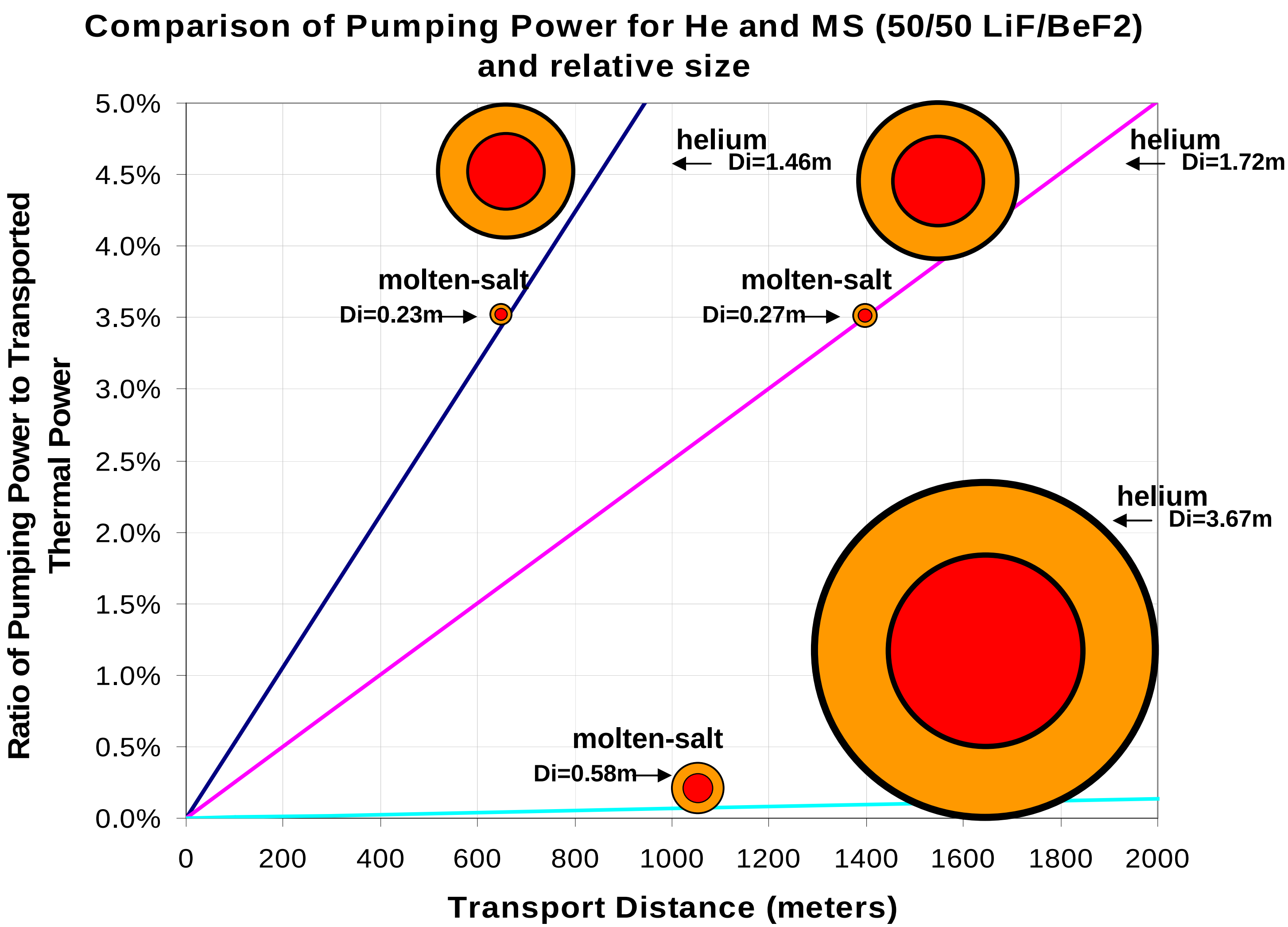
Thermal loss as a function of transport distance for various insulation assumptions - k(insulator)=0.1 W/m*C, Q(transported)=600MW



Thermal conductivity of various thermal insulators

- Asbestos = 0.161 W/m*K
- Rock Wool (10 lb/ft3 @ 533c) = 0.087 W/m*K
- Magnesia (85% dense) = 0.080 W/m*K
- N_2 (@ 700c) = 0.064 W/m*K

Pumping Power Requirements for Thermal Transport - He and Molten Salt Comparison



Heat Transport Configurations - Next Steps

- Complete scoping analyses on pumping powers, heat losses, impact on efficiency, engineering implications, preliminary cost estimates
- Quantify heat transport parameters (losses, pumping req's, P and T drop differences) for representative configurations)
- Develop conceptual design(s) for selected configurations
- Identify materials (liquid salt), heat exchanger experimental requirements

BUDGETS FY05 and Beyond

Year	EC Plan Budget
FY05	600 K$
FY06	1000K$
FY07	1000K$
FY08	2000K$

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

James J. Laidler

National Technical Director

AFCI Separations Technology Development

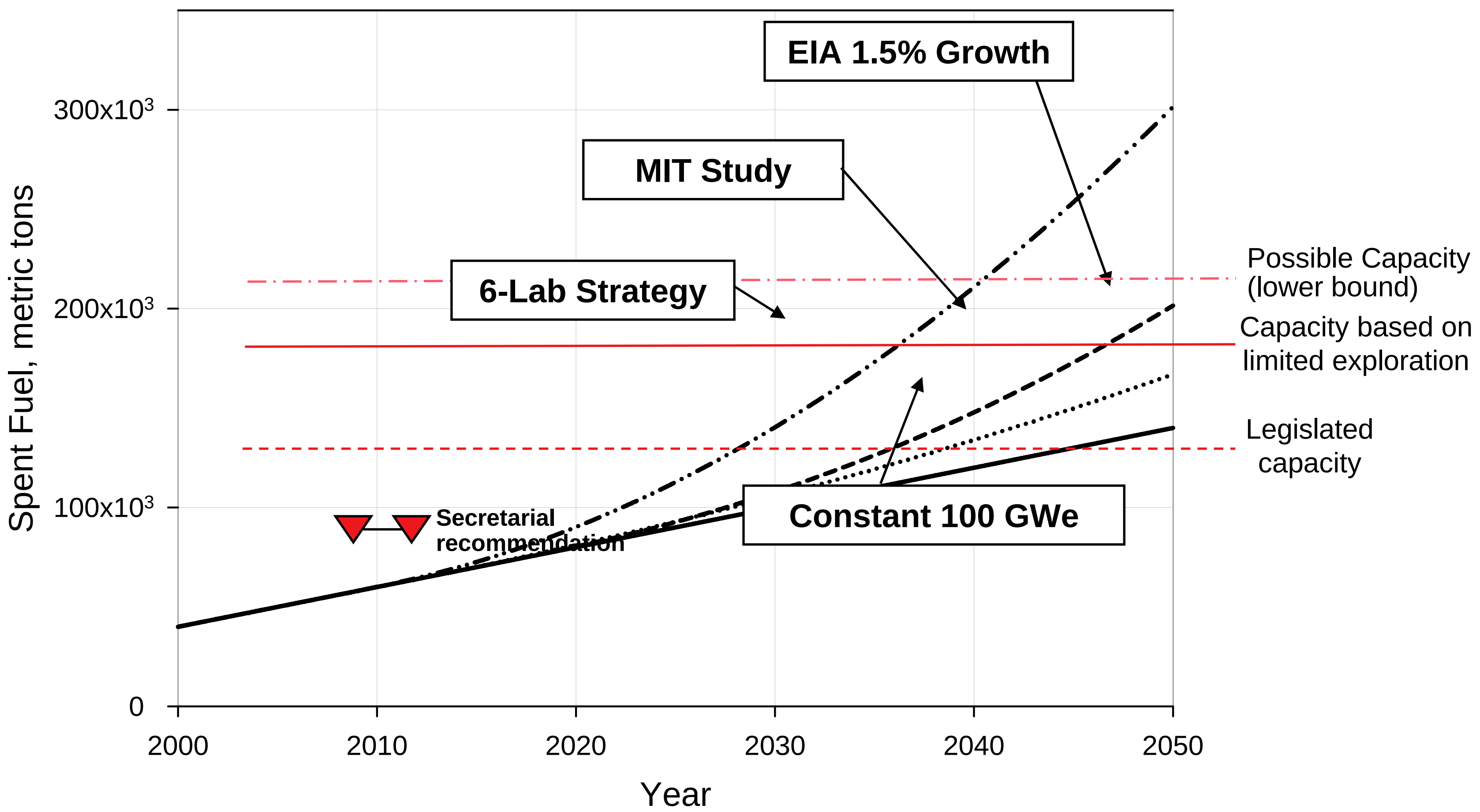
Argonne National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Program Objectives

- Preclude or significantly delay the need for a second geologic repository in this century
- Reduce the volume of high-level nuclear waste requiring repository disposal and lower the cost of its disposal
- Separate long-lived highly radiotoxic transuranic elements for destruction by fissioning in thermal or fast spectrum reactors
- Reclaim the valuable energy content of spent nuclear fuel
- Reduce the proliferation risk of the closed nuclear fuel cycle
- Facilitate the closure of the Generation IV fuel cycle and ensure the sustainability of nuclear energy

Projected Spent Fuel Accumulation without Processing



Program Elements – Separations Technology

- Develop suite of aqueous processes for LWR spent fuel processing
- Develop specialized pyrochemical processes for fuels expected to be difficult to process by aqueous means
- Develop storage methods for product streams
- Develop improved waste forms for disposal of high-level waste

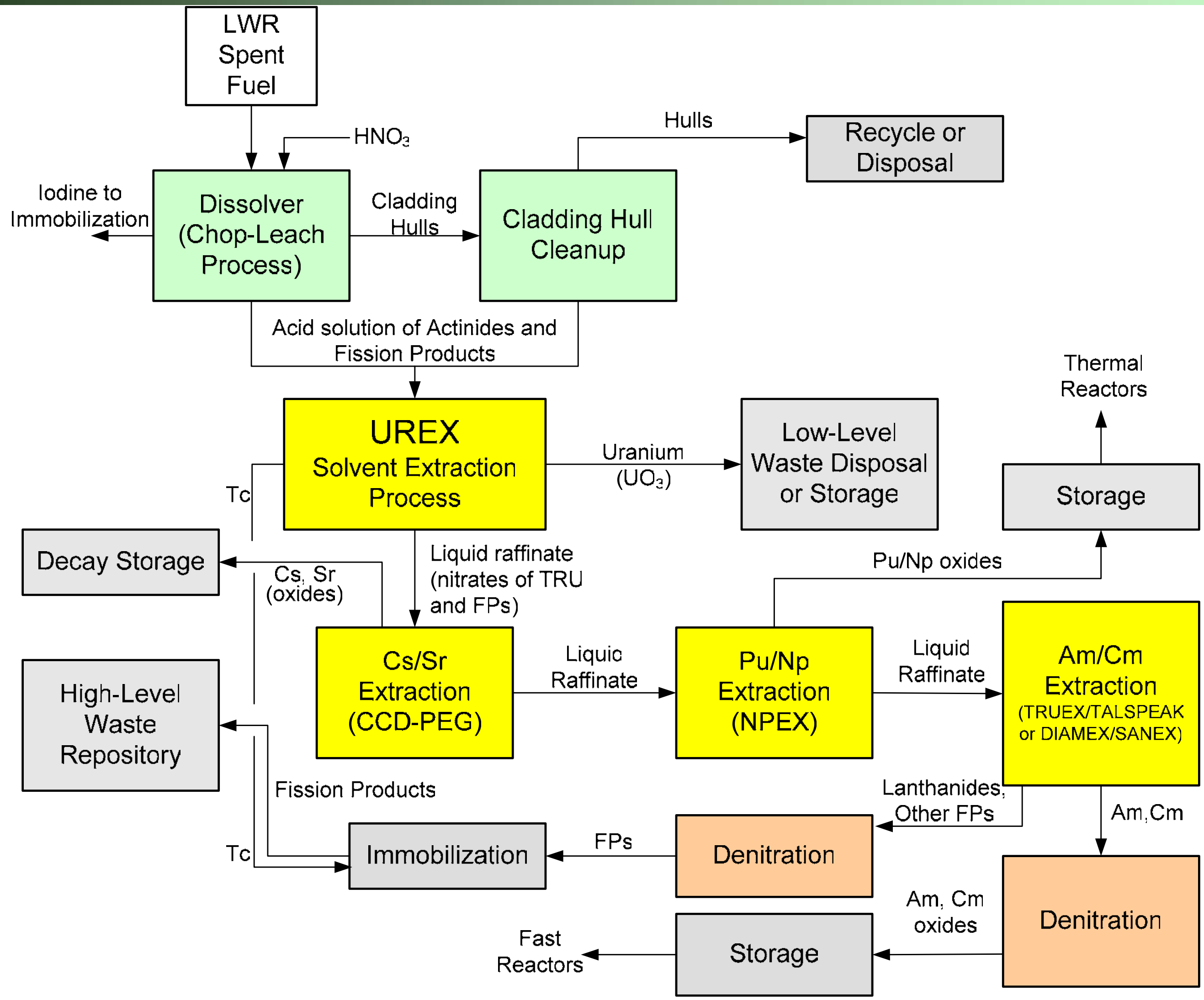
Treatment Technology Choices

- **Aqueous processing has the necessary technological maturity for deployment in the next 30-40 years**
  - Very long lead time to large plant operation (up to 25 years)
  - Minimum technical and financial risk
  - Continuous process (vs. batch)
  - Economy of scale
- **If spent fuel processing is introduced in the U.S., there will be a need for sequential construction of multiple large plants**
  - Allows for introduction of advanced technologies when they are proven feasible
- **U.S. is in the position of being able to design a plant from the ground up with advanced safeguards technologies**
  - Time is available to demonstrate the practicality of these advanced technologies and gain international endorsement

UREX+ Process for LWR Spent Fuel Treatment

- Separate pure uranium for disposal as low-level waste or storage for re-use
- Separate cesium and strontium in pure form to eliminate short-term heat load on repository
- Recover transuranics for recycle to thermal or fast spectrum systems

Basic UREX+ Process



Suite of UREX+ Processes

Process	*Prod #1*	*Prod #2*	*Prod #3*	*Prod #4*	*Prod #5*	*Prod #6*	*Prod #7*
UREX+1	U	Tc	Cs/Sr	TRU+Ln	FP		
UREX+1a	U	Tc	Cs/Sr	TRU	All FP		
UREX+2	U	Tc	Cs/Sr	Pu+Np	Am+Cm+Ln	FP	
UREX+3	U	Tc	Cs/Sr	Pu+Np	Am+Cm	All FP	
UREX+4	U	Tc	Cs/Sr	Pu+Np	Am	Cm	All FP

Notes: (1) in all cases, iodine is removed as an off-gas from the dissolution process.
(2) processes are designed for the generation of no liquid high-level wastes

U: uranium (removed in order to reduce the mass and volume of high-level waste)
Tc: technetium (long-lived fission product, prime contributor to long-term dose at Yucca Mountain)
Cs/Sr: cesium and strontium (primary short-term heat generators; repository impact)
TRU: transuranic elements (Pu: plutonium, Np: neptunium, Am: americium, Cm: curium)
Ln: lanthanide (rare earth) fission products
FP: fission products other than cesium, strontium, technetium, iodine, and the lanthanides

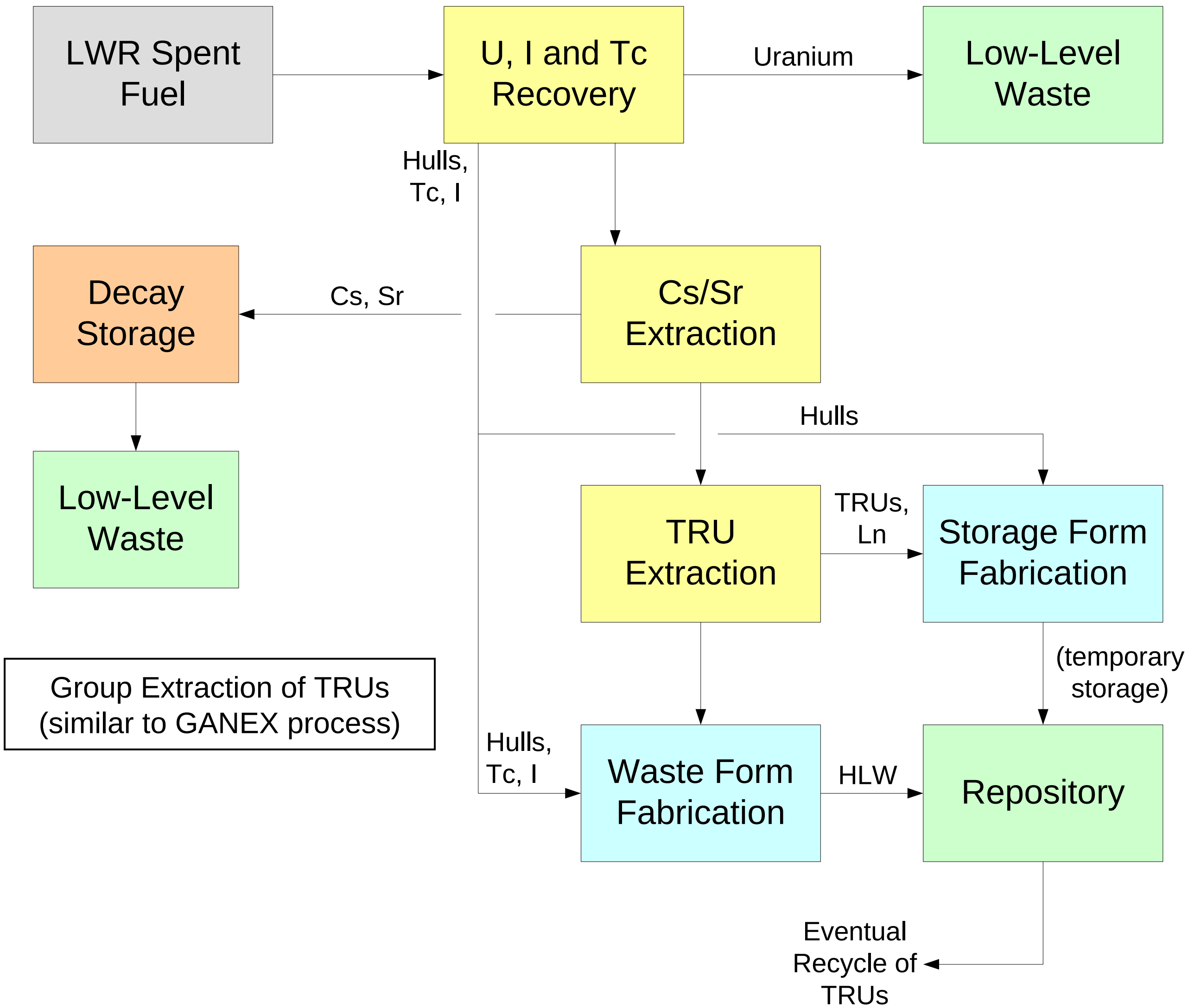
Alternative Processing Methods – Group Separation

- **Both aqueous and pyrochemical processing methods can be operated to recover actinides as a group, without separating plutonium**
- **Such methods effectively preclude the recycle of the recovered actinides in thermal spectrum reactors**
  - Fuel fabrication difficulties, need for special fuel design (e.g., CORAIL)
- **Group separation requires deployment of a large number of fast spectrum reactors**
  - Commercial spent fuel continues to accumulate until then
- **Could be useful in a waste management function by placing transuranics in a compact storable/disposable form**

Group Separation of TRU Could improve Perceived Proliferation Resistance

- **One variant of the UREX+ process now under study is a group extraction of transuranics (UREX+1 process)**
  - Uranium (plus technetium and iodine) extracted first
  - Cs/Sr extraction next
  - Then all TRUs are separated together with the lanthanide (rare earth) fission products, leaving a waste stream containing only the fission products (less Cs, Sr and rare earths)
- **This would accomplish the objective of expanding the Yucca Mountain technical repository capacity**
- **Extracted TRUs could be stored in the same way as that proposed for Am/Cm**
  - Criticality issues would mandate a change in storage geometry, including addition of neutron absorbers
  - Product would be self-protecting for at least 60 years (dose from ^{241}Pu and ^{154}Eu)

UREX+1 Process

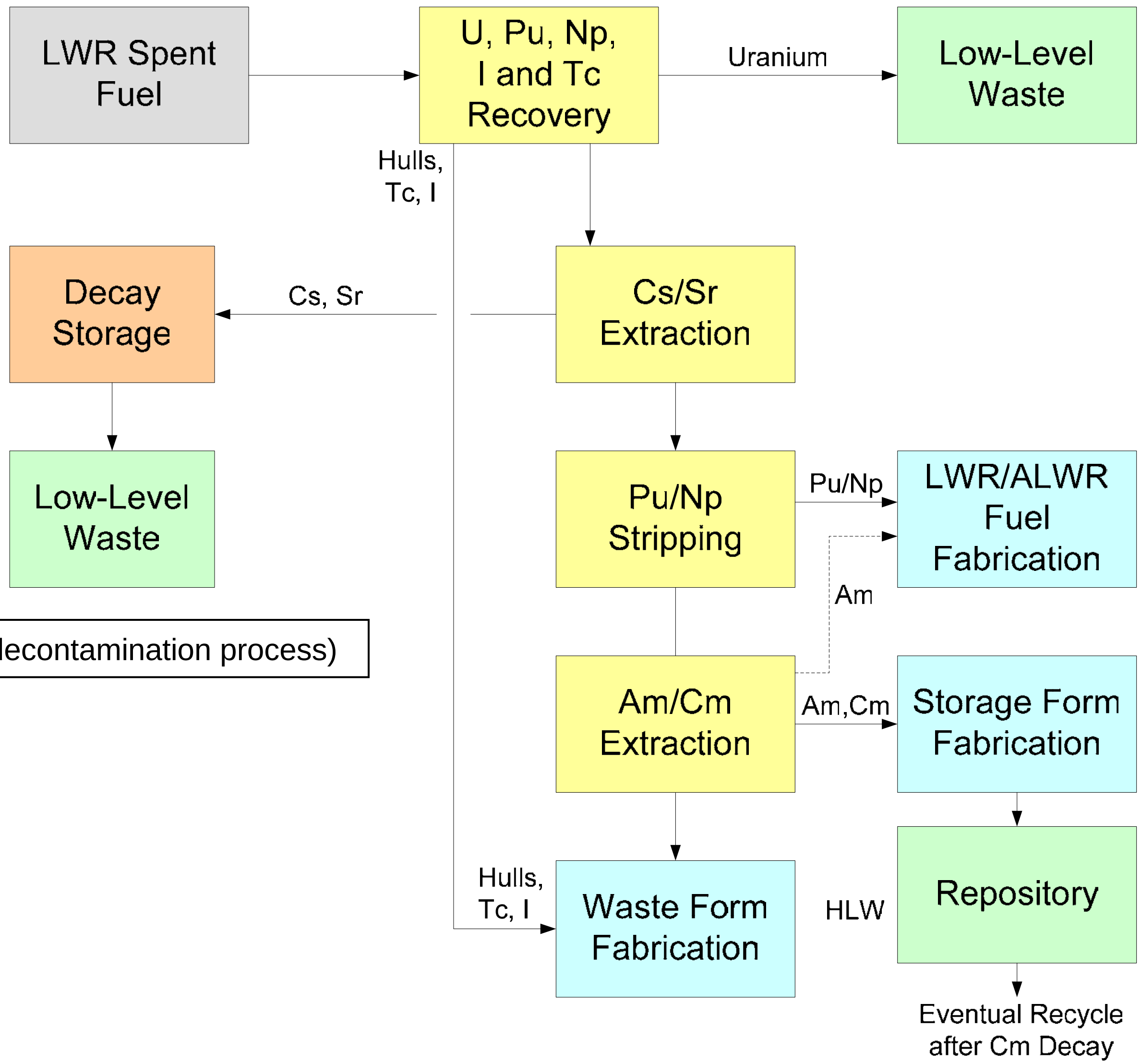


UREX+2 Process is Designed for Thermal Recycle

- **Separations options**
  - Pu+Np recycle, with Am/Cm stored for later transmutation in fast spectrum systems
  - Pu+Np+Am recycle, with Cm stored for decay to Pu+Am

- **Process has been demonstrated with actual LWR spent fuel in FY-2004**
  - High recovery efficiency, >99.7%
  - Excellent product purity
    - Uranium disposal as LLW
    - Cs/Sr disposal as LLW once decay period is complete
    - TRU streams meet ASTM spec for MOX fuel

UREX+2 Process

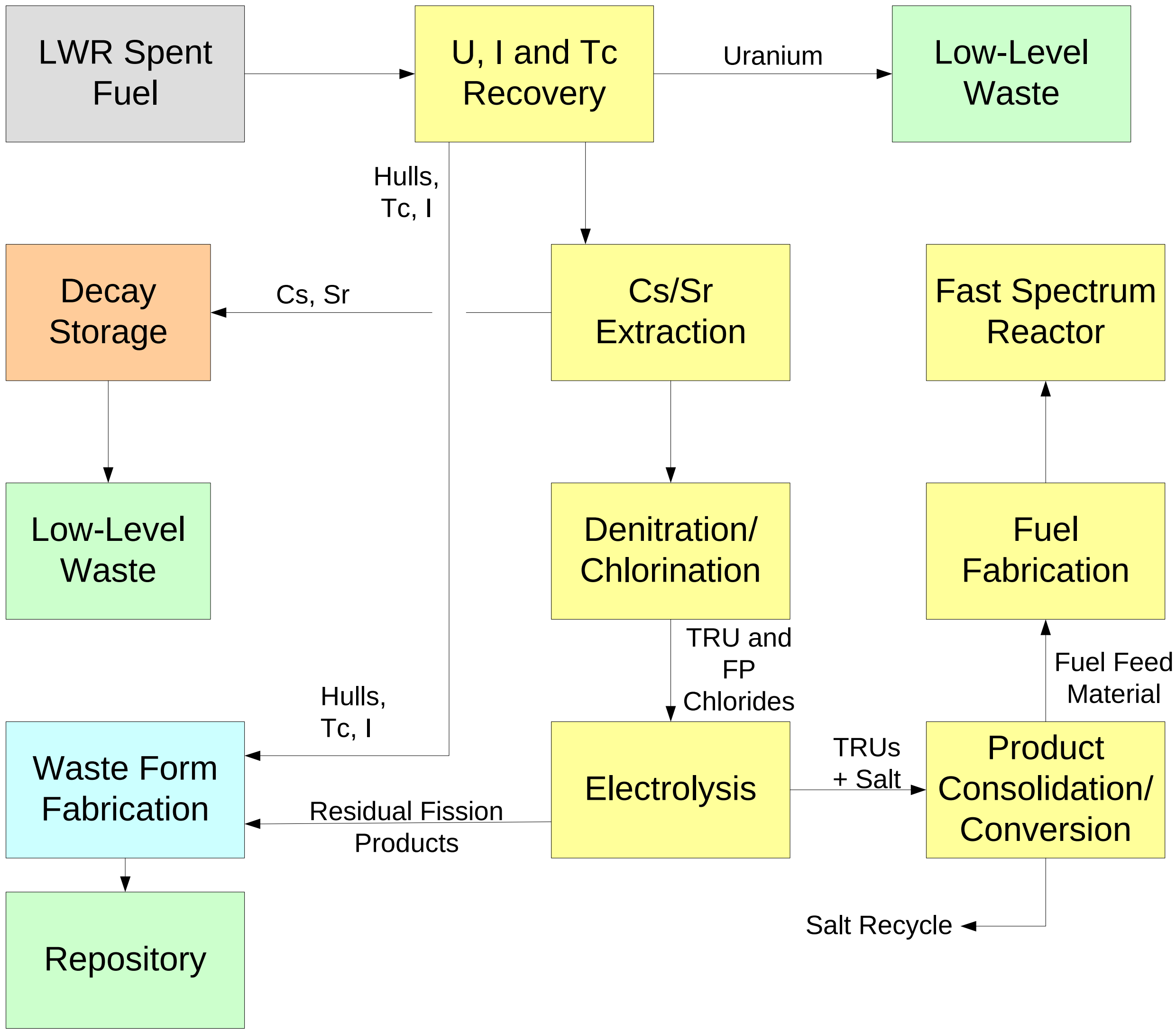


(Co-decontamination process)

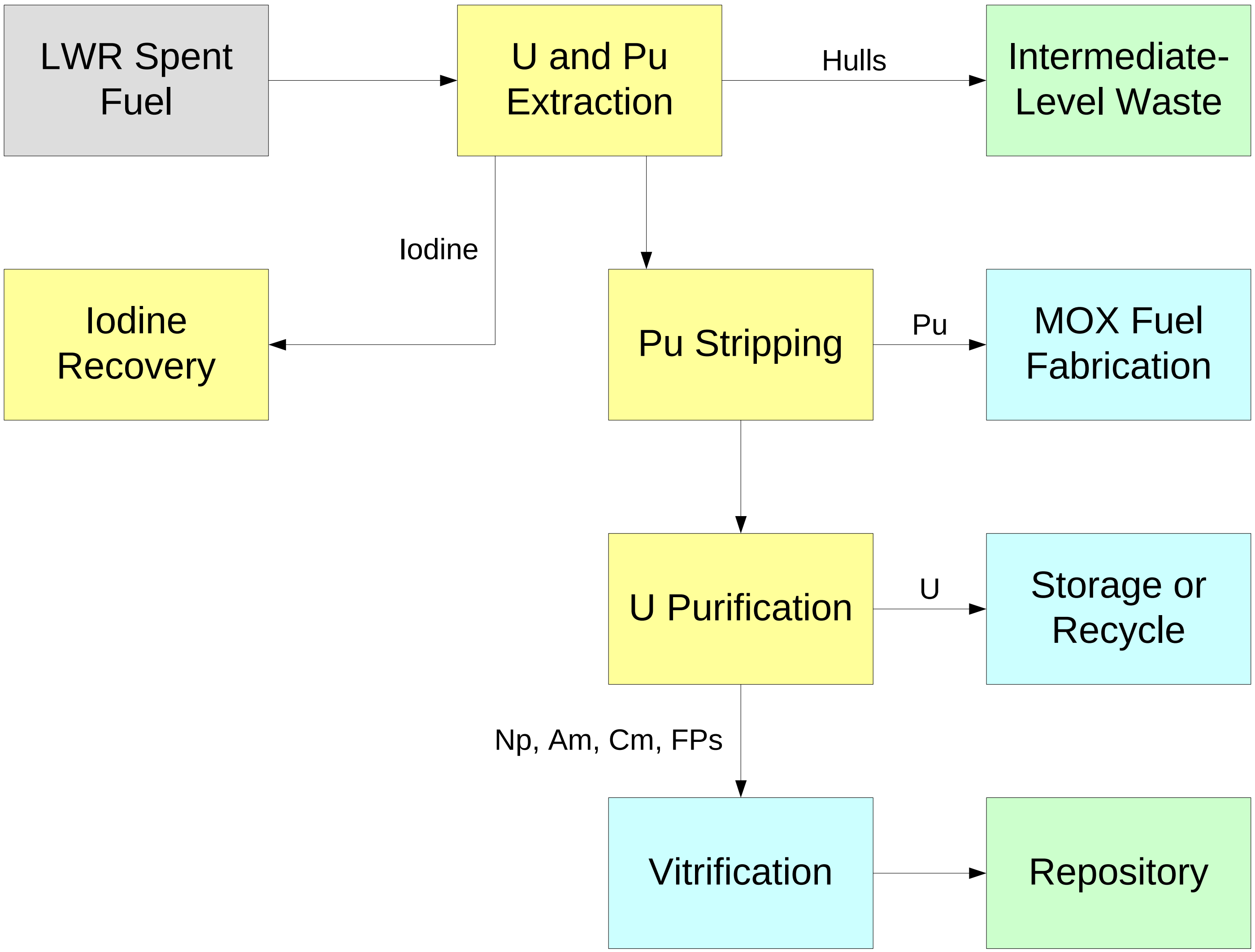
Hybrid Process Under Development for LWR Spent Fuel Treatment

- ♦ Uses advanced aqueous process for removal of uranium, technetium, iodine, cesium and strontium
- ♦ Uses pyrochemical process for recovery of transuranic elements for fast reactor recycle
- ♦ Avoids cumbersome pyro process steps for removal of uranium and for reduction of oxides
- ♦ Potential to capitalize on best features of both processes

Hybrid Process



PUREX Process



Current Research Needs - 1

- **Evaluation of the chemistry of plutonium extraction in the UREX+ processes.**
- **Conceptual development of a hybrid aqueous/non-aqueous process for the treatment of LWR spent fuel that minimizes process complexity and leads to reduced operating costs.**
- **Development of a process for the conversion of technetium strip solution from the UREX+ processes to metallic form for incorporation in a metallic waste form.**
- **Conceptual development of a aqueous solvent extraction separations process incorporating advanced head-end processes such as carbonate dissolution or uranium crystallization.**

Current Research Needs - 2

- **Modeling and design of organic extractants having acceptable radiation stability that can be used in a one-step separation of:**
  - Americium and curium from lanthanide fission products with a decontamination factor $>10^4$.
  - Americium from curium, after lanthanide removal, with a decontamination factor $>10^4$.
- **Synthesis of stable advanced extractant solvent molecules with high specificity for minor actinides (Np, Am, Cm).**

Current Research Needs - 3

- Development of corrosion-resistant stable materials for use in process vessels and crucibles for containment of (1) molten salts containing actinides and fission products, (2) molten actinide metals and chloride salts, and (3) molten non-actinide metals including zirconium.
- Analysis of the effects of small additions of common anions (Br^-, F^-, PO_4^{3-} , I^-) to molten chloride salts for use in electrochemical recovery of specific transuranic elements.
- Conceptual development of a dry process for the treatment of spent TRISO fuel elements discharged from a high-temperature gas-cooled reactor.

Current Research Needs - 4

- Measurement of the thermal properties of the americium/curium storage form.
- Development of durable waste forms, fabricated at low cost, for the geologic disposal of krypton, iodine and tritium.
- Conceptual development of a storage form for the UREX+1 combined transuranic/lanthanide product stream and evaluation of possible inexpensive container designs for temporary repository storage of this form.
- Assessment of the feasibility of incorporating the fission products barium, yttrium and rubidium in the steam reforming process for the production of the cesium/strontium storage form.

Current Research Needs - 5

- Development of a comprehensive plant operations simulation code, perhaps using the ASPEN® framework, for evaluation of process technology options prior to the pre-conceptual design of the large spent fuel treatment facility; the code must provide for plant design parameter variation studies and produce a complete mass balance evaluation of all process streams for the chosen flowsheet and process technology.
- Development and demonstration of advanced on-line, near real-time analytical instrumentation for use in rapid and precise analysis of process streams, with the intention of providing a state-of-the-art system for the monitoring and control of process operations and the accounting of actinide materials for safeguards purposes.

Instrumentation Needs - 1

- **Improved means for analyzing composition of feed material to a spent fuel treatment facility; supplement to calculation of composition**
- **Precise, rapid on-line chemical analytical instrumentation**
  - Mass spectrometry (ICP-MS)
  - Gamma spectrometry
  - Neutron spectrometry (induced fission, resonance/fluorescence)
  - Optical spectroscopy
  - Tank liquid volume and mass measurement
- **Methods to deal with potential inhomogeneous distributions in process vessels; i.e., improved sampling methods**

Instrumentation Needs - 2

- **Improved analytic methods with rapid separations of actinide elements, to favor the real-time use of alpha spectrometry methods**
- **Enhanced process models with provision for detecting secondary indications of diversion**
  - Unusual changes in reagent concentrations and product stream compositions
  - Changes in isotopics; altered mother/daughter radionuclide equilibrium
  - Use of hidden tags, introduced remotely in process vessels
- **Intention is to include these technology advances in the design of the future spent fuel treatment facility and to test them in the INL Engineering Scale Demonstration Facility**

Design of Proliferation-Resistant Spent Fuel Treatment Facility

- **Goal is to integrate advanced safeguards technology into the facility design**
- **First step is to specify a complete facility concept**
- **Each point in the process is examined as a possible key measurement point**
  - Potential for containment and surveillance evaluated
  - Safeguard options and technology needs identified
- **Diversion pathways are analyzed in detail**
  - Design refined to account for problem areas
  - Accountancy procedures developed accordingly
- **AFCI-NNSA joint study complete for a pryoprocessing facility**
- **Joint study being conducted for aqueous (UREX+) facility**
- **Phase 2 proposed to develop the necessary safeguard technology and demonstrate in current facilities**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Philip E. MacDonald

Very High Temperature Gas-cooled Reactor (VHTR)

Idaho National Laboratory

Workshop for Universities
Doubletree Hotel, Rockville, MD
June 16-17, 2005

VHTR Objectives

- *Demonstrate a full-scale prototype VHTR that is commercially licensed by the U.S. Nuclear Regulatory Commission*

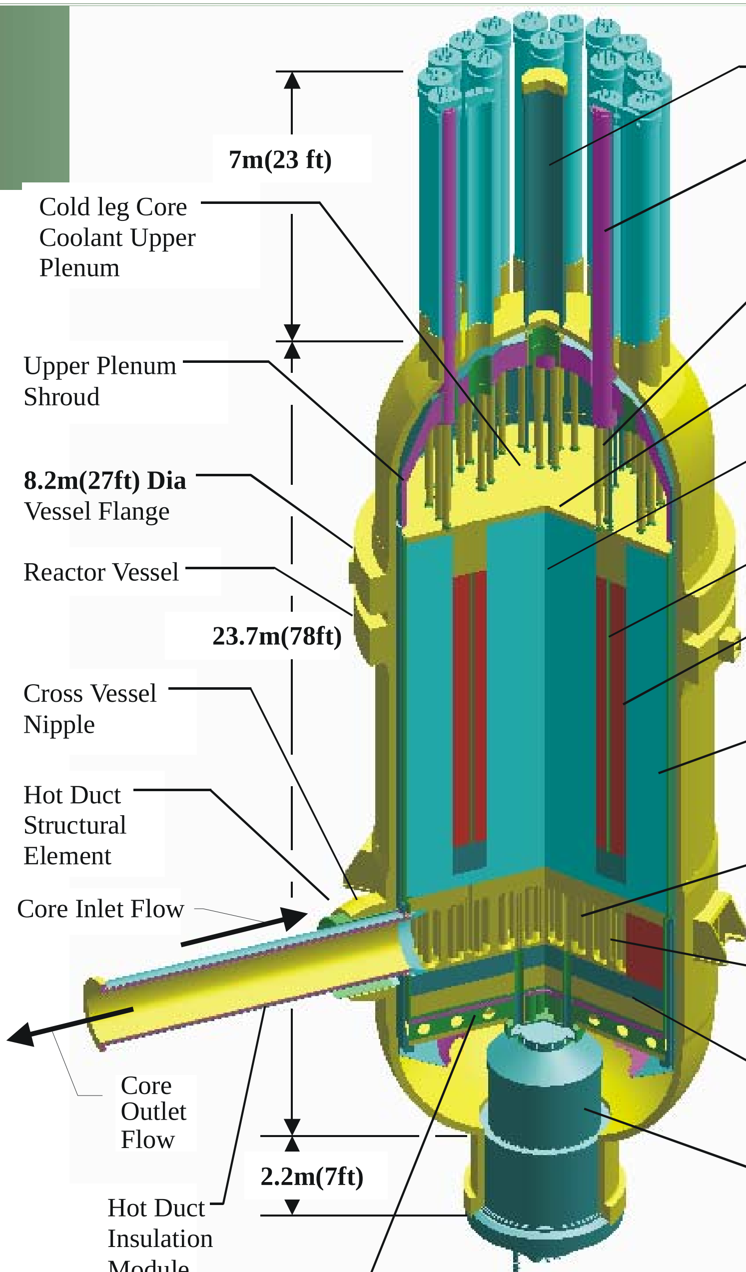
- *Demonstrate safe and economical nuclear production of hydrogen and electricity*

The High Temperature Gas Reactor is the Current Reference Design

- *Utilize inherent characteristics*
  - *Helium coolant - inert, single phase*
  - *Refractory coated fuel - high temp capability, low fission product release*
  - *Graphite moderator - high temp stability, long response times*
- *Simple modular design:*
  - *Small unit rating per module*
  - *Low power density*
  - *Silo installation*
- *Passively safe design:*
  - *Annular core*
  - *Large negative temperature coefficient*
  - *Passive decay heat removal*
  - *No powered reactor safety systems*

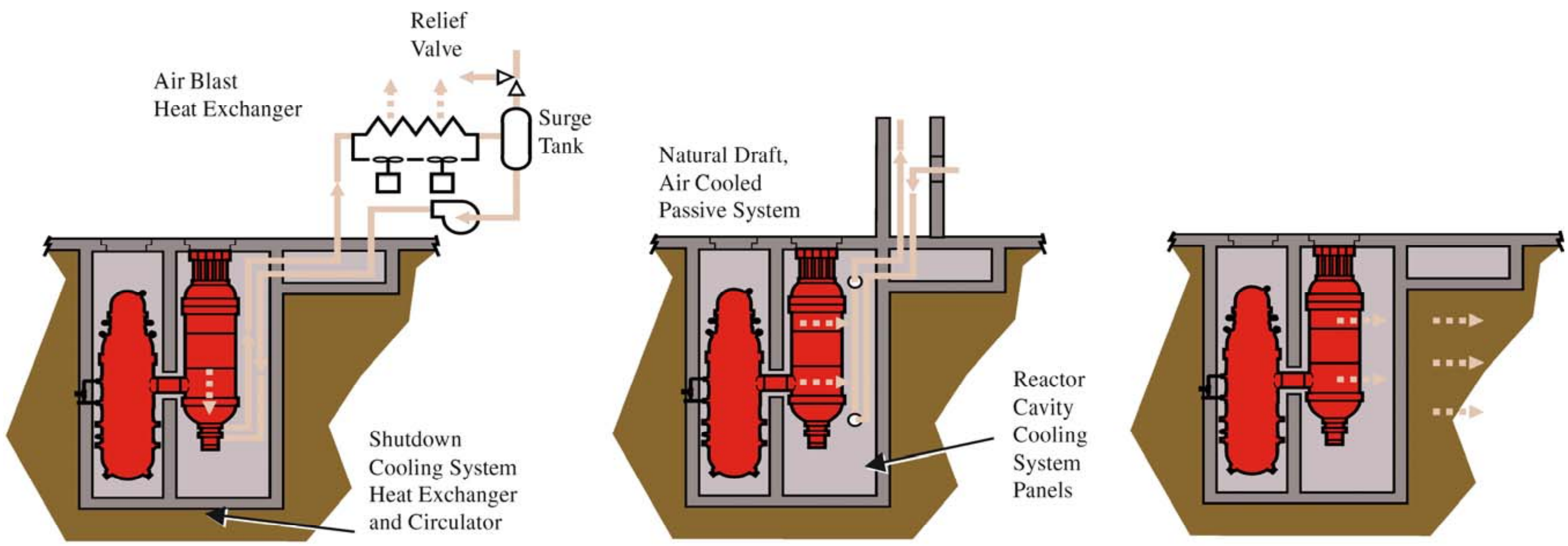
Fort St. Vrain Reactor, 1976-1989

GT-MHR Reactor Layout – An Example of a Possible VHTR



Sketch Courtesy of General Atomics

Multiple Decay Heat Removal Paths



A) Active Shutdown Cooling System

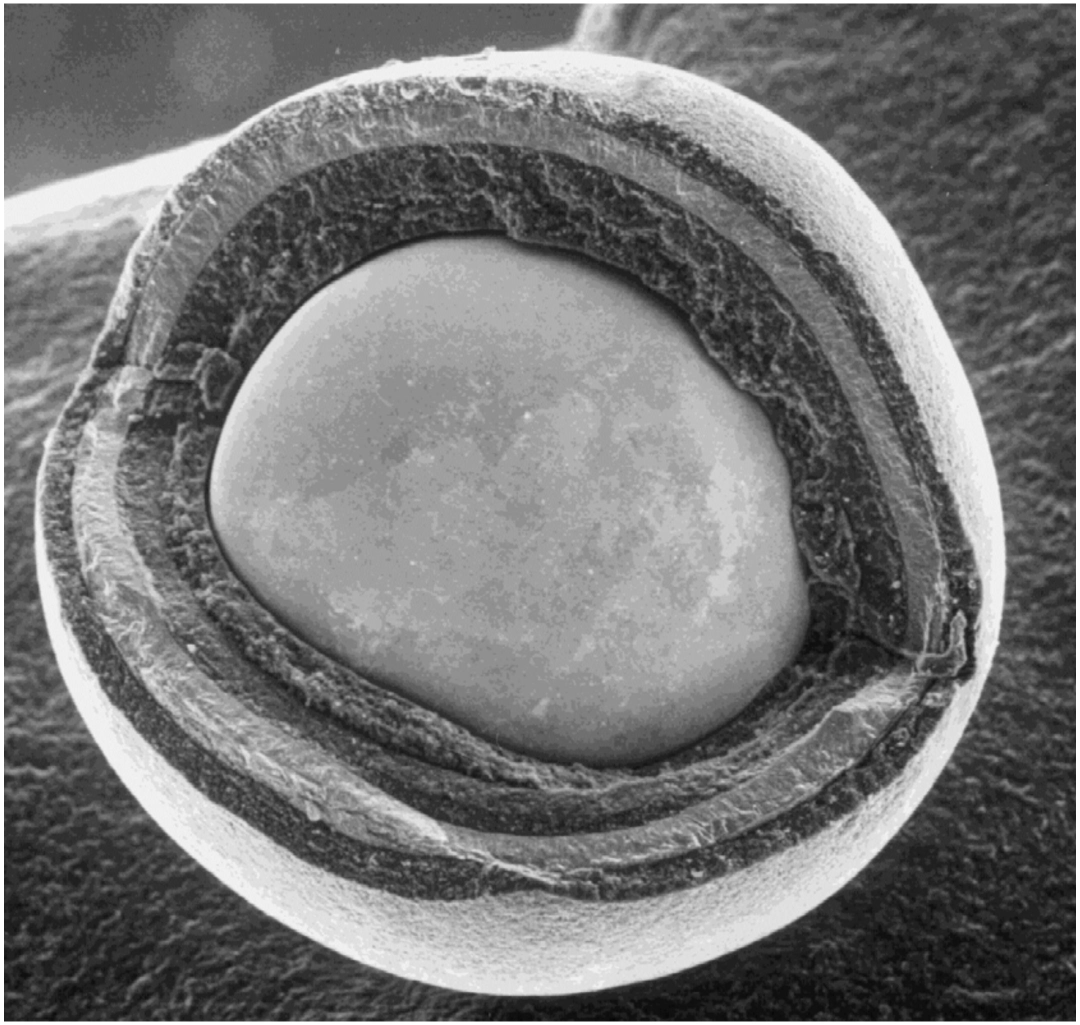
B) Passive Reactor Cavity Cooling System

C) Passive Radiation and Conduction of Afterheat to Silo Containment (Beyond Design Basis Event)

. . . DEFENSE -IN-DEPTH BUTTRESSED BY INHERENT CHARACTERISTICS

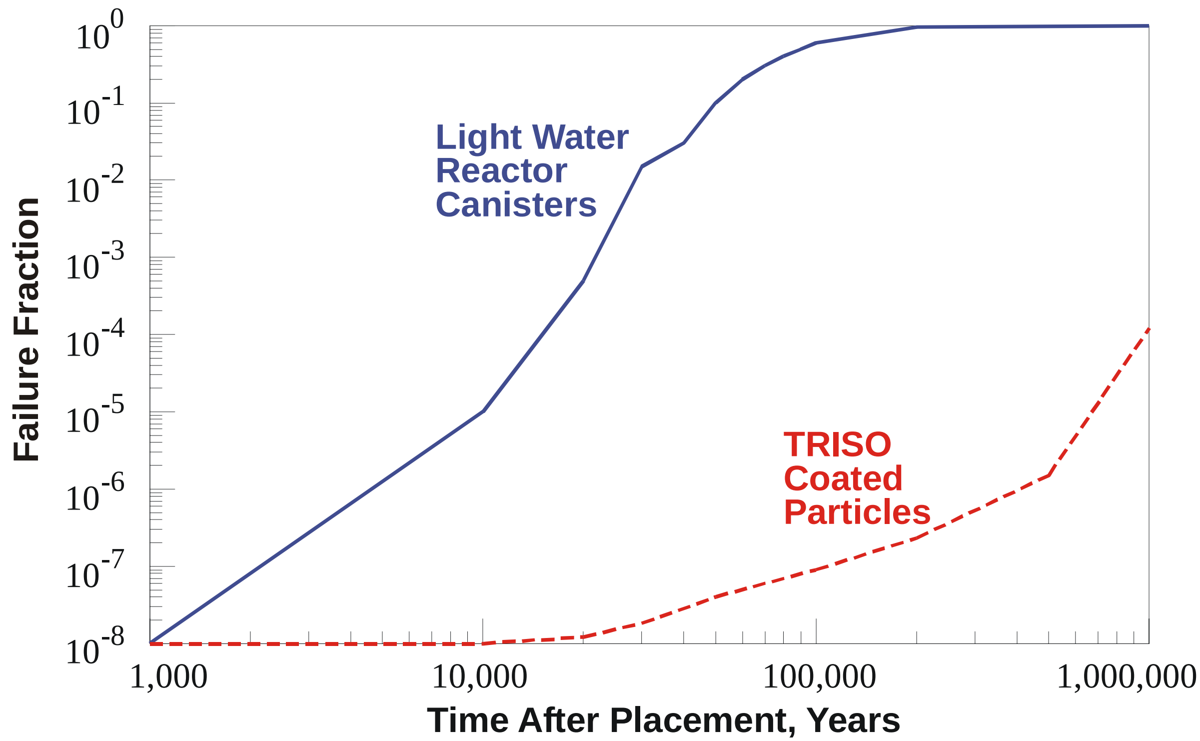
Sketch Courtesy of General Atomics

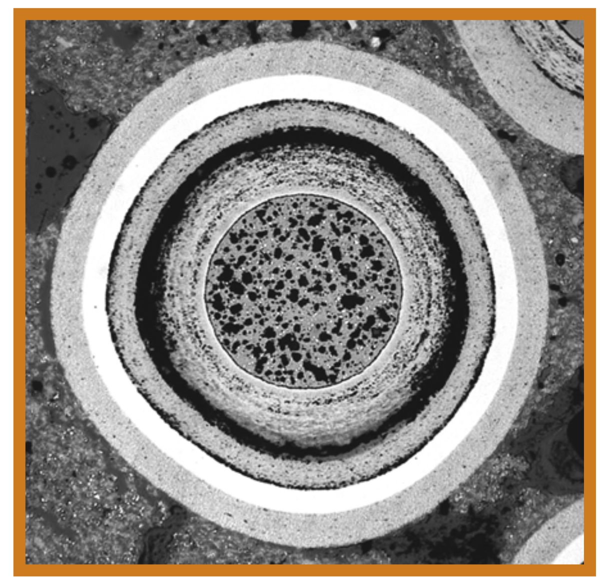
TRISO Fuel Will Retain Fission Products at Temperatures Up to 1,600 °C



The Waste Form Is Suitable For Long Term Underground Storage

- *The VHTR fuel burnup will be about 3 times the burnup of LWR fuel*
- *Therefore, there will be less waste heat and rodiotoxicity per unit energy produced*
- *The TRISO fuel form is extremely resistant to water corrosion and the estimated failure fraction even after 1 million years is very low*



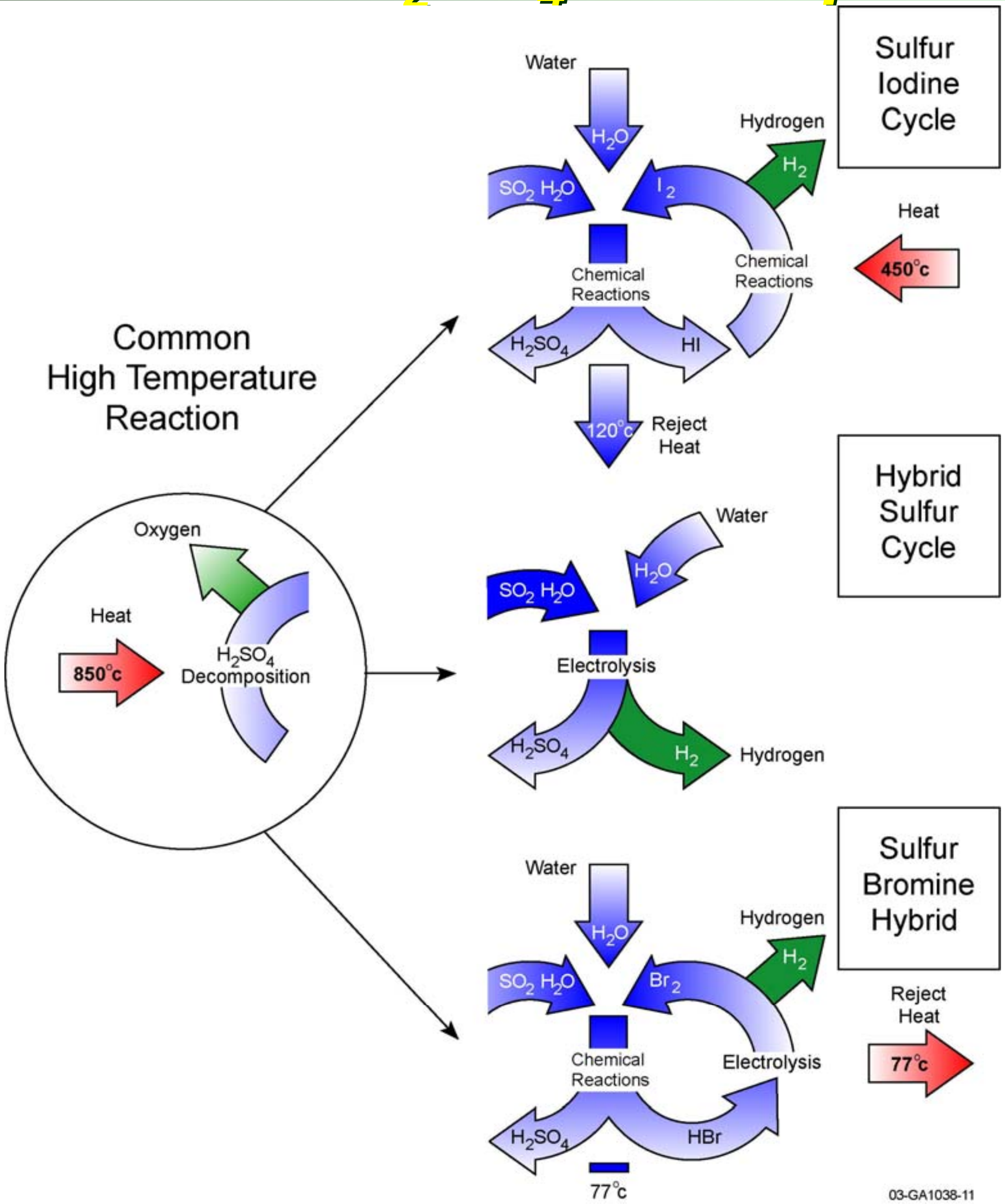


UCO TRISO coated fuel irradiated to ~78% burnup

Hydrogen Production Technologies

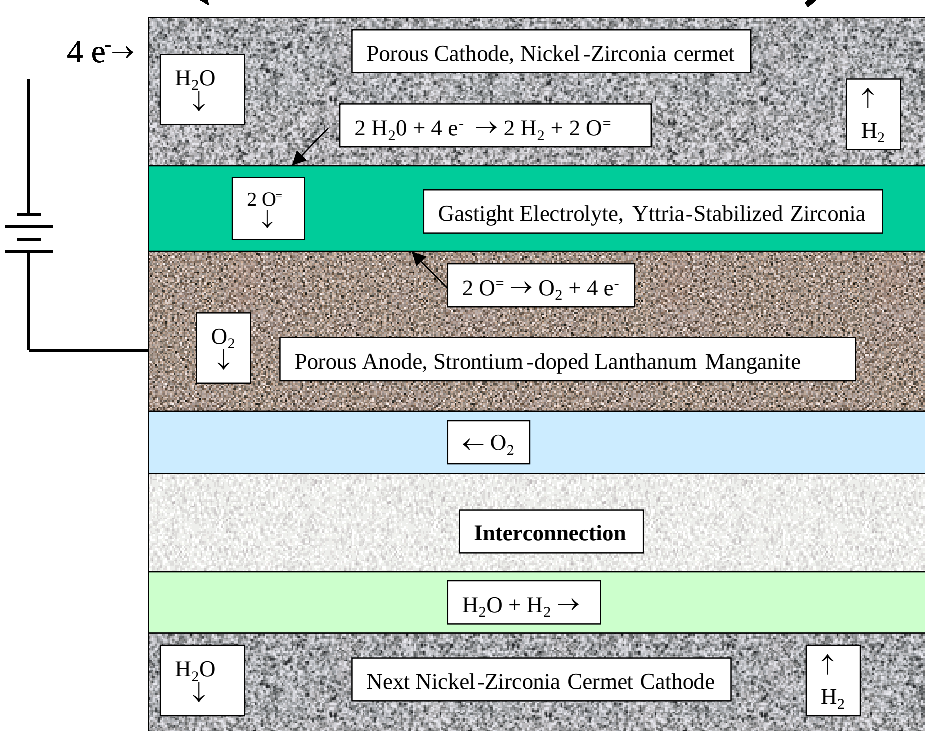
- Two technologies will use the heat from the high-temperature helium coolant to produce hydrogen
- The first technology of interest is the thermo-chemical splitting of water into hydrogen and oxygen
- There are a large number of thermochemical processes that could produce hydrogen, the most promising of which are sulfur-based and include the sulfur-iodine, hybrid sulfur-electrolysis, and sulfur-bromine processes
- The second technology of interest is thermally assisted electrolysis of water
- The efficiency of this process can be improved by heating the water to high-temperature steam before applying electrolysis

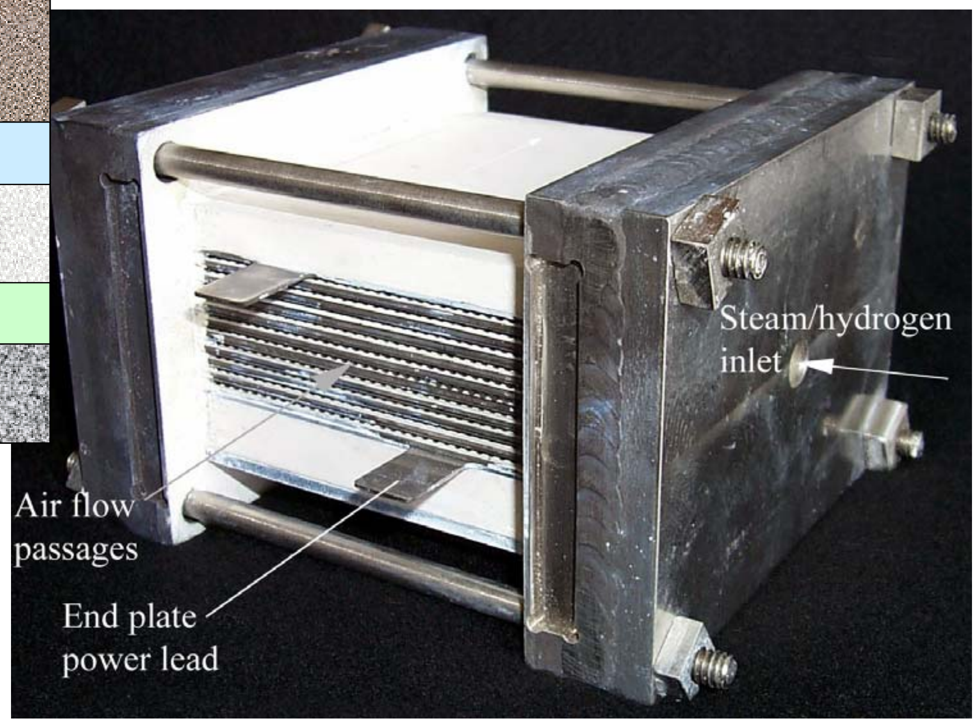
Sulfur Thermo-Chemical Cycles
All have common H_2SO_4 decomposition step



INL High Temperature Electrolysis Process

$90\ {}^{v}/_{o}\ H_2O + 10\ {}^{v}/_{o}\ H_2$ $\quad$ $10\ {}^{v}/_{o}\ H_2O + 90\ {}^{v}/_{o}\ H_2$





This Research Area Includes

- The current Generation IV/VHTR R&D includes work on:
  - Fuels
  - Materials
  - Methods
- There is also hydrogen and BOP electricity production R&D funded separately from the Generation IV program

Materials R&D Task Descriptions

- *Task 1- Nuclear graphite testing and qualification (INL and ORNL)*
- *Task 2 -Development of improved high temperature design methodology (HTDM) (INL and ORNL)*
- *Task 3- Support of ASME code and ASTM standards (INL and ORNL)*
- *Task 4- Environmental testing and thermal aging of high temperature metals (INL and ORNL)*
- *Task 5- Reactor pressure vessel materials irradiation facility (ORNL)*
- *Task 6- Composites R&D (INL, PNNL, and ORNL)*

Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification

- ***Graphite Selection Strategy***
  - *Performed site visits of prospective nuclear graphite suppliers in Europe. Nuclear-grade graphite is not currently domestically produced.*
  - *Negotiated no-cost agreements for graphite samples to be used in the ATR graphite capsule.*
  - *Negotiated cost agreements for billet quantities of nuclear graphite*
  - *The Graphite Selection Strategy Report was issued 4/30/05*

Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification (cont.)

- ***HFIR Rabbit Irradiations of NBG-10 Graphite***
  - *16 of 18 capsules have completed irradiation and are in the hot cell being disassembled*
  - *Last 2 capsules irradiations will be completed at the end of the next HFIR cycle*
  - *Each capsule contains two NBG-10 graphite flexure specimens*
  - *PIE will be completed in July or August 2005*

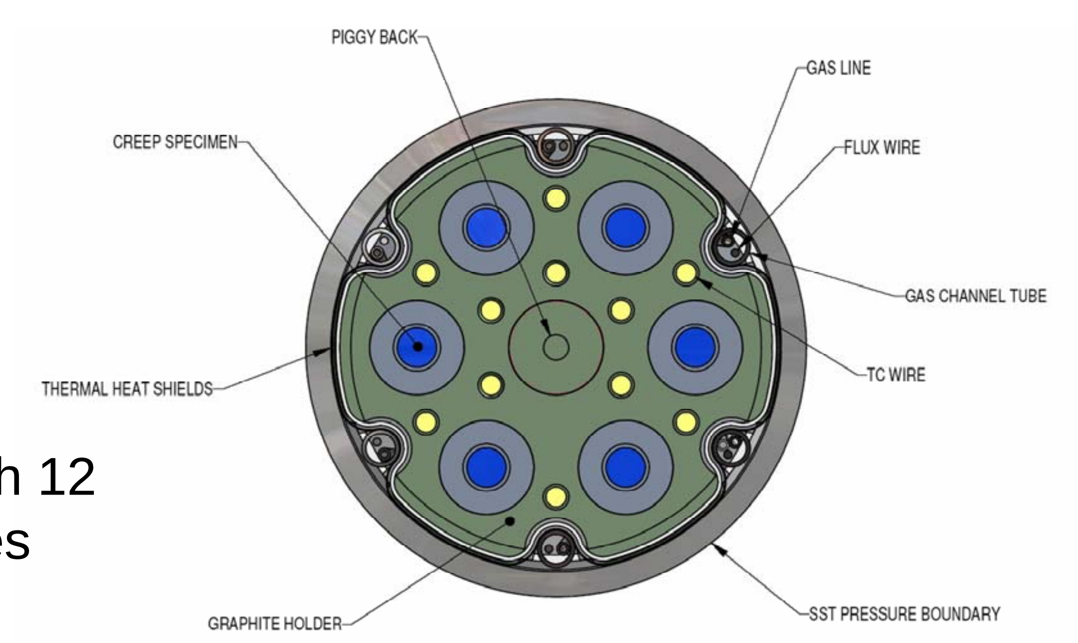
Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification (cont.)

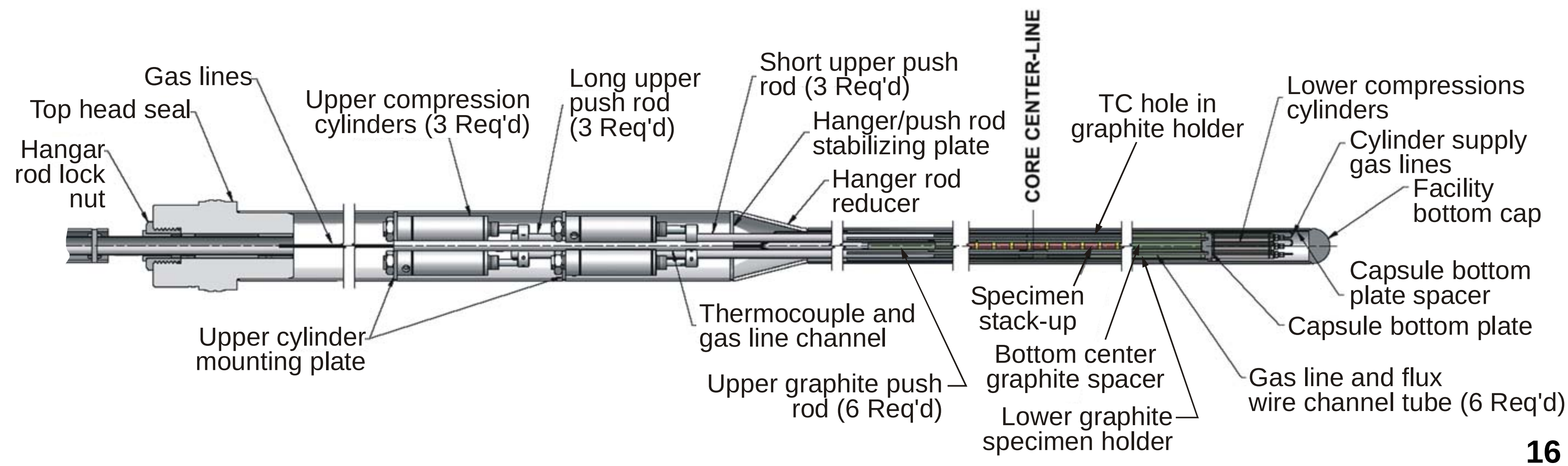
- ***ATR Graphite Compressive Creep Capsule Design***
  - *Preliminary capsule design and initial neutronics analyses are complete. Initial thermal analyses are underway*
  - *Capsule will have 90 creep specimen (stressed & unstressed) pairs and over 300 piggyback specimens*
  - *Five grades of graphite will make up the 90 creep specimens (H-451, IG-110, PCEA, NBG-17,NBG-18 and IG-430)*
  - *The piggyback specimens will be used to acquire irradiated properties for nine new prospective nuclear-grade graphites (HLM, PGX, NGB-25, S-2020, PCIB, PPEA,BAN, NBG-10, HOPG, and A3 Matrix)*

Highlights: VHTR Materials Program

ATR Graphite Compressive Creep Capsule Design



Capsule is fully instrumented with 12 thermocouples, Nb & Ni flux wires and SiC temperature monitors



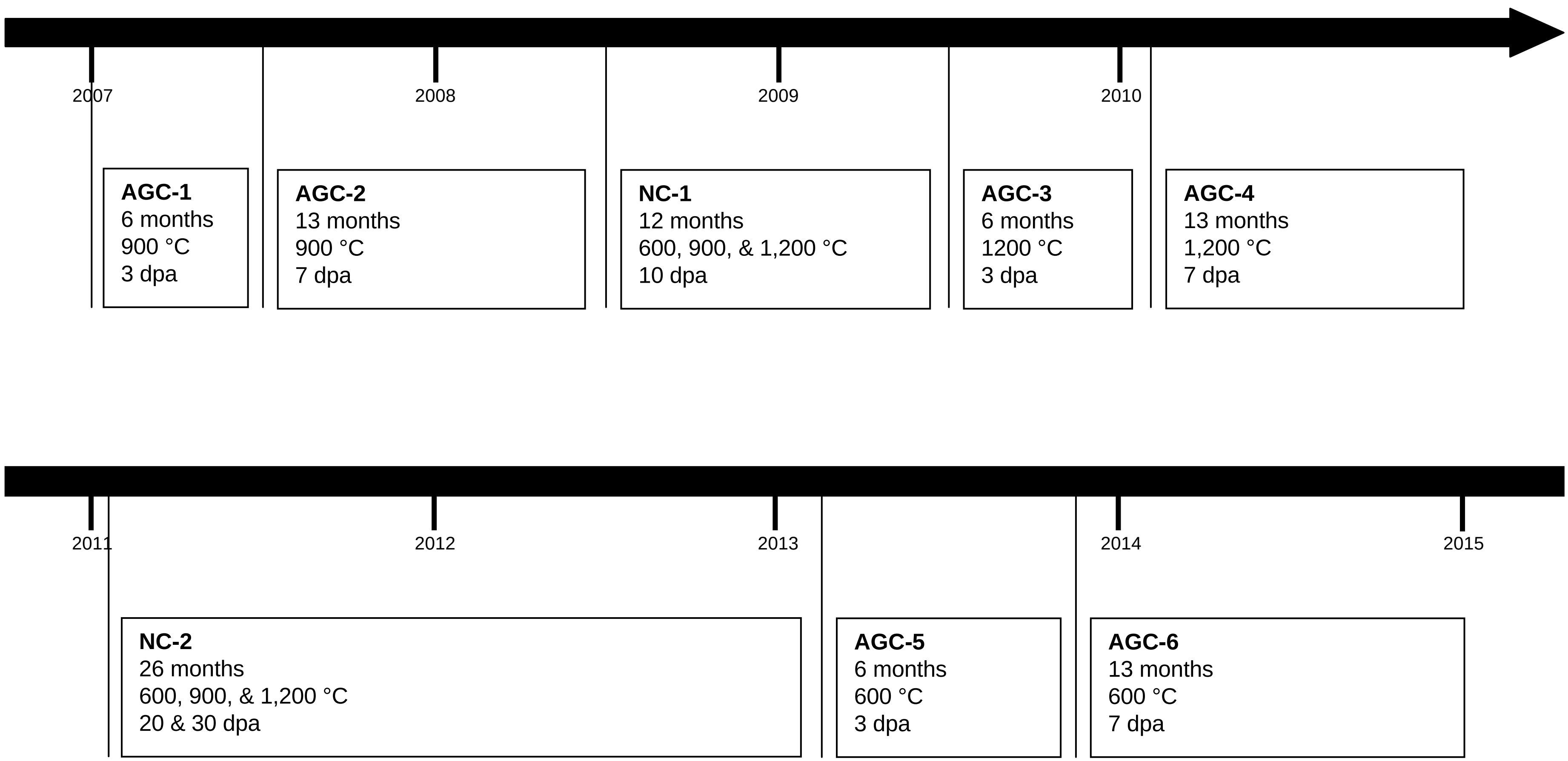
Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification (cont.)

- ***ATR Graphite Compressive Creep Capsule Design (cont.)***
  - *There will be six irradiation capsules operating at 600, 900, and 1,200 °C and at two fluence levels for each temperature*
  - *The capsules will be operated so that the samples will be maintained at constant load and temperature during reactor power level changes*
  - *We plan to issue an "ATR Creep Experimental Plan" by 5/30/05*
  - *We plan to issue an "ATR Creep Experiment Design Report" by 8/30/05*

Highlights: VHTR Materials Program

Projected Timeline for Graphite and Composites Testing in the ATR South Flux Trap



Highlights: VHTR Materials Program

Task 1: Nuclear Graphite testing and qualification (cont.)

- ***Develop ASTM Nuclear Graphite Fracture Toughness Standard***
  - *Ruggedness testing of proposed test method has been completed*
  - *Round-robin testing began in April 2005*
  - *Draft standard and round-robin test results to be discussed at June 2005 ASTM D02.F committee meeting.*
  - *A draft test fracture toughness standard is due 7/31/05*

Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification (cont.)

- ***Nuclear Graphite Modeling***
  - *A new workstation finite element software has been purchased*
  - *An irradiation creep model is under development*
  - *A report entitled "Review and Development Plans for Physically Based Models of the Behavior of Nuclear Graphite under Neutron Irradiation" will be issued by 6/30/05*

Highlights: VHTR Materials Program

Task 1: Nuclear graphite testing and qualification (cont.)

- ***High Temperature Graphite Irradiation Experiment***
  - *HFIR irradiation capsules will operate at 1,200 °C*
  - *Graphite to be included:*
    - *NBG-17, NBG-18 (SGL Carbon)*
    - *PCEA, PCIB (GrafTech International)*
    - *IG-110, IG-430 (Toyo-Tanso)*
  - *Preliminary capsule design and experimental plan will be issued 7/31/05*

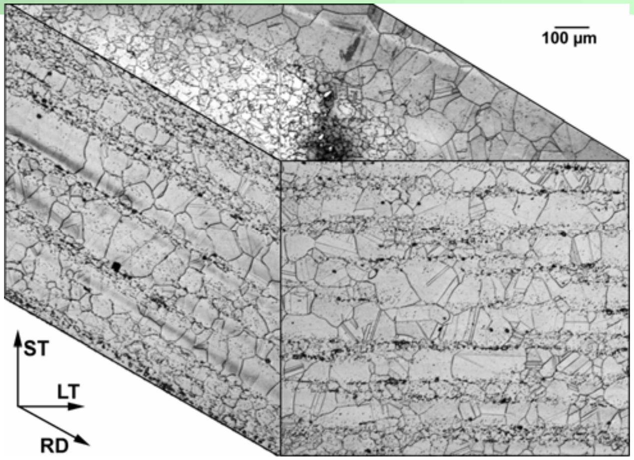
Highlights: VHTR Materials Program

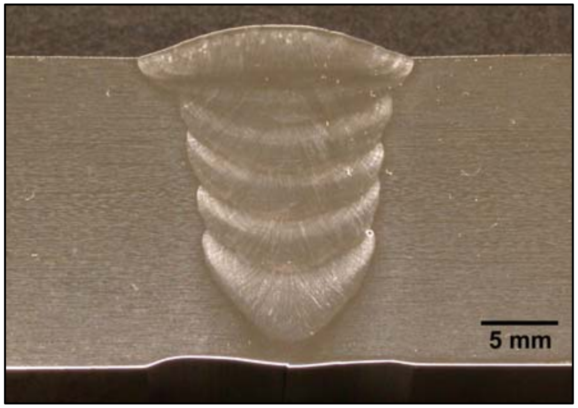
- ***Proposed FY-06 nuclear graphite testing and qualification R&D***
  - *Complete the fabrication of the hardware for AGC-1*
  - *Construct AGC gas control system*
  - *Complete all reviews and requirements for irradiation of AGC-1 capsule in ATR starting in FY-07*
  - *Fabricate and inspect specimens for AGC-1 test*
  - *Procure hardware and write test plan for AGC-2*
  - *Continue the graphite model development*
  - *Integrate the HFIR bend-bar irradiated and unirradiated data into models*
  - *Perform PIE on two high dose rabbits from HFIF*
  - *Write test plan for high temperature capsule irradiation in HFIR*

Highlights: VHTR Materials Program

Task 2: Development of improved high temperature design methodology

- ***Procurement of Alloy 617 and Evaluation of Joints***
  - *Standard chemistry Alloy 617 procured and the baseline microstructure and properties have been measured*
  - *A procurement contract for a creep-fatigue environmental test chamber was placed. Delivery - mid-August.*
  - *Fusion welds and braze joints completed. Basic microstructure and property characterization in progress.*
  - *Initial trial diffusion bonds completed*
  - *Furnace for extended aging at 1,000 °C procured*





Highlights: VHTR Materials Program

Task 2: Development of improved high temperature design methodology (cont.)

- ***Study on Controlled 617 specification***
  - *Existing specs have been identified for analysis.*
  - *Data analyses indicates that CCA 617 does not have improved properties at the temperatures of interest.*
  - *A special version of Alloy 617 with potentially improved properties has been obtained from Special Metals. Specimens have been prepared for verification testing.*
  - *Communications have been underway with vendors for additional specimen procurement*
  - *A letter report on development of a controlled specification for Alloy 617 was issued 5/30/05*

Highlights: VHTR Materials Program

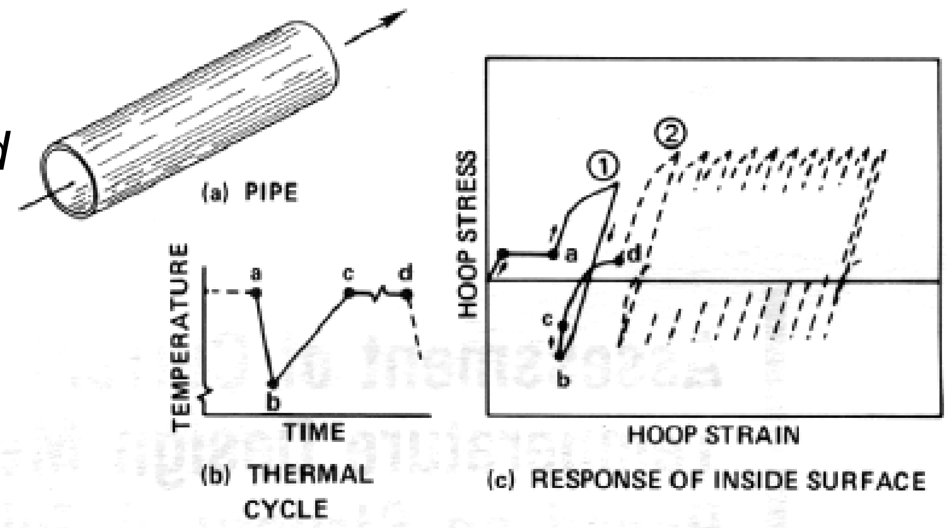
Task 2: Development of improved high temperature design methodology (cont.)

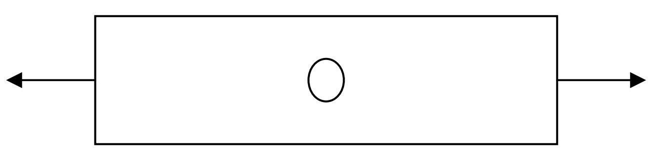
- *Assessing Code adequacy with various operating conditions for various 1-D, 2-D, & 3-D geometries & loadings. Evaluating…*
  - *Load Control Design Rules: resistance to constant loads (creep) with Reference Stress Approach*
  - *Deformation Control Design Rules: resistance to cyclic loads (fatigue, creep-fatigue) with Cyclic Reference Stress Approach*

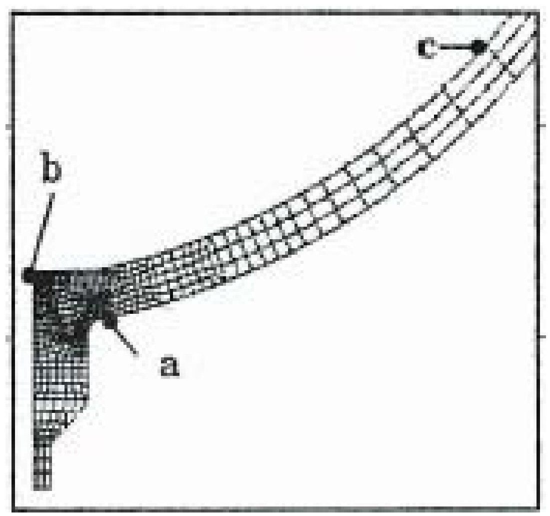
Highlights: VHTR Materials Program

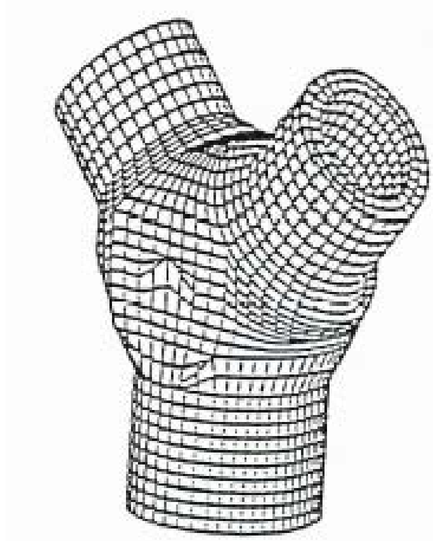
Task 2: Development of improved high temperature design methodology (cont.)

- *Analyses and evaluation of current ASME Code design method (based on simple 1-D cylinder model) relative to other methods*
- *Various test specimens proposed with holes, notches, etc. to capture component behavior*
- *2-D nozzle problem selected to compare current Code simplified design methods*
- *3-D “T” or “Y” problem selected to compare with Code method*









Highlights: VHTR Materials Program

Task 2: Development of improved high temperature design methodology (cont.)

- ***Alloy 617 database development***
  - *Criteria for existing data classification have been developed. Data quality will be determined by verification testing results under the VHTR QA plan.*
  - *Database from previous High Temperature Gas-Cooled Reactors program has been located, data collection from various sources has been underway.*
  - *Existing data compilation format has been developed.*
  - *Assessment of existing data is in progress.*
  - *Compilation of existing data has been initiated.*

Highlights: VHTR Materials Program

Task 2: Development of improved high temperature design methodology (cont.)

- ***Proposed FY-06 R&D***
  - *Procure Alloy 617 based on controlled material specification*
  - *Procure Alloy 230 material and produce and characterize high temperature joints*
  - *Continue creep fatigue testing on standard heat Alloy 617 and start testing of controlled specification specimens with joints in impure He at 800-1000 °C*
  - *Perform scoping tests on controlled material specification Alloy 617 (creep in He and stress-strain evolution) to provide time-dependent input for HTDM constitutive equation development*
  - *Continue to perform simplified methods development as noted on the prior slide*

Highlights: VHTR Materials Program

Task 3: Support of ASME code and ASTM standards development (overview)

- *New divisions within Section III subcommittees are being created to address new reactors designs and concepts*
- *The Section III subcommittee and the design subcommittee will establish the design criteria for any new reactor concepts*
- *All reactor designers will seek NRC approval of their reactor design and view the ASME as the pathway to NRC approval*
- *The Board on Nuclear Codes and Standards (BNCS) has formed a task group on new reactors seeking reactor designer's input*

Highlights: VHTR Materials Program

Task 3: Support of ASME code and ASTM standards development

- ***Graphite Codes and Standards Development***
  - *ASME project team formed, and design requirements for the graphite core support structure's identified.*
  - *ASME draft code currently being prepared.*
  - *ASTM Nuclear Graphite Materials Specification drafted and submitted for sub-committee ballot 2/26/05.*
  - *Standards for fracture toughness, XRD, and air oxidation currently are in preparation.*

Highlights: VHTR Materials Program

Task 3: Support of ASME Codes and ASTM Standards Development (cont.)

- ***Graphite Codes and Standards Development***
  - *Perform round robin testing of the draft oxidation standard*
    - *Draft graphite air oxidation prepared*
    - *Gravimetric oxidation rig constructed at ORNL*
    - *Round-robin will commence in July 2005*
  - *Develop ASTM Nuclear Graphite Fracture Toughness Standard*
    - *Ruggedness testing of proposed test method completed*
    - *Round-robin testing to began in April 2005*
    - *Draft standard and round-robin test results to be discussed at June 2005 ASTM D02.F committee meeting*
    - *A draft fracture toughness standard will be issued by 7/31/05*

Highlights: VHTR Materials Program

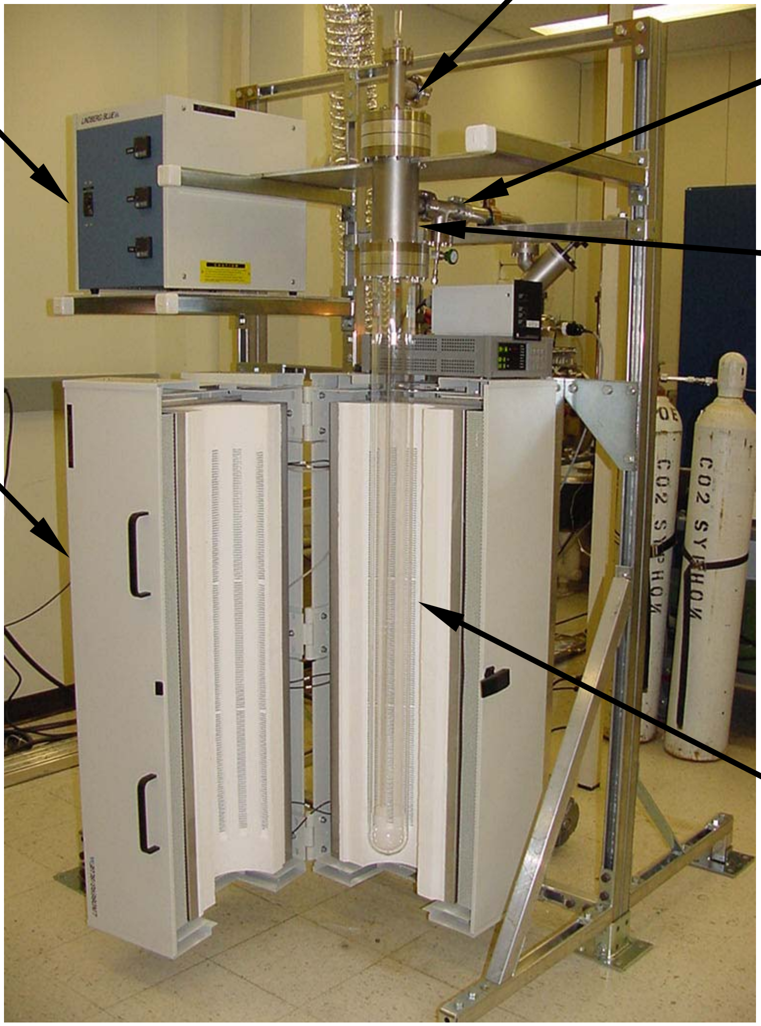
Task 3: Support of ASME code and ASTM standards development (cont.)

- ***SiC/SiC Composite Test Standard Development***
  - *A new working group addressing fabrication issues and test standards for composite ceramic tubes has been formed under ASTM Subcommittee C28.07*
  - *Representatives from all core labs (ORNL, INL, PNL) and other interested research institutions (U of Michigan) will populate this new working group*
  - *The definition of the task group's activity was approved at the ASTM Subcommittee meeting held on Jan. 23, 2005, in Cocoa Beach, Florida*
  - *A recommendation was provided by the ASTM task group regarding the tensile test scheme for SiC/SiC tube samples*

Highlights: VHTR Materials Program

Task 4: Environmental testing and thermal aging of high temperature metals

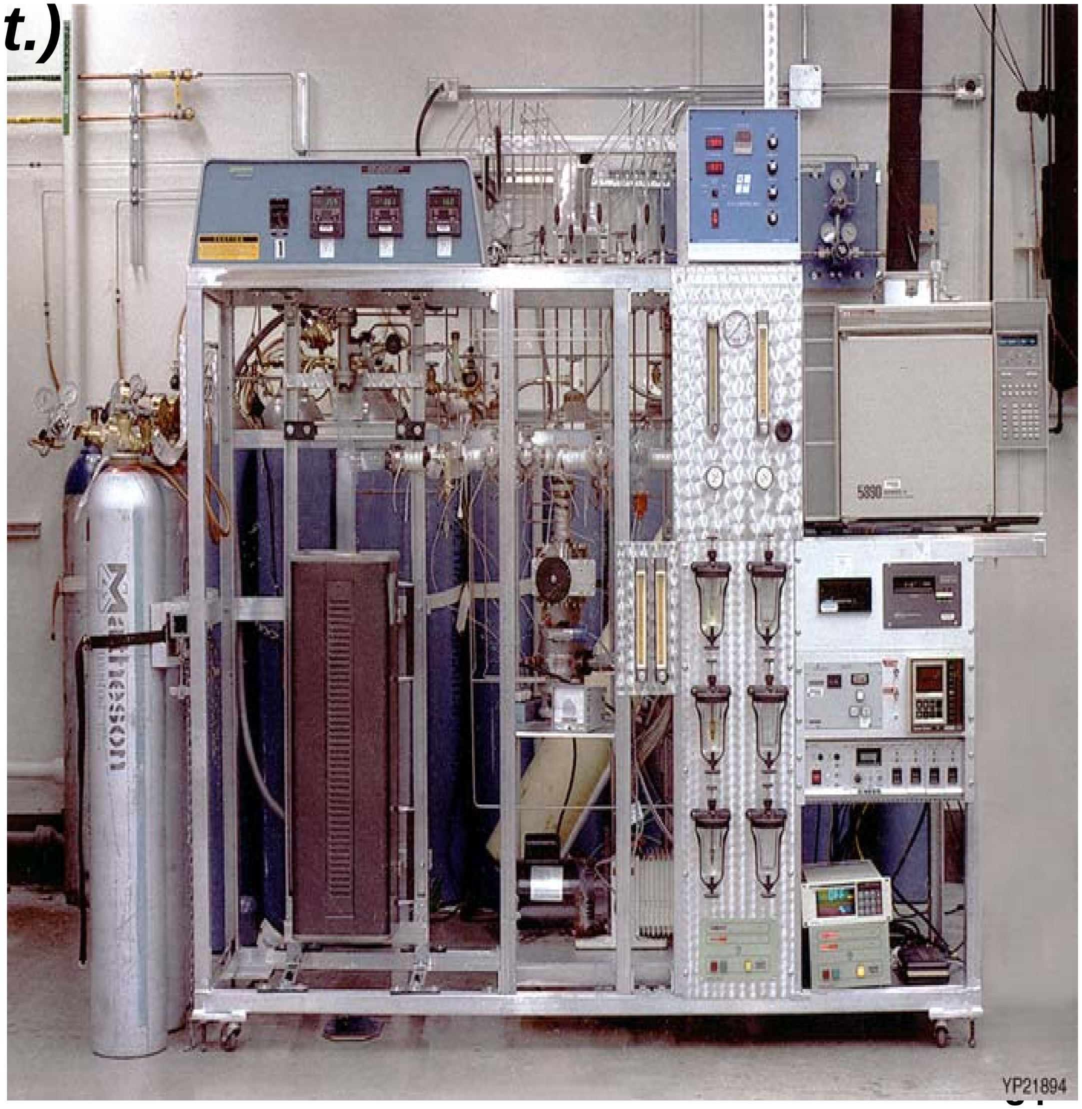
- *INL low velocity He loop:*
  - *Retort, gas handling, and furnace section have been assembled*
  - *Retort and gas handling system have been vacuum tested to $2x10^{-6}$ torr*
  - *Utilities have been upgraded in the lab space*



Highlights: VHTR Materials Program

Task 4: Environmental testing and thermal aging of high temperature metals (cont.)

- ***Determination of testing environment and rejuvenation low flow rate recirculating helium loops at ORNL***
  - *Provide for gas/gas reactions to test dynamic stability of proposed test environments.*
  - *Allow for exposures of selected materials at in controlled helium chemistry at high temperatures*



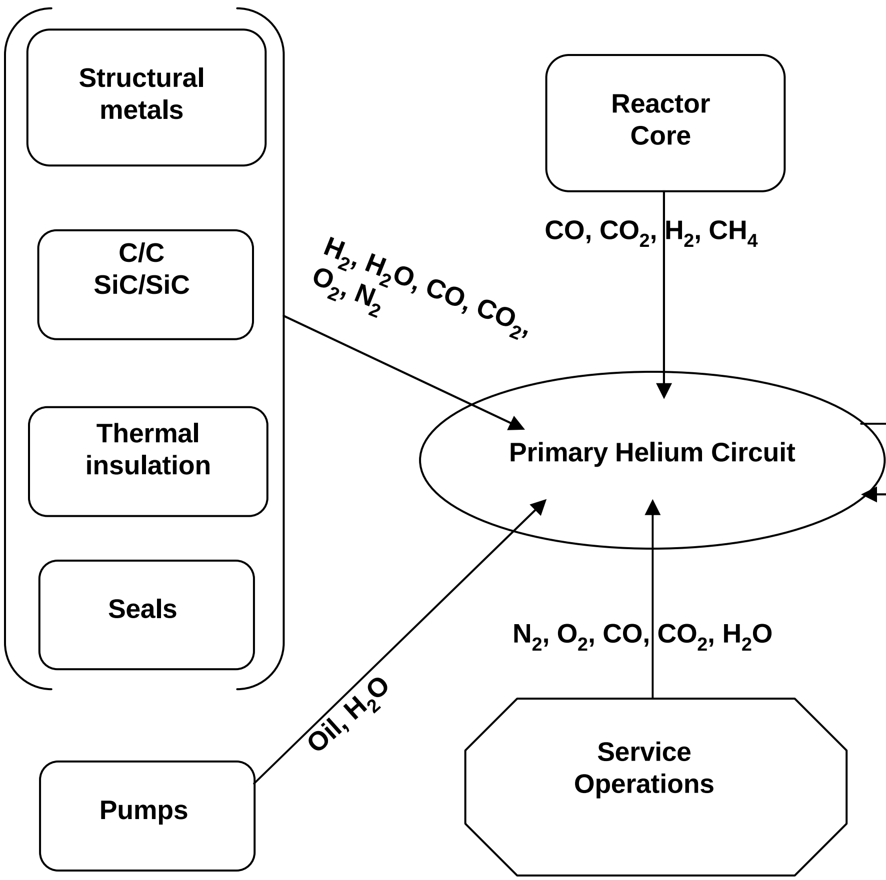
Highlights: VHTR Materials Program

Task 4: Environmental testing and thermal aging of high temperature metals (cont.)

- ***Alloy 617 Aging & Environmental Effects***
  - *Specimens for creep, tensile and fatigue testing have been designed*
  - *Acquired the materials for the aging testing*
  - *Strategies for aging testing has been developed*
  - *Review and analyses of existing data are underway for important thermal aging and environmental factors as the prerequisite for developing the testing program*

Highlights: VHTR Materials Program

Task 4: Environmental testing and thermal aging of high temperature metals - determination of testing environment



- *Review of helium gas chemistry used in past helium gas cooled reactor programs is in progress*
  - *Needed to establish a test helium chemistry envelope in the absence of specific reactor design*

Composition helium environments (advanced HTGR) used in past tests

Program	H_2 (μatm)	H_2O (μatm)	CO (μatm)	CO_2 (μatm)	CH_4 (μatm)	N_2 (μatm)	He (atm absolute)
NPH/HHT	500	1.5	40		50	5–10	2
PNP	500	1.5	15		20	<5	2
AGCNR	400	2	40	0.2	20	<20	2

NPH: Nuclear process heat
HHT: High temperature helium turbine systems
PNP: Prototype Nuclear Process Heat
AGCRNR: Advanced Gas Cooled Nuclear Reactor

Highlights: VHTR Materials Program

Task 4: Environmental testing and thermal aging of high temperature metals

- ***Proposed FY-06 Scope for Environmental Testing and Thermal Aging***
  - *Plan a test program for the low velocity He loop and initiate testing*
  - *Perform long term thermal aging and environmental effects testing at 800-1,000 °C on Alloy 617 (controlled specification) and Alloy 230*

Highlights: VHTR Materials Program

Task 5: RPV materials testing and qualification

- *The primary activity on this project is joint DOE and NRC development of a relatively low flux RPV irradiation facility,.*
  - *This facility will replace the NRC-sponsored IAR (Irradiation/Anneal/Reirradiation) facility at the Ford Reactor, University of Michigan, which was shut down in July, 2003.*
  - *Presentations were made to DOE-NE and NRC Research with a resultant decision by both agencies to proceed.*
  - *A Memorandum of Understanding was drafted by DOE with subsequent discussions and revisions made by both agencies, but a final MOU is not yet available.*

Highlights: VHTR Materials Program

Task 5: RPV materials testing and qualification

- *Six different potential research reactors were visited and preliminary proposals evaluated. The following facilities are currently under consideration:*
  - *BR2 at SCK-CEN, Mol, Belgium*
  - *LVR-15 Nuclear Research Institute, Rez, Czech Republic*
  - *MITR at Massachusetts Institute of Technology, Cambridge, MA*
  - *MURR at University of Missouri, Columbia, Missouri*
  - *MNR at McMaster University, Hamilton, Ontario, Canada*
  - *R2 at Studsvik Nuclear AB, Studsvik, Sweden*

Highlights: VHTR Materials Program

Task 6: Composites R&D

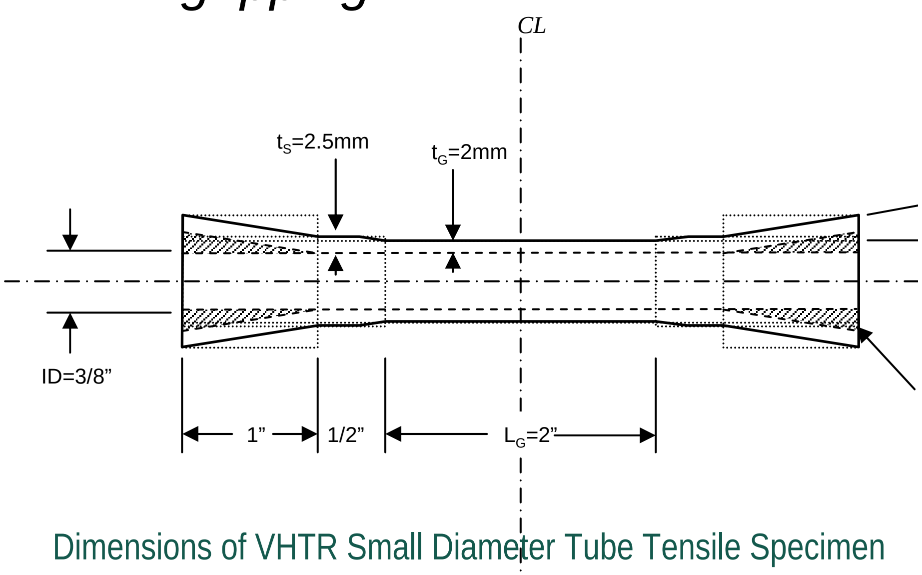
- ***SiC/SiC Composite Component Development***
  - *Fiber reinforcement architecture for the Phase-I tube fabrication was determined to be orthotropic bi-axial braiding, upon the preliminary result of the tube failure mode analysis for the SiC/SiC control rod.*
  - *A detailed plan of the studies on tube size effects on mechanical properties was developed. Tube /flat plate mechanical property correlations was developed.*
  - *Phase-I fabrication of the tubes and flat plates at Hypertherm High Temperature Composites is underway.*

Highlights: VHTR Materials Program

Task 6: Composites R&D

♦ SiC/SiC Component and Test Method Development

- *A tubular specimen with constant inner diameter and integrated ceramic end inserts was designed for evaluation of tensile strength.*
- *A novel fabrication method for the tapered composite tubes, involving the SiC matrix infiltration to SiC fiber preforms braided over the mandrel and end inserts, was developed.*
- *The double-shouldered grip sections were employed in order to force fracture within the gauge section. Two-piece collars will be utilized for gripping.*



Dimensions of VHTR Small Diameter Tube Tensile Specimen

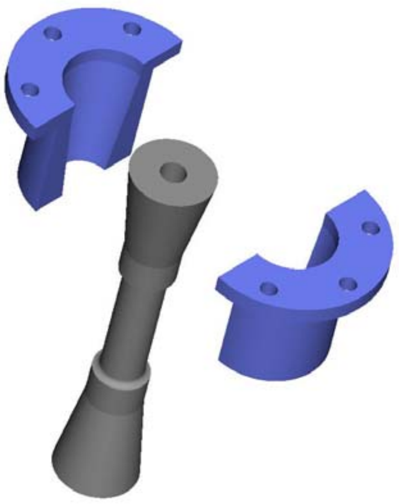


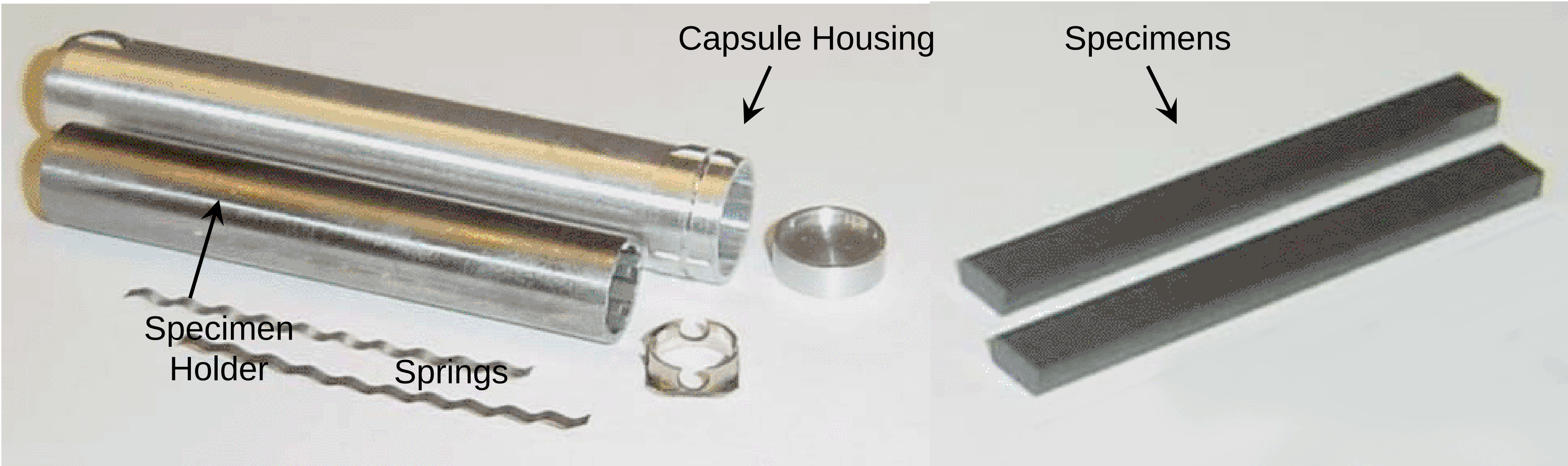
Illustration of Gripping Scheme

Bi-axially Braided Tube

Highlights: VHTR Materials Program

Task 6: Composites R&D

- ***VHTR Composite Bend Bar Capsules for Neutron Irradiation Study***
  - *The VHTR SiC/SiC composite bend bar rabbit capsules have made and inserted in HFIR. Some of the 10 dpa irradiation capsules have completed irradiation.*
  - *Construction of the graphite composite capsules for neutron irradiation in HFIR has been initiated.*



Highlights: VHTR Materials Program

Task 6: Composites R&D

- ***Creep crack growth model and creep testing at the INL***
  - *PNNL is applying their established creep crack growth model to thin, flat geometries and validating the results with experimental studies. Crack growth rates as well as crack growth thresholds will be determined as a function of material condition, temperature, and gas environment.*
  - *A high temperature out-of-pile creep testing program for ceramic composites (up to 1,700 ºC) has been initiated at the INL.*
  - *High temperature creep furnaces, environmental chambers (retorts), and controls have been identified and ordered.*

Highlights: VHTR Materials Program

- ***Proposed FY-06 Composites Work Scope***
  - *Time dependent fracture and crack growth modeling will continue at PNNL*
  - *High temperature out-of-pile strength and creep testing will be performed in support of planned in-pile experiments*
  - *Complete ATR creep experiment plan*
  - *Complete final ATR creep experiment design*
  - *Determine creep capsule specimen design*
  - *Procure SiC/SiC and C/C materials for testing noted above and to fabricate irradiation test samples*
  - *Continue to support the irradiation of SiC/SiC and C/C specimens in the HFIR*
  - *Perform PIE on specimens removed from 10dpa capsules tested in HFIR*

VHTR Materials Program I-NERI Summary

- **US/Japan I-NERI submitted 2/05**
  - Proposed research will demonstrate viability of tubular geometry composite for control rod and guide tube structures. Primary focus will be on C/C composites
  - Lead Japanese organization will be JAERI. Participating US organizations will be INL, ORNL and PNNL
  - Proposed project period is 5 years starting in FY-05
- **US/French I-NERI revised 2/05**
  - Focus on fabrication and evaluation of radiation resistant SiC/SiC composite tubes
  - Participating French organizations will be CEA, University of Bordeaux, and Snecma Propulsion Solide. Participating US organizations will be INL, ORNL and PNNL
  - Technical program was agreed between US and France at the planning meeting held in January, 2005
  - Proposed project period is 3 years starting in FY-05

Materials Qualification and Selection Focus Area

Based on current information obtained from VHTR materials tasks, R&D at universities would be particularly valuable in the following areas:

- *Materials and design methodology for very high temperature nuclear systems. This area could include the integration of required data generation, and a proposed approach to ASME Code Case issues, including design methodology.*
- *Implications of graphite radiation damage on the neutronic, operational, and safety aspects of very high temperature reactors. This area could include a proposed modeling approach and a proposed strategy for dealing with the effects of irradiation creep.*
- *An approach to facilitate the use of composites or other very high temperature materials in very high temperature reactor systems. This area could include data generation in several areas and an approach to modeling composites for applications directed at specific reactor components.*

VHTR Design Methods Development & Validation FY-05 Status Summary

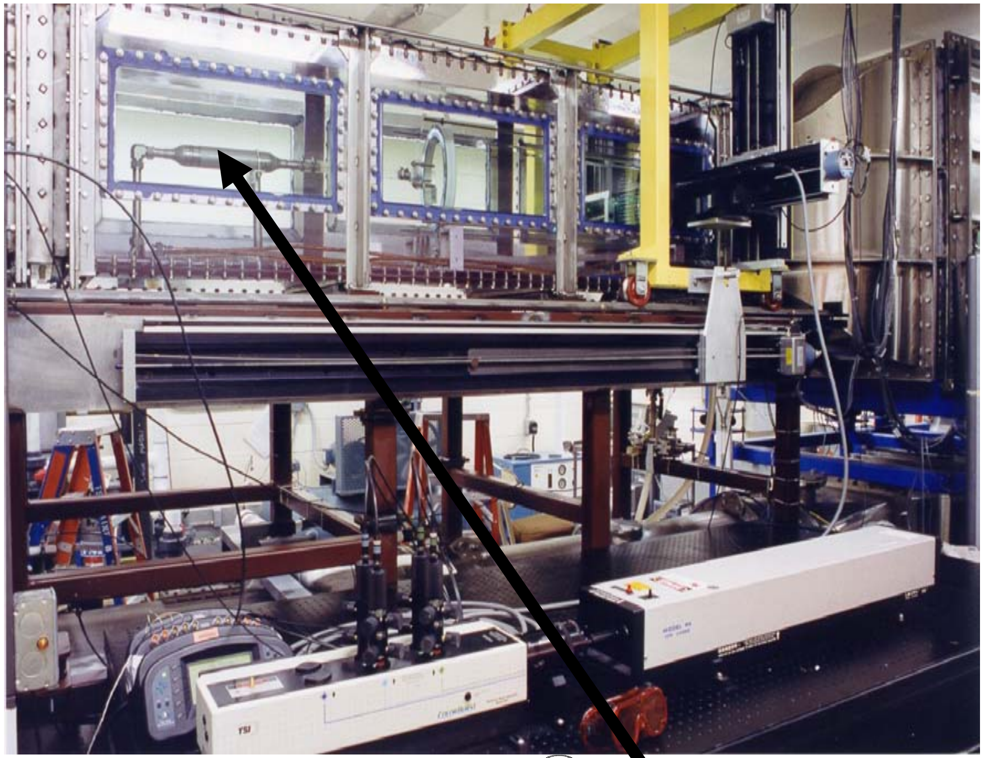
FY-05 Methods development and validation tasks are:

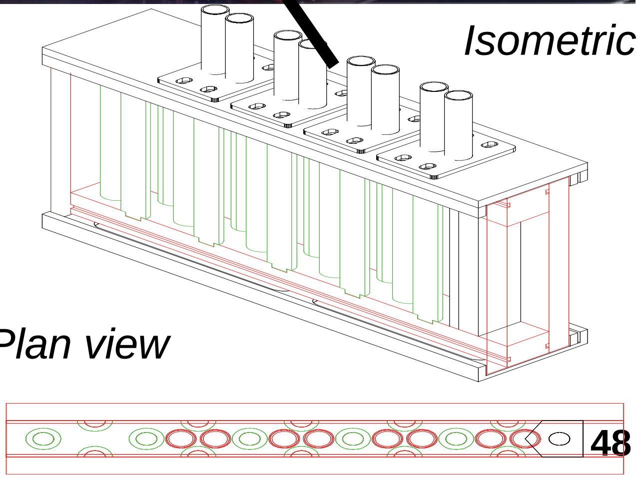
1. *CFD code validation experiments: lower plenum, hot channel, and reactor cavity cooling*
2. *Validation of thermal-hydraulic software, e.g., CFD calculations of exit fluid temperature from hot-channel and lower plenum turbulence; core analysis methods development*
3. *Core physics methods development*
4. *Nuclear data tasks*
5. *Liquid salt-cooled VHTR methods development and design assessments*

VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 1: CFD code validation experiments (INL)

- Scaling and conceptual experiments for "hot channel" and "hot streaking in lower plenum" of prismatic VHTR concept have been defined
- Scaling studies show isothermal experiments adequate for normal operational conditions
- Lower plenum experimental design completed for mixed-index of refraction (MIR) experiment
- Proceeding with fabrication of hardware; first data will be taken prior to end of fiscal year

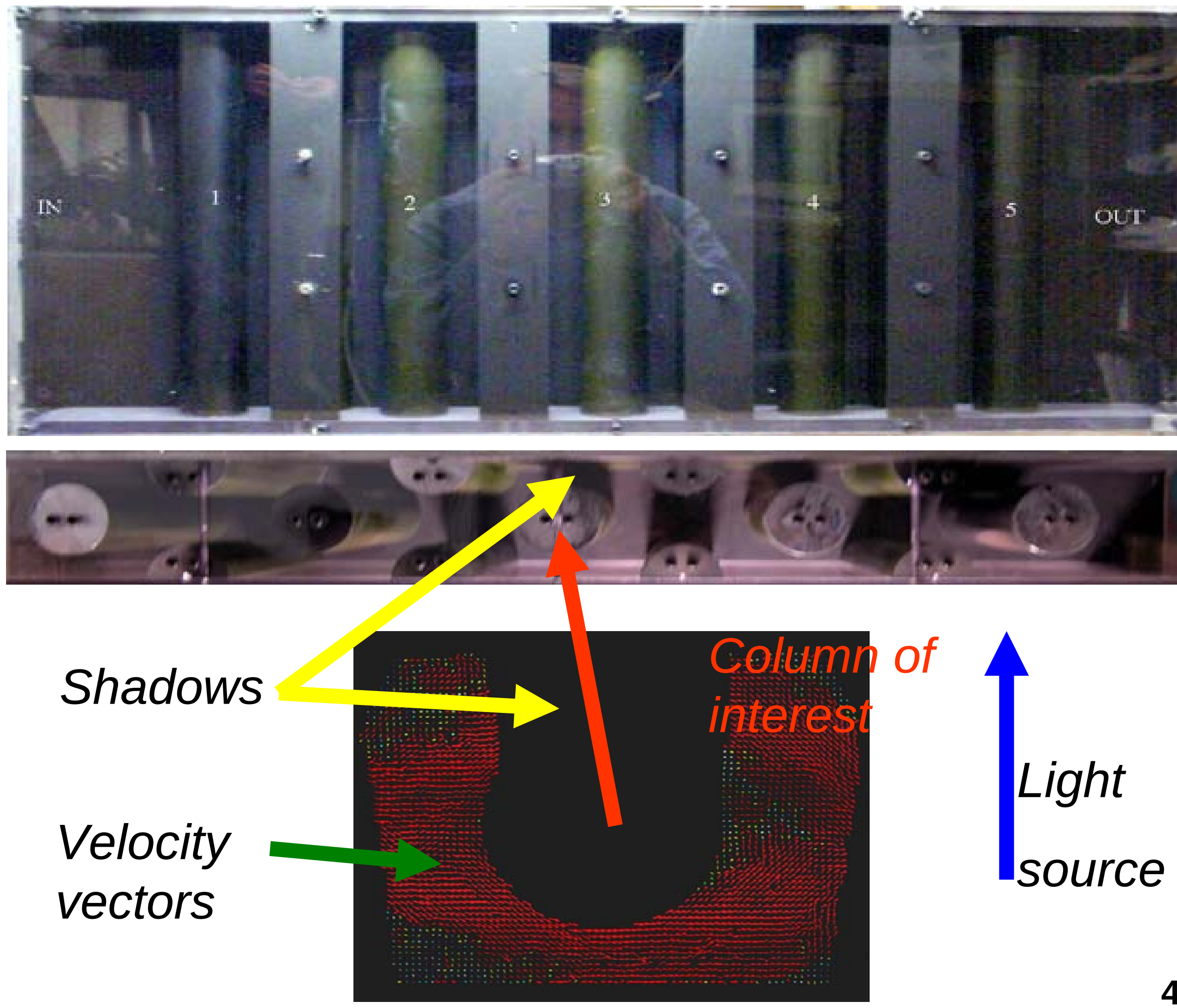




VHTR Design Methods Development & Validation FY-05 Status Summary

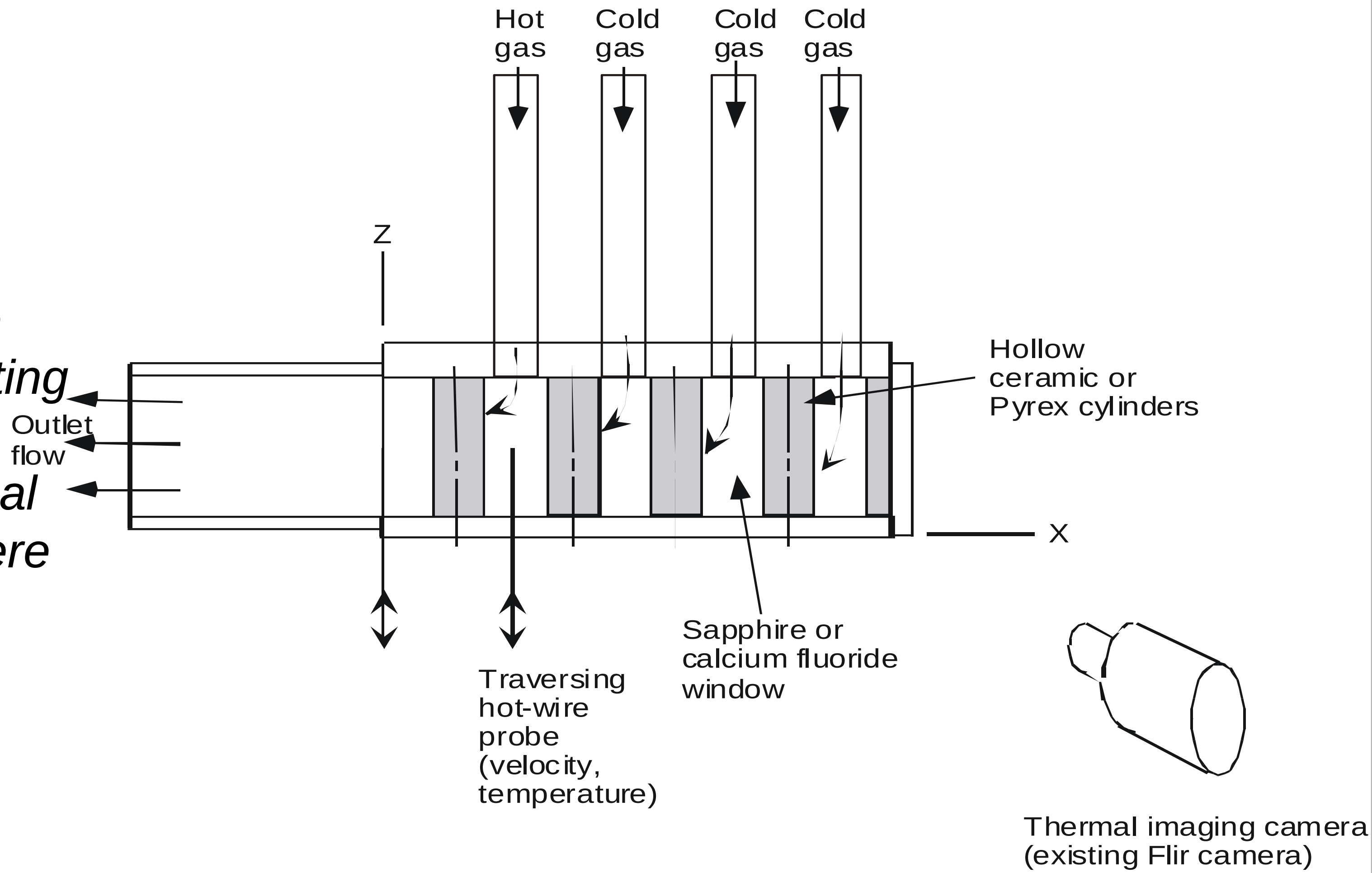
Activity 1: Utah State University Performing Supporting Experiments…

- *Using particle image velocimetry*
- *Flow velocities are being measured.*
- *Seeking minimum Reynolds number required for cross-flows to be in the mixed flow regime*



Activity 1: CFD code validation experiments (INL)

Heated experiment conceptual designs for evaluating the effects of heating on mixing, for non-operational conditions, were completed.

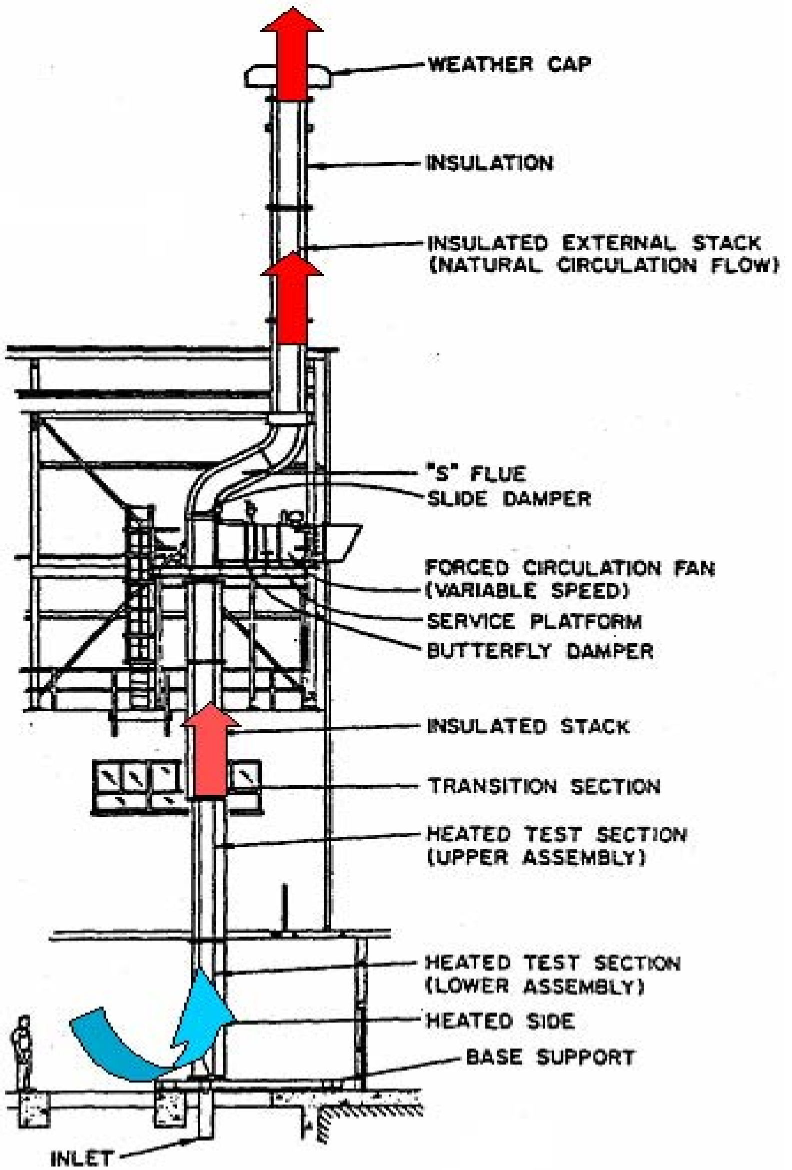


VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 1: CFD code validation experiments (ANL)

♦ ***Evaluate feasibility of using NSTF for generating RCCS data***

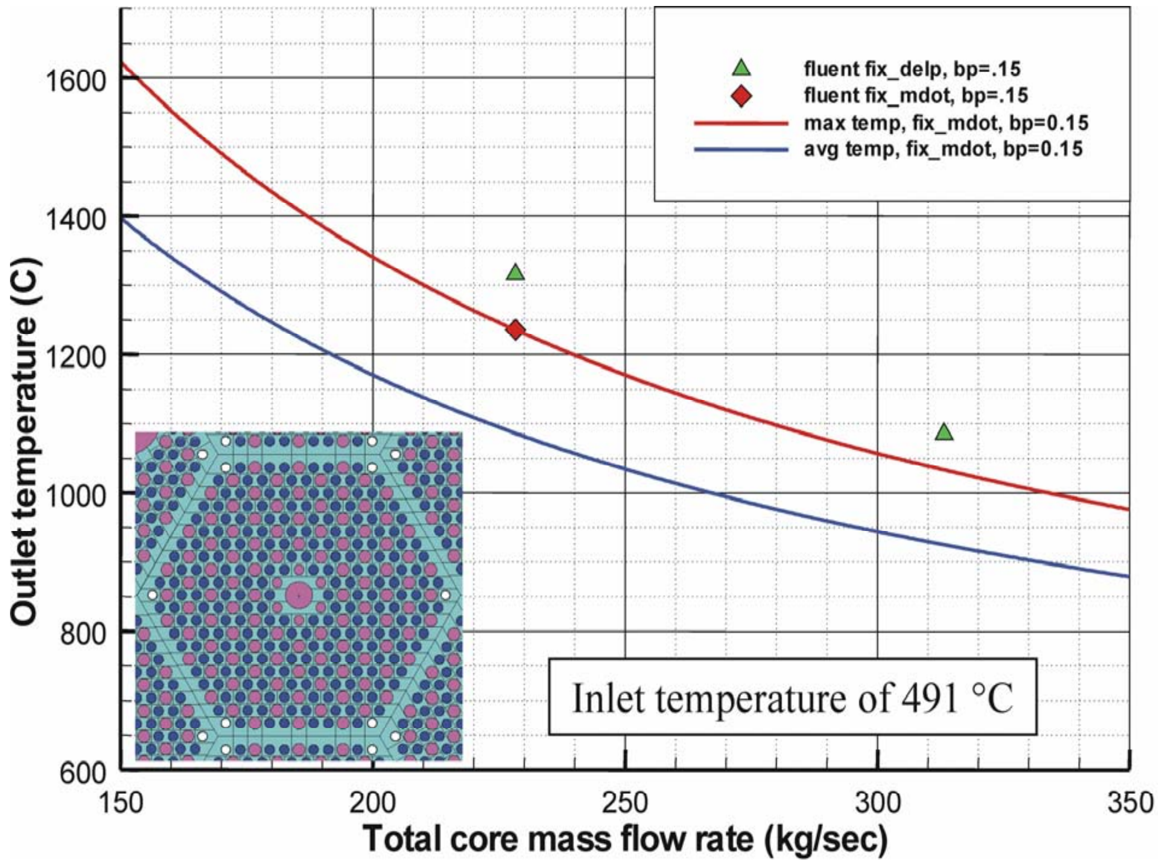
- *The initial facility evaluations indicate that a dozen prototypic cross-section air-cooled RCCS tubes can be installed in the NSTF test section*
- *Water cooled RCCS configurations can also be accommodated*
- *A RELAP5 NSTF model has been constructed to assess facility transient start-up procedures for the conduct of experiments*
- *Needed facility modifications are being costed*



Activity 2: Validation of thermal-hydraulic software

♦ ***Hot channel temperature assessment with CFD codes (INL)***

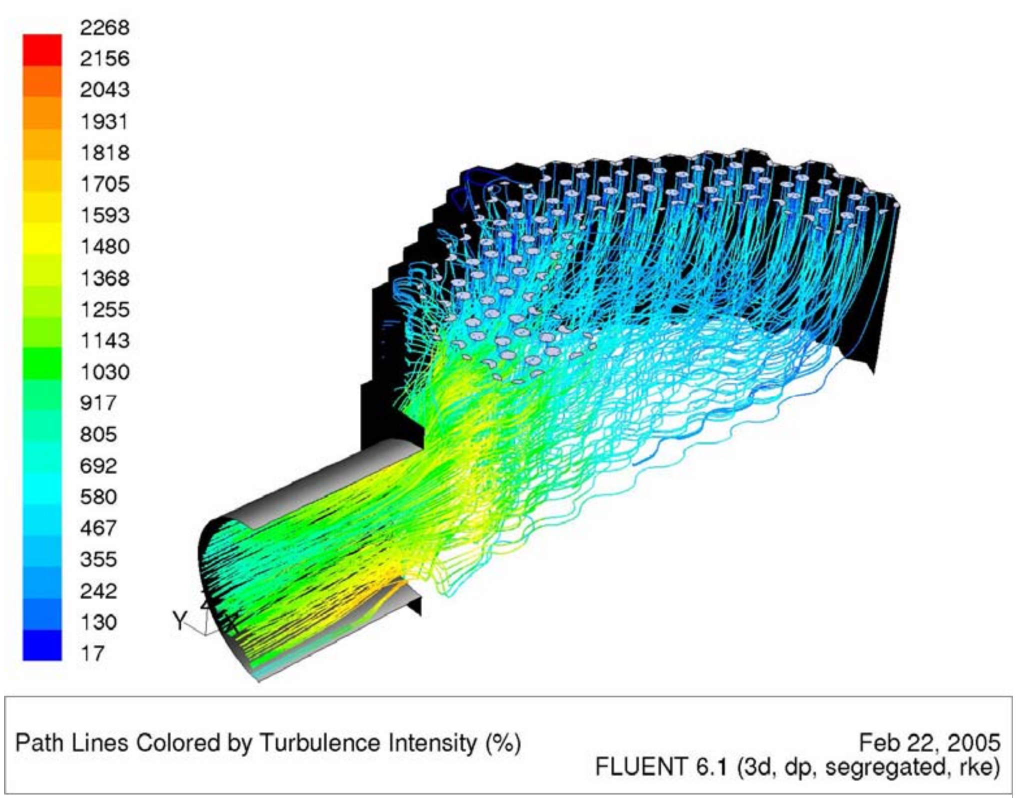
- *Using commercial CFD code (Fluent) and journal-quality accuracy requirements—determined typical "hot channel" exit temperatures*
- *Huge range of potential bypass flow—translates to large range of "hot" channel exit temperatures*



VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 2: Validation of thermal-hydraulic software (INL)

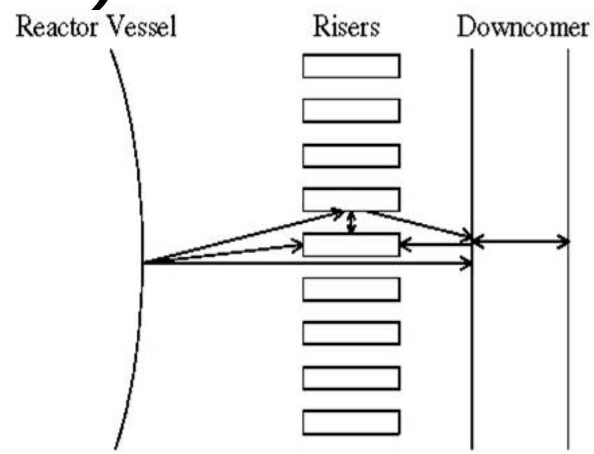
- *Preliminary evaluation of turbulent intensity in prismatic reactor lower plenum completed.*
- *Studies ongoing to perform validation of turbulence in lower plenum and to estimate turbulence in MIR hardware.*
- *RELAP5-3D was assessed for heat transfer & flow conditions prismatic reactor depressurized conduction cooldown—times of 0.5 to 10 hrs*.



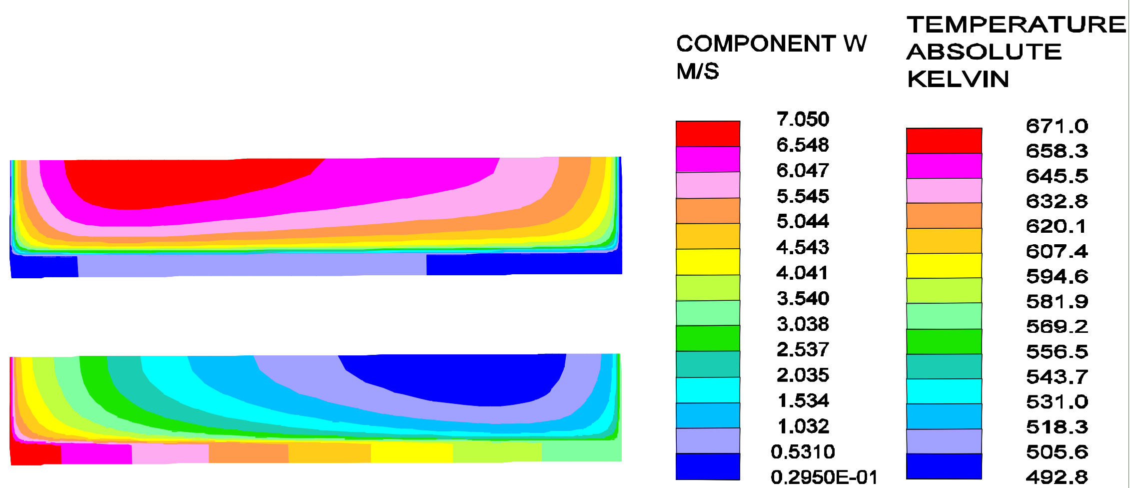
VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 2: Validation of thermal-hydraulic software (INL)

♦ ***Determine scalability of the ANL Natural- convection Shutdown heat removal Test Facility (NSTF)***



- *Identified major scaling parameters & phenomena and constructed a semi-analytical scaling model for air cooled RCCS*
- *Evaluated available RCCS designs, reviewed archival NSTF data, and identified needs for additional sets*
- *Constructed CFD models of available RCCS designs and NSTF and performed accident condition analyses which showed strong 3-D effects and heat transfer differences with existing 1-D correlations*



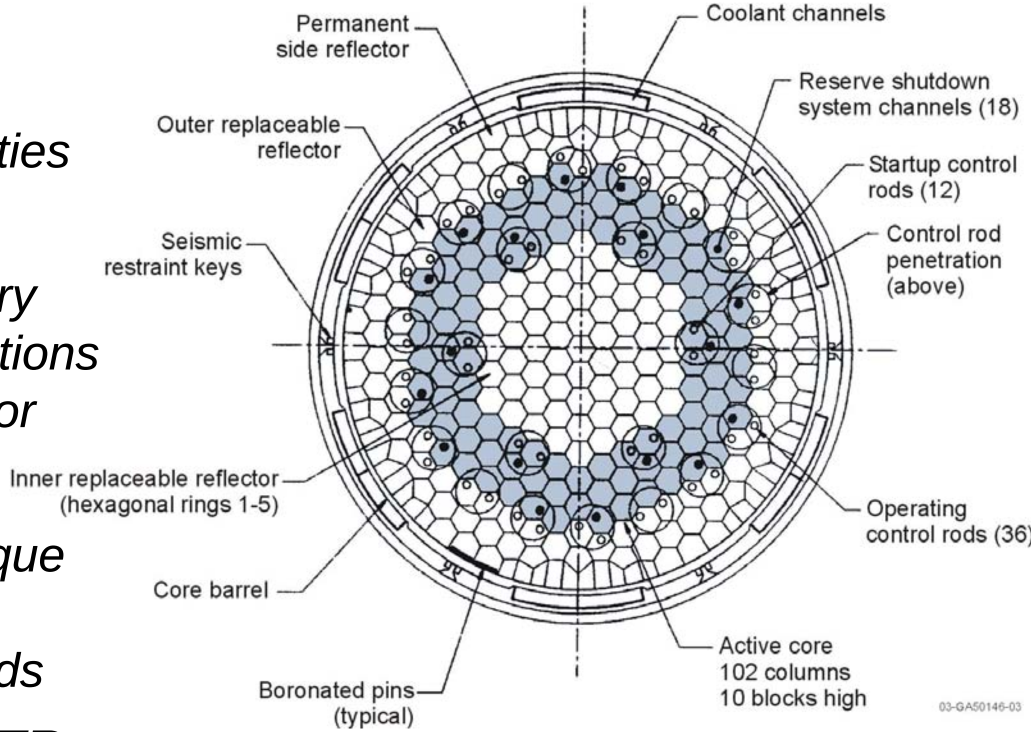
VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 3: Neutronics methods development and validation

- ***Objectives & approach:***
  - *Define a complete suite of codes to perform accurate and valid neutronics analyses for VHTRs*
  - *Identify and begin implementation of needed modifications*
  - *Improve an existing cross section generation code to properly treat low energy resonances and doubly heterogeneous fuel using Dancoff factors*
  - *Assess modeling requirements for characterization of temperature-dependence of displacement threshold energy*

Activity 3: Neutronics methods development and validation (ANL/INL)

- ***Evaluation of DRAGON lattice capability for VHTR Analysis***
  - *Evaluate DRAGON code capabilities and models*
  - *Identify deficiencies and necessary fixes, and any additional modifications that would make code attractive for VHTR analysis*
  - *Interact with Ecole de Polytechnique (Montreal) to obtain details of DRAGON capabilities and methods*
- ***Identify a complete suite of VHTR analysis codes and evaluate strengths and weaknesses, propose upgrades***



VHTR Design Methods Development & Validation FY-05 Status Summary

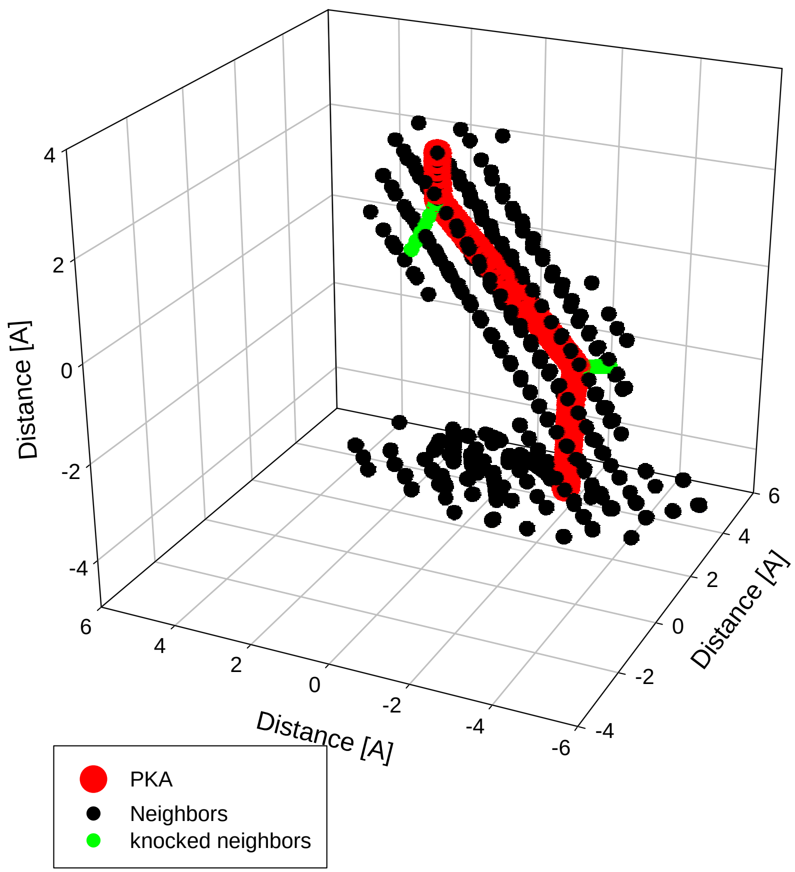
Activity 3: Neutronics methods development and validation (INL, ANL)

- ***Coupling between PEBBED pebble-bed neutronics code and THERMIX thermal-hydraulic code for steady-state and DLOFC analysis was completed.***
- ***Work began on construction of HTR-10 and PBMR-400 neutronics (PEBBED) and thermal-hydraulic (THERMIX,MELCOR) benchmarks. Participation in international working group.***
- ***Completed development of a rigorous method for computing Dancoff factors in double-heterogeneous fuel (to be implemented in the PEBBED code suite)***
- ***Full-range multigroup transport method identified for simultaneous treatment of thermal upscattering and resonances. Implementation underway in COMBINE.***

Activity 3: Neutronics methods development and validation (NCSU)

♦ ***Modeling and computation of radiation damage in graphite***

- *Improved values of irradiation damage at high temperatures requires better displacement threshold energies*
- *The effects of radiation damage and subsequent annealing on reactor feedback mechanisms at high temperatures and fluences (typical of the VHTR) can be as high as 20% (in thermal neutron scattering kernels)*
- *These data and models will be incorporated into safety analysis codes.*



VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 4: Nuclear data tasks

- *First meeting of the International Reactor Physics Benchmark Experiments (IRPhE) Program was held in October of 2004*
- *Four in-core reactor physics benchmarks considered to be nearing a publishable state were formally discussed, along with seven additional evaluations still requiring independent peer review.*
- *The HTR-10 benchmark is underway. A PEBBED reactor physics model has been developed and a COMBINE model is currently being developed*
- *We are quantifying the need for additional cross section measurements for VHTR applications - preliminary sensitivity and uncertainty analyses were performed for various core parameters: k_{eff}, peak power, temperature reactivity effect, and burnup reactivity swing*

VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 5: Molten salt-cooled VHTR methods development and design assessments (ORNL, INL, ANL)

- *The properties of four liquid salts were inserted into RELAP5-3D to enable the RELAP5-3D systems analysis code to be used for liquid-salt VHTR safety analyses*
- *The salts incorporated were: flibe, flinak, 92%NaBf4-8%NaF, and 50%NaF-50%ZrF4*
- *Using RELAP5-3D, liquid salt-cooled VHTR safety analyses are in progress*
- *Due to positive reactivity insertion that may occur due to a complete loss-of-coolant accident, approaches to mitigate problem have been investigated*

VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 5: Molten salt-cooled VHTR methods development and design assessments (ORNL, INL, ANL)

- *Develop revised core design with improved reactivity response (ORNL, INL, ANL)*
- *Perform decay heat removal analyses for revised core geometries (INL)*
- *Assess requirements, trade-offs, existing data, and R&D needs for candidate salts, including properties, chemistry, and material compatibilities (ORNL)*

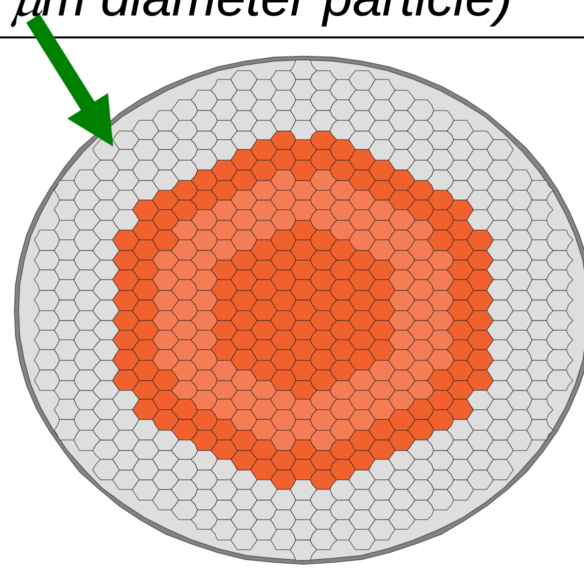
•265 fuel block columns

•10 blocks per column

•10 MW/m^3 power density

•Li_2BeF_4 (Flibe) salt with 99.995% ^7Li enrichment

•425 μm diameter UCO fuel kernel (845 μm diameter particle)



VHTR Design Methods Development & Validation FY-05 Status Summary

Activity 6: Manage the INL VHTR Design Methods Development & Validation project.

- *Four interaction rounds between INL staff, ANL staff, ORNL staff, and several universities have resulted in more detailed defined R&D needs for the VHTR*
- *An informal Advisory Group has been defined—but not formalized—resources are not sufficient*
- *At the GIF VHTR Design & Safety Project Management Board meeting in Washington DC:*
  - *Japanese indicated willingness to release HTTR and other facility data sets*
  - *EU indicated willingness to release AVR and other facility data sets, such as Oberhausen*
- *Development, by the GIF VHTR Methods Project Board, of 10 areas of collaborative R&D between GIF members, will permit a number of the validation requirements for VHTR to be achieved*

Design Methods & Validation International Collaborations

Ten groups of collaborative project areas were defined and will be investigated in the future to develop collaborative efforts via the GIF Methods:

(1) CFD and thermal-hydraulics—lead: U.S.

(2.1) Prismatic core physics & nuclear data—leads: Japan & France

(2.2) Pebble-bed core physics & nuclear data—S. Africa, & EU

(3.1) Air ingress—EU & S. Africa

(3.2) Fission product transport & plate-out—EU & S. Africa

(3.3) Fuel modeling—U.S., France, FCPMB

(4.1) Prismatic reactor & plant dynamics—Japan & U.S.

(4.2) Pebble bed reactor & plant dynamics—EU & S. Africa

(4.3) Power conversion systems—Japan and France,

(5) PIRT—lead: U.S.

Design Methods & Validation I-NERI Projects

- *US/France I-NERI project: “Thermal-hydraulic analyses and experiments for GCR safety,” initiated in February with CEA, ANL, and Utah State University*
- *US/S. Korea I-NERI project: “Development of Safety Analysis Codes and Experimental Validation for a VHTR, 2003-013-K” with KAIST and SNU*
- *US/S. Korea I-NERI project (through WPs: A0802K01 and I0802K01): “Screening of Gas-Cooled Reactor Thermal-Hydraulic & Safety Analysis Tools and Experiment Data Base ,” initiated in 2004 with KAERI and ANL; ANL lead*

Design Methods & Validation Focus Areas

Based on the important phenomena identified to date, R&D at universities would be especially valuable in the following areas…

- *Factors that quantify the bypass flow in both prismatic & pebble-bed reactors. These factors include material changes due to irradiation; core construction and assembly tolerances; and reactor flow distributions as a function of power, geometrical configuration, and inlet conditions.*
- *CFD turbulence modeling techniques required to calculate flow mixing behavior over a wide range of conditions in the presence of cross flow in a single plenum; investigation of RANS, LES, and DNS calculational approaches.*
- *Quantification of potential compressor surges and other initiators that may induce pressure pulse propagation that may influence dust dislodgement and structural challenges.*

Design Methods & Validation Focus Areas

Possible university work continued:

- *Computational simulation and characterization of radiation damage and annealing in graphite and silicon carbide and their effects on neutronic, thermal, and structural properties*
- *Transport methods for generating diffusion theory cross sections in double-heterogeneous fuel and for the integrated treatment of transport zones in diffusive systems*
- *Novel kinetics treatment for high temperature reactors accounting for all feedback phenomena*
- *Experimental validation of pebble flow models*
- *Modeling, analysis, and optimization of heat deposition in high temperature reactors*

Design Methods & Validation Focus Areas

Possible university work continued:

- *Innovative methods for acceleration of Monte Carlo computations involving complex geometric models with high scattering ratios and high dominance ratios*
- *Development of Integral Benchmark Evaluations pertinent to advanced reactor designs according to the procedures of the OECD/NEA International Reactor Physics Benchmark Evaluation Program chaired by INL*
- *Possible collaboration in analysis of data from advanced cross section measurement experiments proposed by INL*

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

S. Michael Modro

Supercritical Water Cooled Reactor (SCWR)

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Demonstrating technical feasibility of a LWR operating above the critical pressure of water, and producing low-cost electricity.**
- **The SCWR project is part of the Generation-IV program.**
- **The Generation-IV program calls for the development of:**
  - the next generation of nuclear systems for production of high-value energy products such as electricity and hydrogen (VHTR and **SCWR**), and
  - development of fast reactor systems for the actinide management mission. (GFR, LFR, SFR)

SCWR R&D BASIS

- **The U.S. Generation-IV SCWR Program operates under the following assumptions which are consistent with the SCWR's focus on electricity generation at low capital and operating costs:**
  - Direct cycle
  - Thermal spectrum
  - Light-water coolant and moderator
  - Low enriched uranium oxide fuel,
  - Base load operation

SCWR R&D ELEMENTS

♦ Key R&D areas identified in the Generation IV R&D Roadmap

- System Design
  - Establishment of a baseline design as a reference for further feasibility and performance evaluation
- Basic Thermal-Hydraulic Phenomena, Safety, Stability and Methods
  - Addresses knowledge gaps in areas such as thermal-hydraulic phenomena expected during normal operation and accidents, system performance under variety of operational conditions, analytical methods needed for safety and system performance assessment
- Materials and Chemistry
  - Identify and develop materials that will assure safe and reliable system operation

FY03/04 KEY ACCOMPLISHMENTS

- Established U.S. reference SCWR design (INL)
- Identified candidate materials for all SCWR components (ORNL, INL)
- Demonstrated SCWR stability against core-wide oscillations (ANL, MIT))
- Developed conceptual design of SCWR containment and established requirements for safety systems (Westinghouse, INL)
- Designed power conversion cycle and identified control and start-up strategies (Burns & Roe)
- Tested variety of alloy samples in SCW (UW, UM)
- Performed initial sub-channel analyses

Topical reports are available from SCWR SIM upon request .

FY04 WORK FOCUS

$850k, 5 Work Packages, 8 Organizations

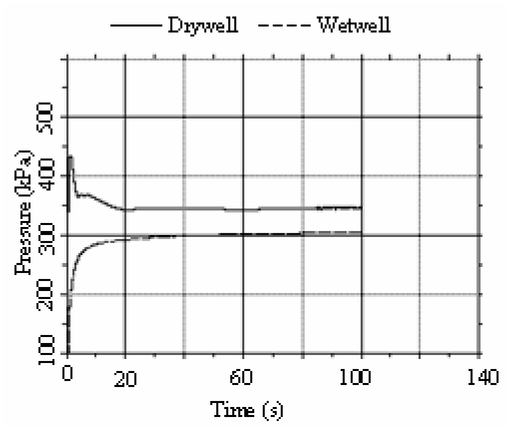
1. **Safety system and containment design (Westinghouse)**
2. **Stability analysis (ANL, MIT)**
3. **RELAP analysis of start-up (BREI)**
4. **Corrosion testing (U-Wisconsin)**
5. **SCC testing (U-Michigan)**
6. **Water chemistry control strategy (ORNL)**
7. **Program management and design support (INL)**

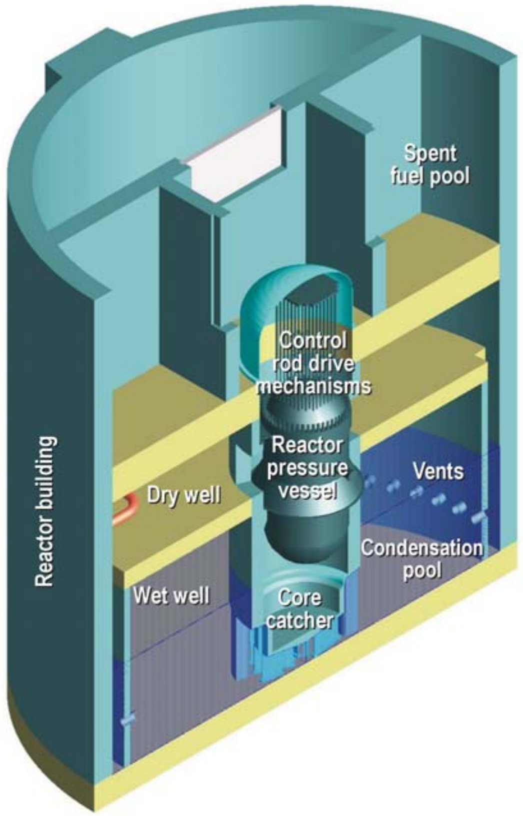
FY04 ACCOMPLISHMENTS (1 of 9)

SCWR CONTAINMENT

- **Pressure suppression type with condensation pool (same as modern BWRs)**
- **Conservative design for severe accidents (as per the European Utilities Requirements):**

  - In 2004 the containment design base developed in previous year was studied and containment responses to loss-of coolant accidents such as pipe breaks was evaluated.
  - The analyses demonstrated that pressures and temperatures are within design limits for all pipe breaks. The design limits have been set within the range of values used for existing containment buildings.





FY04 ACCOMPLISHMENTS (2 of 9)

DYNAMIC ANALYSIS OF THE SCWR REFERENCE PLANT

- **Investigation of thermal-hydraulic and thermal-nuclear coupled stabilities.**
- **Frequency domain linear stability analysis using simplified thermal-hydraulic, fuel heat transfer, and nuclear kinetics models.**

- 04 model improvements:
  - heat transfer from the coolant channel to the water rod
  - delayed reactivity feedback through water rod density variation
  - Jackson correlation for forced convection.
- The effect of water rods on the thermal hydraulic stability is not significant.
- The thermal nuclear coupled stability deteriorated noticeably, due to the delayed moderator feedback effect.

FY04 ACCOMPLISHMENTS (3 of 9)

DYNAMIC ANALYSIS OF THE SCWR REFERENCE PLANT (continued)

- The decay ratios are well below the limits traditionally imposed for BWR stability (0.5 for thermal hydraulic and 0.25 for thermal-nuclear oscillations for normal operation). This implies that the core-wide in-phase oscillations would decay quickly at normal operating power and flow conditions.
- The effect of different heat transfer correlations on the thermal-hydraulic stability was not significant.
  - Jackson correlation resulted in smaller decay ratios for the thermal nuclear coupled stability because of reduced heat transfer from coolant channel to water rod.
- Sensitivity studies performed for various power-to-flow ratios showed the possibility for the current reference SCWR design not to satisfy the BWR stability criteria at reduced power operation
  - At power fractions less than ~25%, the system becomes unstable even with the power-to-flow ratio equal to the normal operation condition.

FY04 ACCOMPLISHMENTS (4 of 9)

STABILITY ANALYSES

- Develop models and evaluate the SCWR stability features at full power supercritical pressure and at partial power subcritical pressure.
- To suggest a suitable SCWR sliding pressure startup procedure that will encounter no instability and burnout problems.

- Through a system response matrix decay ratio calculations, it was found that the U.S. reference SCWR design, will be stable with proper orificing.
- A non-dimensional analysis of the conservation equations in a three region model under supercritical pressure was performed showed that the U.S. reference design would operate in a stable region with a large margin.

FY04 ACCOMPLISHMENTS (5 of 9)

STABILITY ANALYSES (continued)

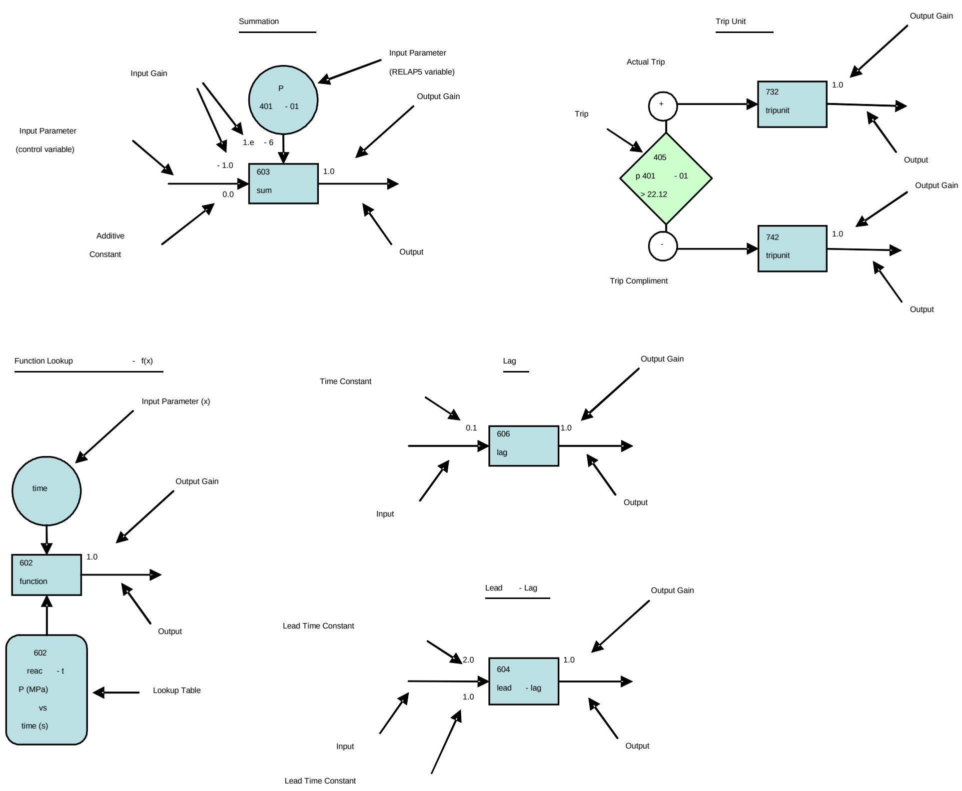
- In the subcritical pressure region the homogeneous non-equilibrium model (HNEM) will predict the most conservative stability boundary at high Subcooling numbers, the homogeneous equilibrium model (HEM) will result in the most conservative stability boundary at low Subcooling numbers. Also, the stability boundary differences between the different models will decrease as the pressure increases.
- A two channel model has been developed for the U.S. reference SCWR core design, including a hot channel and an average channel. It is found that the predicted channel-to-channel stability boundary lies between the single channel stability boundaries of these two channels, i.e., the hot and the average channels.
- By taking CHF avoidance and stability assurance into account, a sliding pressure startup procedure for the U.S. reference SCWR design has been suggested.

FY04 ACCOMPLISHMENTS (6 of 9)

SIMULATION OF SCWR STARTUP USING RELAP5

- Variable pressure startup instability observed shortly after the transition to saturated conditions and at the transition to supercritical conditions.

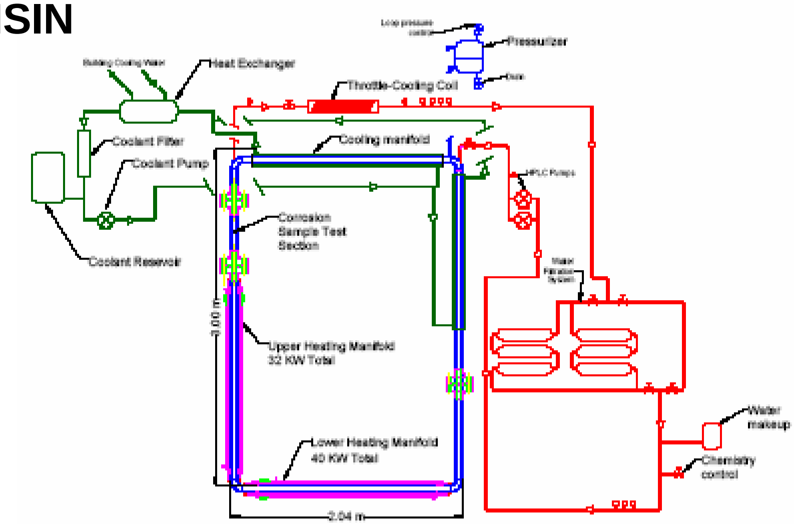
- Constant pressure start-up in which the system pressure is rapidly increased to the critical pressure and then the reactor power is increased until the target power is achieved showed no instabilities.



FY04 ACCOMPLISHMENTS (7 of 9)

CORROSION TESTS AT UNIVERSITY OF WISCONSIN

- 2 three week tests at 500^{o}C
- 25 ppb Oxygen:
  - the oxide thicknesses are greatest in ferritic-martensitic steels, with thinner oxides associated with higher bulk chromium concentration
  - Nickel-base alloys had the smallest oxide growth
  - Austenitic stainless steels had oxides with thickness between the ferritic-martensitic steel and the nickel base alloys
  - alloy 625 did showed signs of pitting corrosion.
  - Silicon carbide did not form an oxide but slowly dissolved
- Increasing the oxygen concentration to 2 ppm increased the oxide thickness for most alloys



500°C 25 ppb Oxygen

Material Type	Material Selected
Austenitic Stainless Steel	316, D9, 347, 800H
Ferritic-martensitic Stainless Steel	HT9, T91, HCM12A, NF616
Surface-modified Ferritic-martensitic Stainless Steel (oxygen implanted)	HT9-sm, T91-sm, T122-sm
Precipitation Hardened Nickel-base Alloy	625, 718
Silicon Carbide	SiC

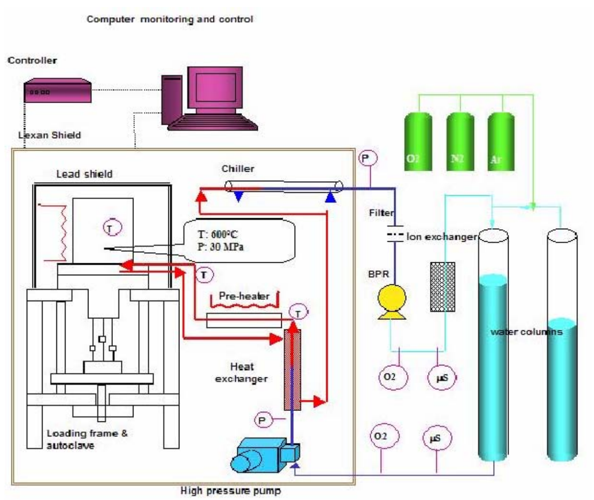
500°C 2 ppm Oxygen

Material Type	Material Selected
Austenitic Stainless Steel	316, D9, 347, 800H
Ferritic-martensitic Stainless Steel	HT9, T91, HCM12A, NF616, T22
Solid Solution Nickel-base Alloy	690, C22*
Precipitation Hardened Nickel-base Alloy	625, 718, 825
Silicon Carbide	SiC**

FY04 ACCOMPLISHMENTS (8 of 9)

STRESS CORROSION CRACKING TESTING AT UNIVERSITY OF MICHIGAN

- Design and construction of the Irradiated Materials Testing Complex that will be used to conduct stress corrosion cracking and corrosion testing in supercritical water, and subsequent fracture and surface analysis on neutron-irradiated materials.
- The facility is currently under construction and will be fully operational and ready to accept neutron-irradiated materials by October 1, 2005.



FY04 ACCOMPLISHMENTS (9 of 9)

COOLANT CHEMISTRY ISSUES AND CONTROL STRATEGY

Experience from fossil plant operation is that controlling the purity of the steam is one of the most important criteria for ensuring the reliability, hence lifetime, of steam turbines.

- Suggested approach to fluid chemistry control in the SCWR
  - **Minimize the ingress of impurities;**
  - **Optimize the condensate and feedwater chemistry so as to minimize the generation of impurities from corrosion/dissolution of the surfaces touched by the fluid; and**
  - **Make appropriate adjustments of the fluid chemistry immediately downstream of the reactor core to compensate for radiolysis-induced impurities.**

SCWR VIABILITY ASSESSMENT – STATUS (1 of 2)

- **Solution for safety systems coping with Loss-of-Feedwater transient and other events with quick core voiding was identified.**
- **Preliminary analyses have shown that the containment designed for SCWR will respond to loss-of-coolant accidents with temperatures and pressures within design limits.**
- **Preliminary subchannel analyses have shown extreme sensitivity of the reference design to hot channel factors leading to unacceptable coolant and cladding temperatures.**
- **Stability analyses showed that the current SCWR reference design U.S. reference design would operate in a stable region with a large margin however it does not satisfy BWR stability criteria at reduced power operation.**

SCWR VIABILITY ASSESSMENT – STATUS (2 of 2)

- **RELAP5 simulations showed that constant pressure start-up procedure yields much more stable conditions than variable pressure start-up.**
- **500 hours tests were performed for corrosion resistance at 25 ppm and 2 ppb of oxygen on variety of alloy samples.**
- **A facility for stress corrosion cracking testing was constructed at the University of Michigan.**
- **Coolant chemistry issues were identified and control strategies proposed**

In summary, the key feasibility issues for the SCWR are the thermal-hydraulic core design and the development of in-core materials.

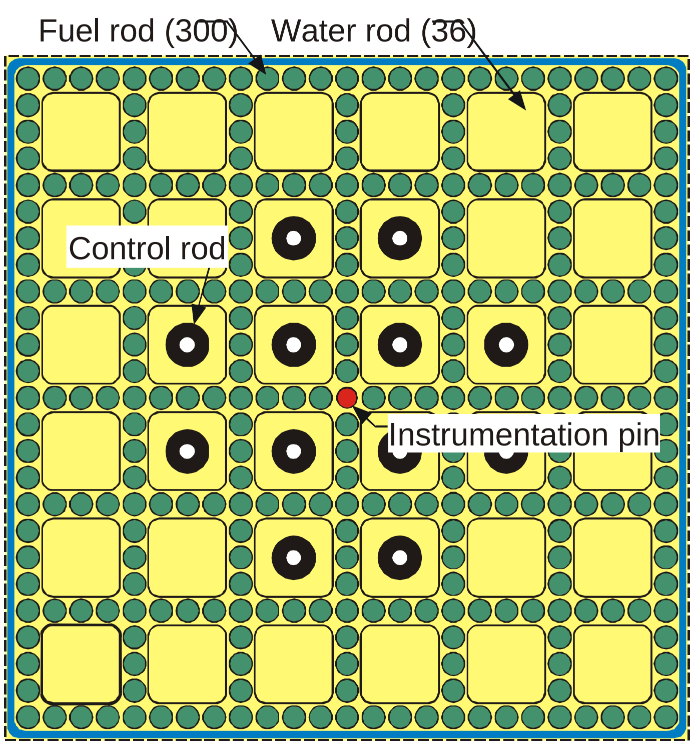
Key Feasibility Issues

♦ Core design

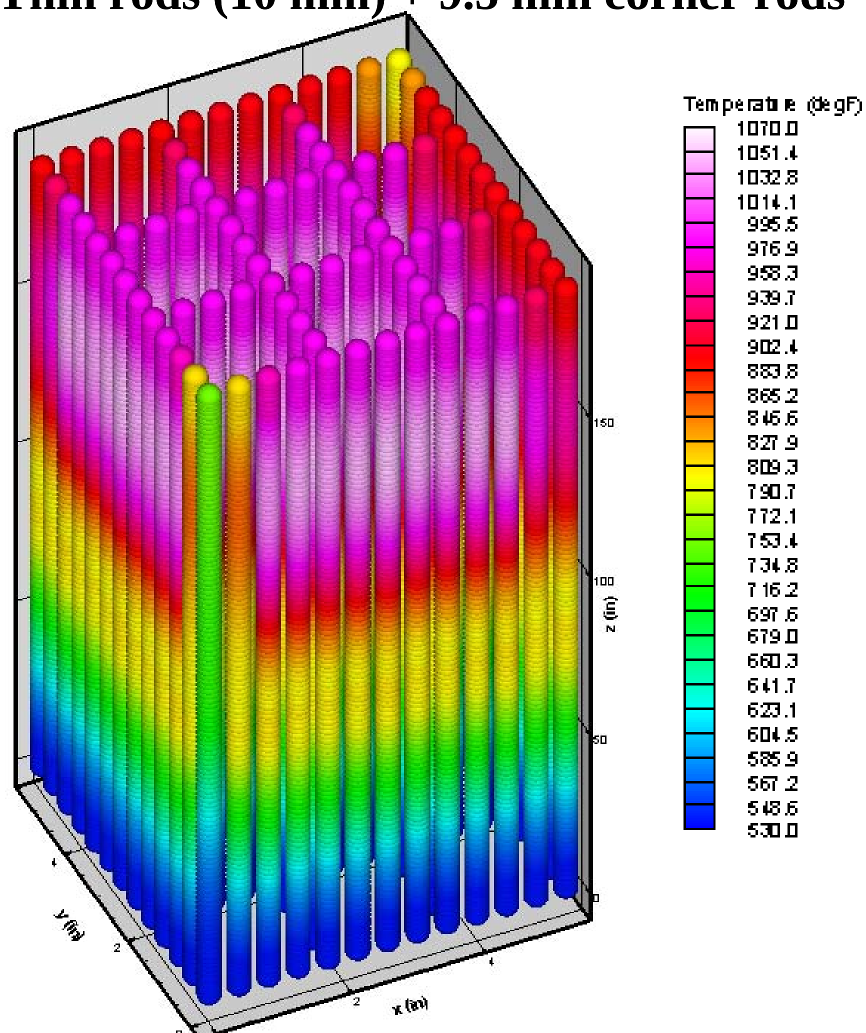
- Large enthalpy rise in the core leads to extreme system sensitivity to hot channel factors resulting possibly in unacceptably high temperatures in some subchannels
- Further studies are needed to enhance understanding of the problem and identify engineering solutions alleviating the problem
  - ***Core design for minimal power peaking***
  - ***Enhancement of turbulence by appropriate mixing devices***
  - ***Enhancement of heat transfer by rough surfaces***
  - ***Enhancement of mass flux by multi-pass core design***
  - ***Optimization of the moderator downward flow ratio***
  - ***Optimization of assemblies and pin arrangements***
- Improvement of analytical fidelity
  - ***Heat transfer modeling***
  - ***Subchannel analysis coupled with neutronics***

Approaches to core design

U.S. Reference design

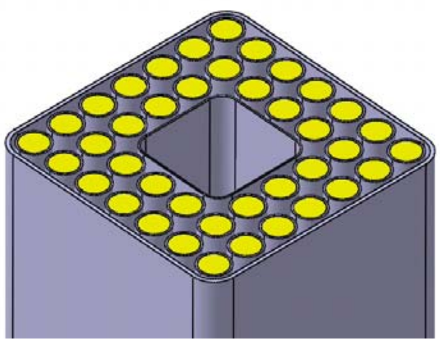


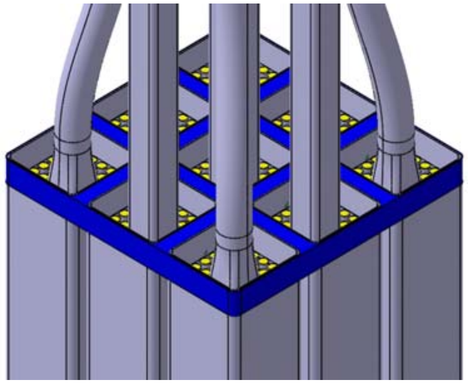
Thin rods (10 mm) + 9.5 mm corner rods

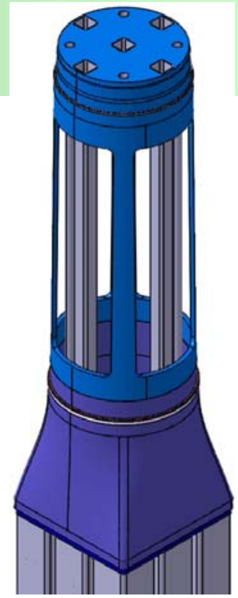


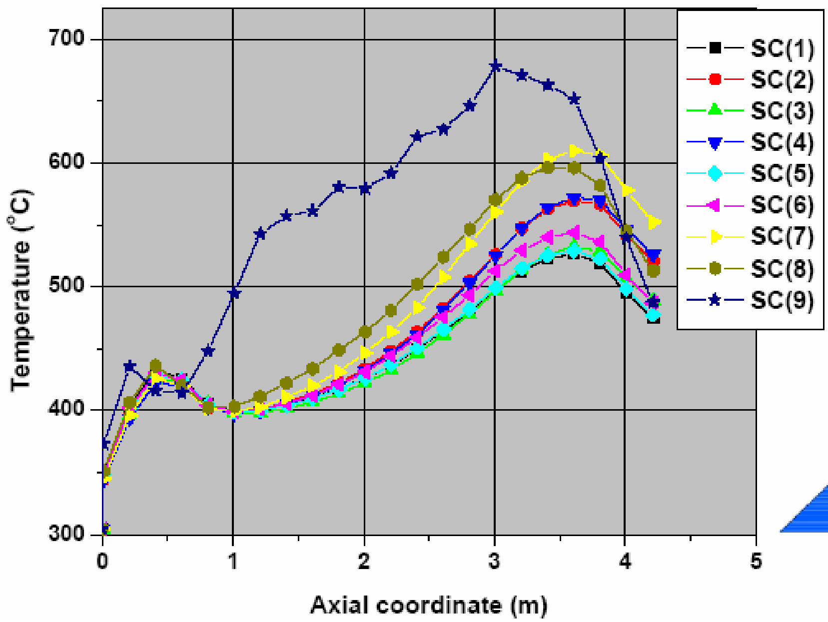
Approaches to core design

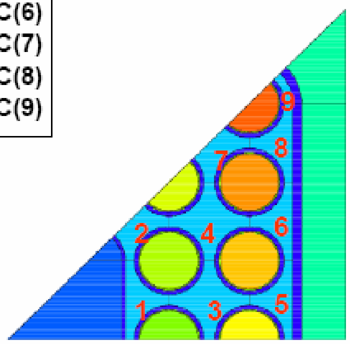
European design

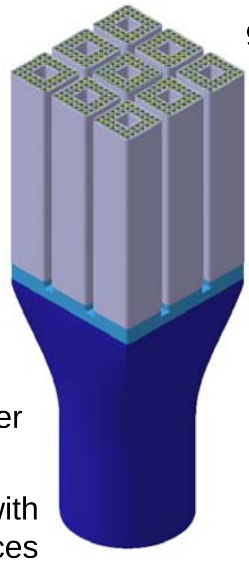












Key Feasibility Issues

- **Cladding and in vessel structural materials**
  - Current core layouts result in high cladding temperatures in subchannels
  - Some designers call for new temperature limits for SCWR cladding (750ºC for normal operation)
  - Future materials R&D needs to address issues such as
    - Surface modification and grain boundary engineering
    - Surface coatings
    - Predictive capabilities of material degradation

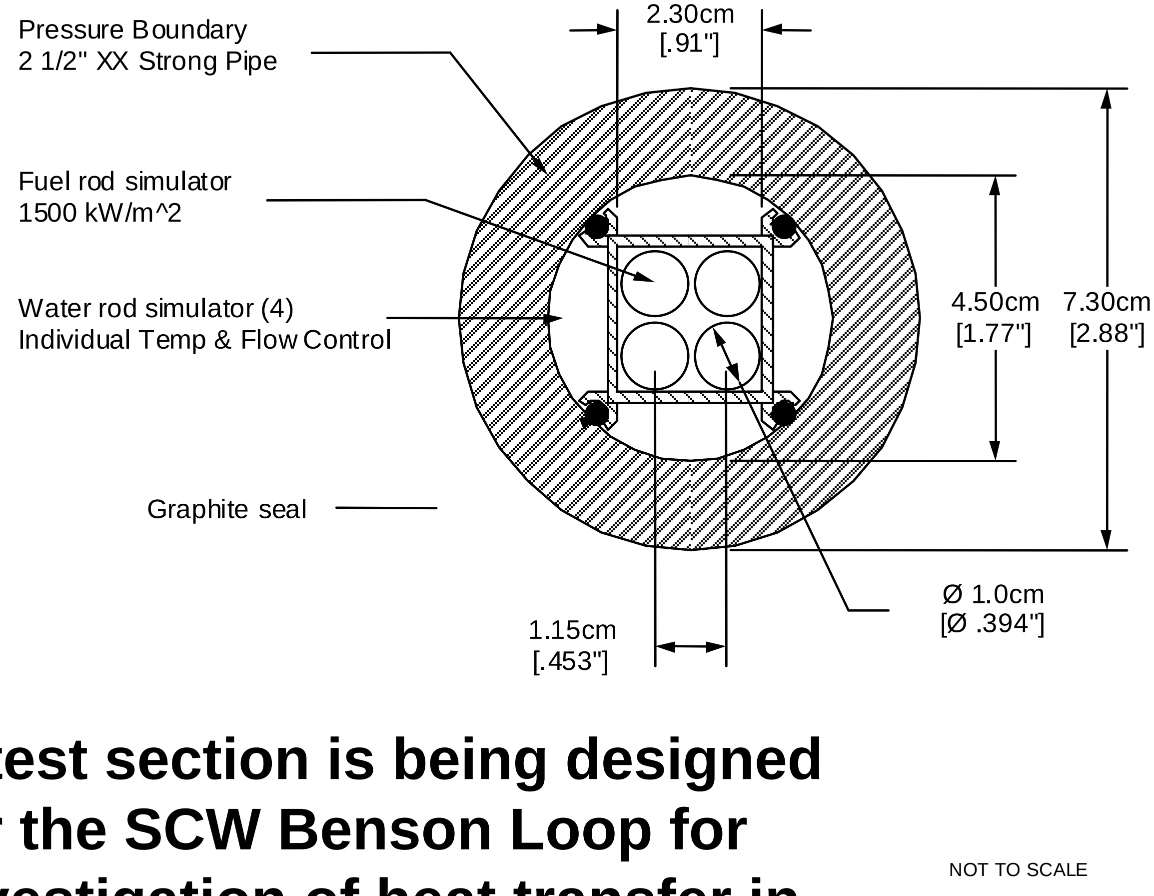
WORK IN PROGRESS FOR FY05

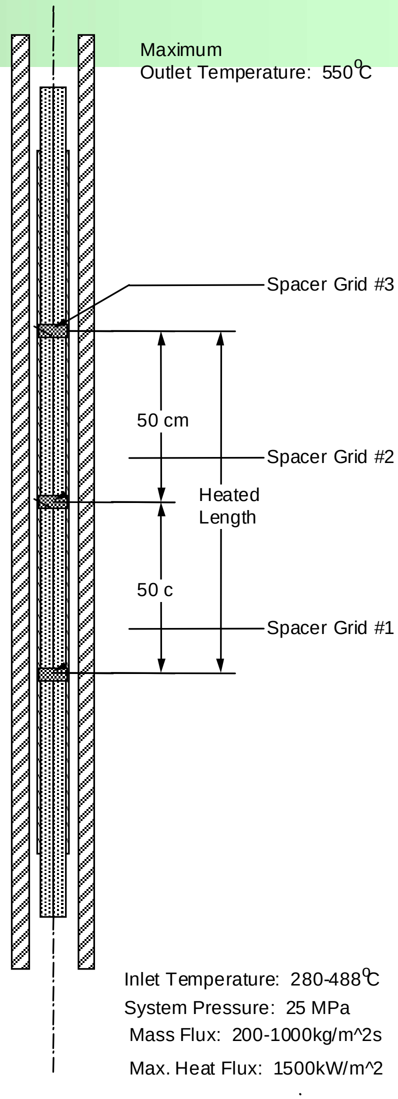
FOCUS ON BASIC THERMAL-HYDRAULIC AND MATERIALS RESEARCH

$800k, 4 Work Packages, 5 Organizations

1. **Stability analysis (ANL)**
2. **Corrosion testing (U-Wisconsin)**
3. **SCC testing (U-Michigan)**
4. **Chemistry and materials review and support (ORNL)**
5. **Bundle test section design and program management (INL)**

WORK IN PROGRESS FOR FY05





- **A test section is being designed for the SCW Benson Loop for investigation of heat transfer in bundles in SCWR conditions**

PLANS FOR FY05-08 (1 of 4)

Focus of the program for the next 3 years will be on:

- Investigation of basic thermal phenomena for the SCWR (e.g., heat transfer in rod bundles, system and CFD codes) – INL, RPI
- Evaluation of dynamic power/flow instabilities – ANL
- System analysis - INL
- Corrosion and stress-corrosion cracking testing of promising materials for the SCWR core and vessel internals (unirradiated and irradiated samples) – UW, UM, ORNL
- Cooperation with foreign partners - INL

PLANS FOR FY05-08 (2 of 4)

Investigation of basic thermal phenomena for the SCWR

- Heat transfer experiments with SCW and surrogate fluids
- Critical flow experiments
- Validation of system codes (e.g., RELAP)
- CFD methods and coupling to system codes

Key milestones:

- Complete construction and shipment of the test-section for the Erlangen facility (2007)
- Setup international project (2006)

International collaborations:

I-NERIs with EU, Canada and Korea

PLANS FOR FY05-08 (3 of 4)

Evaluation of dynamic power/flow instabilities

- Multi-channel instabilities + 3D kinetics
- Instabilities during start-up and transient overpower
- Experiments in SCW and surrogate fluid loops

Key milestone:

Complete preliminary stability analysis (2007)

International collaborations:

I-NERI with Canada

PLANS FOR FY05-08 (4 of 4)

Corrosion and stress-corrosion cracking testing of promising materials for the SCWR core and vessel internals

- Metal alloys (FM, Austenitics, Ni alloys, ODS)
- Unirradiated, p-irradiated and n-irradiated samples
- Test in SCW at controlled temperature, pressure, oxygen, conductivity and pH

Key milestone:

- Complete corrosion and SCC screening tests of unirradiated materials in supercritical water (2007)
- Setup international project (2006)

International collaborations:

I-NERIs with EU, Canada and Korea

LIKELY AREAS FOR UNIVERSITY SUPPORT - NERI

- Core design issues
  - ***Core design for minimal power peaking***
  - ***Enhancement of turbulence by appropriate mixing devices***
  - ***Enhancement of heat transfer by rough surfaces***
  - ***Enhancement of mass flux by multi-pass core design***
  - ***Optimization of the moderator downward flow ratio***
  - ***Optimization of assemblies and pin arrangements***
- Neutronic/thermal-hydraulic coupled analyses (subchannel)
- Comparison of different approaches (e.g., hydride fuels, solid moderators, water rods)
- Bound effects of hydrogen in supercritical water
- Fuel pin behavior – higher than LWR operating temperature
- Materials R&D needs
  - ***Surface modification and grain boundary engineering***
  - ***Surface coatings***
  - ***Predictive capabilities of material degradation***

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Lead Cooled Fast Reactor

Breakout Session

Doug Crawford/INL

Bill Halsey/LLNL

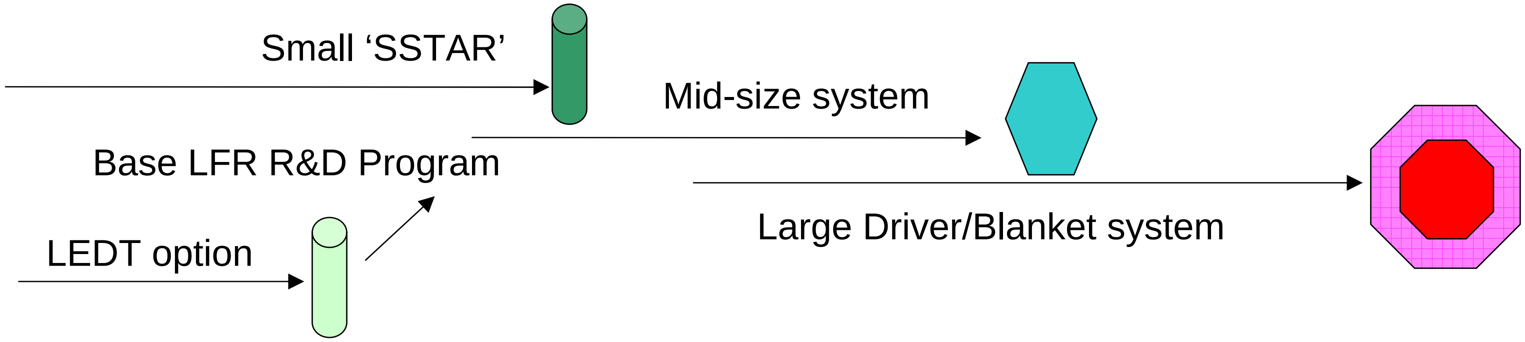
Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Lead-cooled Fast Reactor Research Program

- **R&D objective is a nuclear energy system for deployment in small/remote markets and in developing countries. Low cost, simple initial design enables early LFR technology demonstration and deployment, followed by evolution to larger systems and higher temperatures.**
- **Desired attributes include**
  - Proliferation resistance through long core lifetime with no on-site fuel handling, passive safety, modular factory construction, semi-autonomous load following, and simplified operation with small staff.
- **R&D elements are focused on: definition of the reference system design, coolant and materials issues unique to the LFR, evaluation of the safety case and 'license by test' approach, and understanding deployment and institutional issues unique to small transportable systems.**

LFR Development Pathway

- **Option for early Demo/Test reactor (LEDT):**
  - Comparatively low cost: small, simple, modest technology requirements
  - Demonstrate: autonomous control, natural circulation, safety, …
  - Test: fuels, materials, coolant chemistry, …
- **R&D focus on advanced small Pb reactor (see point design)**
  - Low unit cost, special market lowers barriers to early fast reactor deployment
- **Evolution to medium size modular reactors**
  - Higher temperatures, H-production, …
- **Technology for large base load breeders when needed**



Small, Secure, Transportable, Autonomous Reactor

- **The initial question asked was: How to provide the benefits of nuclear energy to the developing world with strong proliferation-resistance?**
- **Provide energy without need for extensive nuclear infrastructure, technology, materials. This points to a "User-Supplier" paradigm:**
  - Trusted suppliers are responsible to the international community for control of materials, technology and facilities. (Process to gain trusted supplier status.)
  - Users have competitive commercial access to energy.
  - Look at commercial airliner business as model.
- **Developing Countries are the next Major Electric Power Growth Area, and will Look to Developed Countries for Solutions, specifically, they will look to Russia, Japan, France, and the U.S.**
- **Small, low maintenance systems will also be beneficial to remote locations in the U.S. and elsewhere (e.g., Alaska, Hawaii, Japan …)**
- **Improved grid reliability using distributed power sources may also benefit from the availability of small, robust systems**

SSTAR Basis for Small Modular Reactor focus of LFR: addressing regional power needs

- **SSTAR Reactor requirements:**
  - Small increments of electric power (10-50 MWe) for developing markets
  - Simple controls, passively safe and low maintenance power plants, minimal operator intervention desired
  - Reliability in power availability over long periods of time - long core life
  - Stability in the price of electricity and low investment risk
  - No on-site fuel handling, storage or user access - sealed transportable units delivered to site by supplier, long-core life, entire reactor or core cartridge retrieved by supplier at end of life.
- **SSTAR evaluation of LWR, GCR, MSR, LMFR showed benefits of fast spectrum to achieve long core life with near unity conversion ratio, intrinsic safety and potential autonomous operation of low power density LMFR.**

Small Liquid Metal Cooled Fast Reactors are Proposed for a Specific Market Need

Why lead or lead-bismuth: some basic characteristics of liquid metal coolants

Coolant	Melting Point (°C)	Boiling Point (°C)	Chemical Reactivity (w/Air and Water)
Lead-Bismuth Eutectic (Pb-Bi or LBE)	125	1,670	Inert
Lead (Pb)	327	1,737	Inert
Sodium (Na)	98	883	Highly Reactive

Lead and Lead-Bismuth Coolants Offer Promising Overall Characteristics

Small modular reactors are being developed through the DOE Generation IV Program

VISION:

The vision of the LFR program is the future commercial deployment of advanced small reactors (10-100 MWe), which are highly proliferation resistant (with no on-site storage or handling of fuel), employ sealed reactor cores with lifetimes of up to 30 years, are economic and simple to operate, and are deployable virtually anywhere in the world.

MISSION:

The mission of the LFR program is to research, design, develop and demonstrate an advanced small reactor that is safe, secure, transportable and highly proliferation resistant, such that the concept is prepared for commercialization by 2025.

The impact of successfully completing this mission will be the commercial deployment of a new type of small, proliferation resistant reactors

LFR and Advanced Fuel Cycles

- **Advanced fuel cycles can benefit from the ability of fast spectrum reactors to use a wide range of actinides for fuel and the ability to minimize waste and maximize fuel resource utilization. However, small, long-core-life fast reactors such as the current LFR design have unique characteristics in an evolving AFC:**
  - Conversion ratio is near unity to provide long core life.
  - All the fuel is provided 'up front'
  - Fuel is returned to supplier at end of life.
  - Our current design can run on Pu + minor actinide mix from LWR fuel processing, along with depleted uranium. It can also run on equilibrium actinide mix from processing its own spent fuel (plus uranium).
- **The LFR uses a few tons of Pu/MA for several decades and then gives nearly the same amount back when the fuel is processed, effectively providing 20-30 years secure storage while providing energy!**

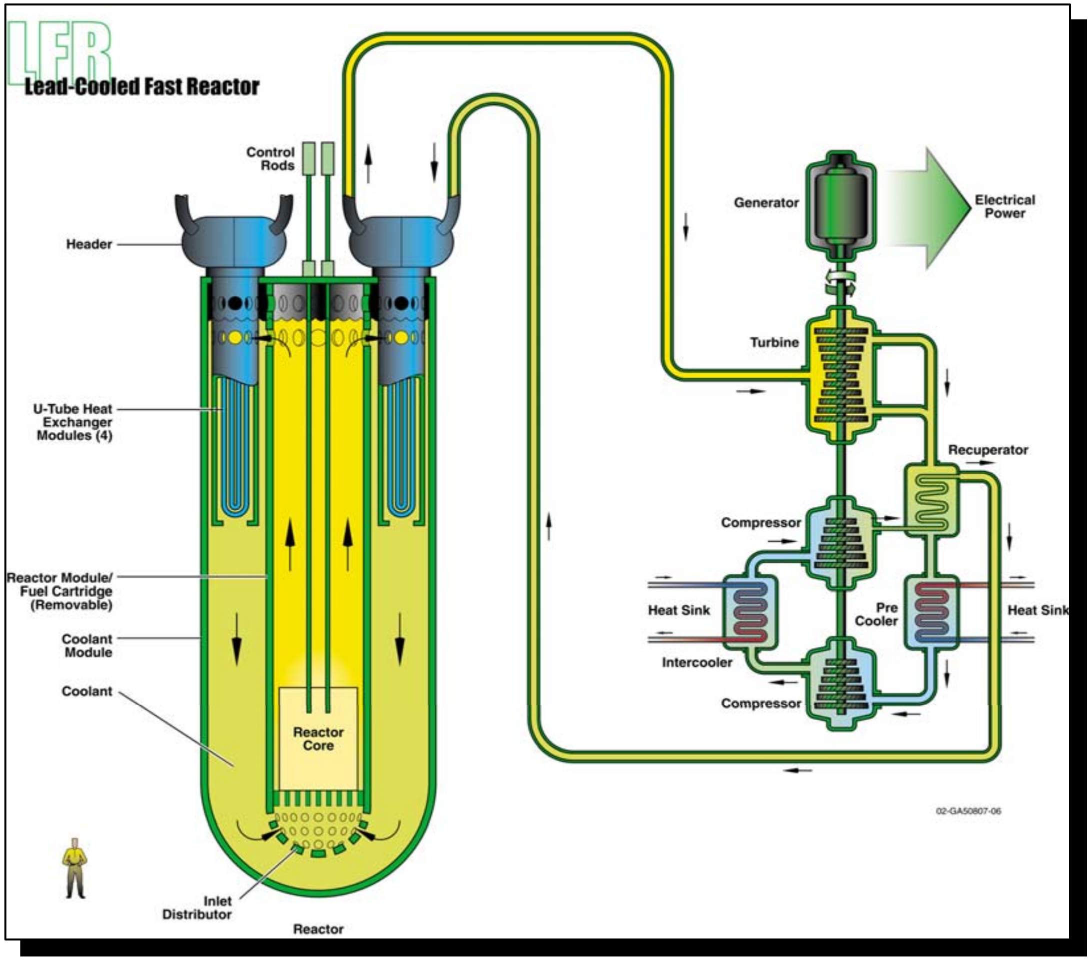
With their modest cost and special market, small LFRs could provide an early path to fast reactor deployment, absorb fissile material from early AFC deployment, and return that material in the future if/when large fast reactors are desired.

Small LMRs have innovative potential for attractive economics

- **Previously considered small reactor designs have not been economic - mostly due to economy-of-scale considerations that motivate use of larger plants**
- **However, innovative attributes of the small liquid metal-cooled concepts offer prospect for improved economics of smaller systems:**
  - Reduced pressurization sources allow small containment, which historically had motivated larger plant sizes
  - A long-life, sealed core serviced by a regional fuel supplier allows deployment with no associated fuel handling or fuel management infrastructure, reducing financial and licensing burdens at the plant site
  - Design for a passively safe core and primary system allows construction of a non-safety-grade balance of plant
  - Incorporation of advanced energy conversion, using supercritical CO_2 or supercritical steam, allows construction of a balance of plant with a considerably reduced footprint
  - Low primary system pressure reduces design demands on the primary system boundary
  - A primary system with enhanced natural circulation allows reduced primary pump requirements, and possibly elimination of primary pumps altogether
  - A plant system with passive thermostructural feedbacks allows incorporation of autonomous load following, reducing the complexity and operational requirements of control systems
- **These attributes, along with the economics inherent in the envisioned deployments, which are different from the economics of large, base-load deployments in the U.S., suggest that an economic small reactor system can be realized**

Design Objectives for the Small Modular LFR

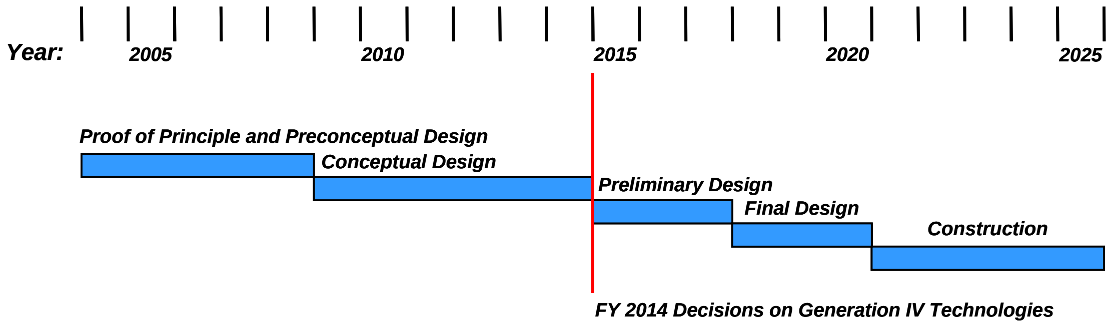
- Deployable in remote locations without supporting infrastructure (output, transportation)
- High degree of proliferation resistance
- 15 to 30-yr core lifetime
- Passively safe under all conditions
- Capable of self-autonomous load following
- Natural circulation primary
- Fuel cycle flexibility
- Options for electricity, hydrogen, process heat & desalination
- Licensable through testing of demonstration plant



Note: Person in figure not drawn to scale

LFR Program Demonstration Objectives

- **Start-up of a demonstration facility by 2025.**
- **Operation of such a facility will demonstrate all or much of the following:**
  - Passively safe operation & autonomous load following
  - License by test, leading to design certification (as with airplanes certified by the FAA)
  - Modular reactor fabrication & assembly
  - Operation with a sealed core
  - Long-life suitability of the core and primary system components



Technical Challenges provide focus for needed R&D

- Material Compatibility – Pb-Bi coolants are corrosive to steels and require careful chemical monitoring and control
- Coolant Activation – Pb-Bi yields radioactive Polonium (Po)
- Durability and inspection of fuel, cladding and structural components for low maintenance long-life core
- Core life limits: fuel fissile content, fast neutron fluence
- Reactor coolant technology: instrumentation, chemistry
- Advanced control concept to enhance autonomous control
- Advanced siting/licensing approach

Background history on lead-bismuth reactors & current research

Russia - Mid 1960's to 1990

- Built and operated 7 “Alpha Class” Submarines (Approx. 60 MWe)
- Built 2 on shore prototypes
- Ongoing work on ADS systems

Germany - 2000 to Date

- FZK has constructed 3 experimental test loops using Lead-Bismuth

Japan - Late 1990's to Date

- Built experimental test loop
- Toshiba concept of a Pb-Bi cooled 4S reactor

Sweden - 2003

- Royal Institute of Technology building large Lead-Bismuth Experimental Loop

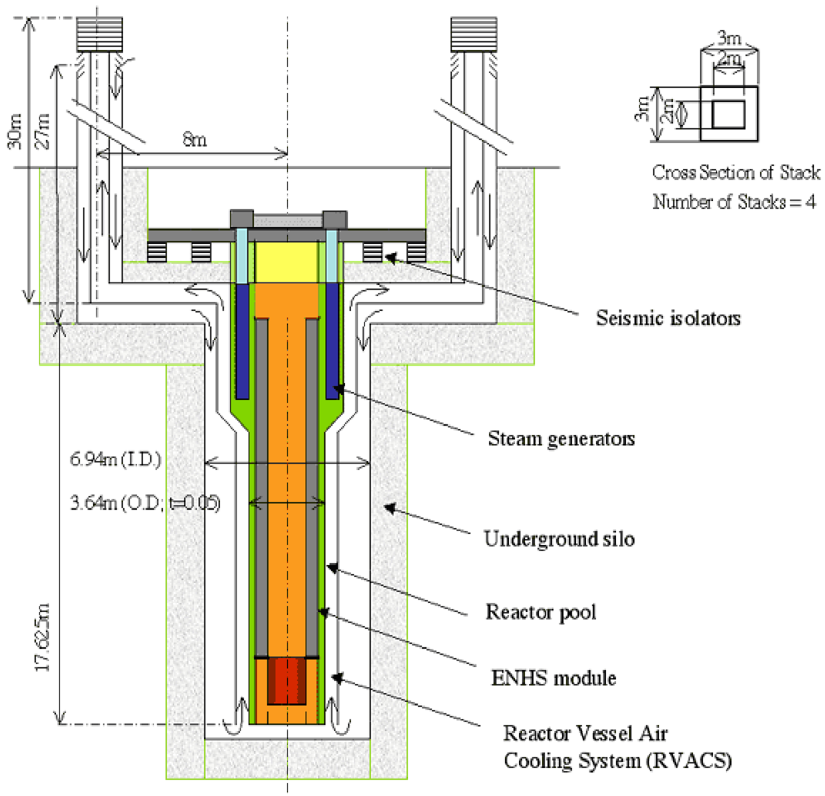
U.S. Programs - 2000 to Date

- Los Alamos National Laboratory - Delta Loop in operation since 2003; corrosion testing
- University of Nevada at Las Vegas - Lead-Bismuth Loop recently installed
- MIT - alloy studies to mitigate corrosion
- LLNL/ANL/LANL - STAR and SSTAR initiatives
- UCB/ANL/LLNL - ENHS

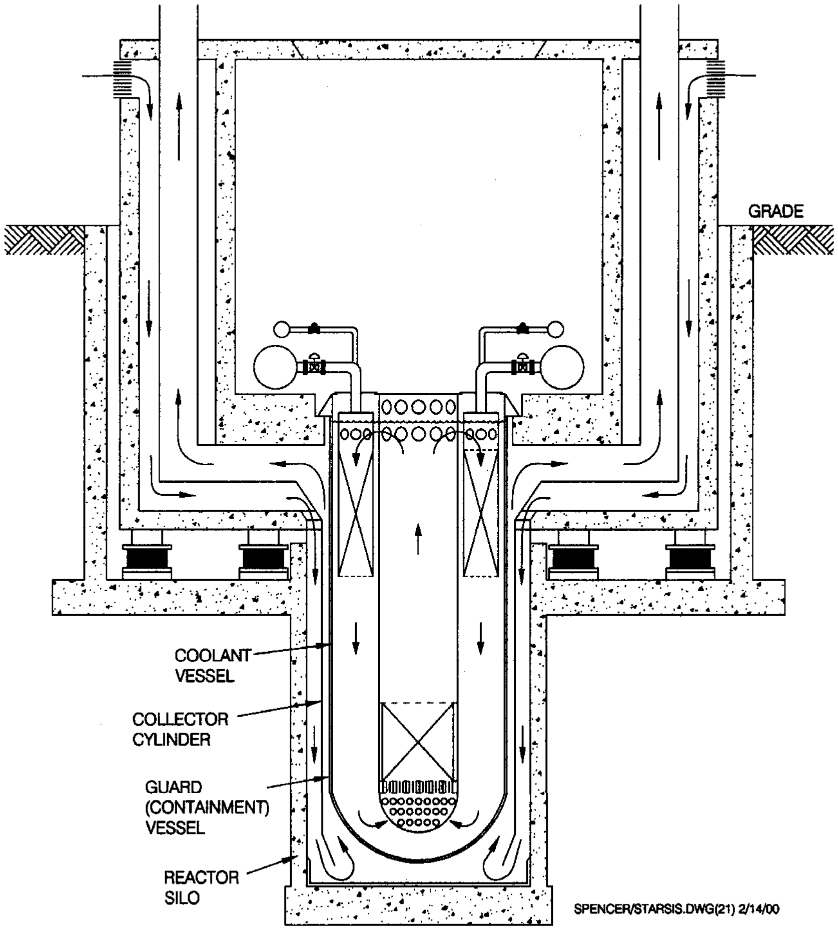
UC Berkeley, LLNL, ANL and Westinghouse developed STAR-ENHS innovative concept

- 3-year NERI study with UCB, ANL, Westinghouse, KAIST and CRIEPI completed in FY02
- Evolutionary concept developed from CRIEPI-Toshiba 4S reactor
- Natural circulation cooling
- Reactor core heat transferred from primary to secondary Pb-Bi through capsule wall
- Fuel contained in capsule throughout fuel cycle
- Engineering feasibility being considered by analysis but economic feasibility remains uncertain

Schematic vertical cut through the ENHS reactor



ANL is developing a SSTAR-LM concept that uses Pb or Pb-Bi coolant



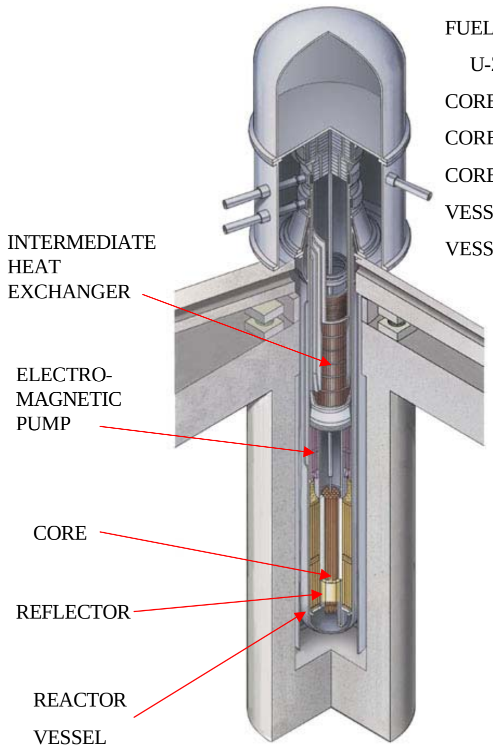
STAR-LM Features

- **More conventional design than ENHS**
- **Natural circulation cooling**
- **Cartridge core design with 15–30 year cartridge life**
- **Core replacement storage and shipping to be developed**
- **Coolant and materials development required**
- **Cost estimates need to be developed**

Japanese collaborator (CRIEPI) and Toshiba have developed the 4S reactor design

- **Inherent safety features are robust**
  - All reactivity feedback coefficients including coolant void reactivity are negative
  - Fully passive decay heat removal system
- **Economic potential needs to be confirmed**
- **Achieving long life and sealed core objectives may depend on selection of coolant, sodium or heavy metal**
- **Origin of ENHS & SSTAR concepts but does not emphasize security features**

REACTOR ASSEMBLY



Major Specifications

FUEL :
U-Zr METALLIC FUEL
CORE LIFETIME : 30Years
CORE HEIGHT : 2m
CORE DIAMETER : 1.1m
VESSEL HEIGHT : 15m
VESSEL DIAMETER: 2.7m

Organization of LFR R&D in U.S.

- R&D elements are directed toward definition of the reference system and then selected to provide design information
- Design and R&D will be focused by preparation of a defendable safety case for the reactor system, which be subsequently used in licensing

- **Reactor Design Objectives**
  - Core and heat transfer design options
    - Long-life & natural circulation
  - Safety evaluation
    - Passive response
  - Control design
    - Autonomous load following
    - Simplified mechanical design
  - Coolant properties
    - Verification of flow and heat transfer characteristics

- **Materials R&D Objectives**
  - Materials selection
    - Long-exposure times
    - Compatibility with liquid metal coolant
  - Irradiation performance
  - Database for material qualification
    - Compilation of properties
    - Demonstration of performance
    - Licensing support

U.S. LFR R&D organization (cont.)

- **Objectives for Deployment and Institutional Issues**
  - Economics & Proliferation Assessment
    - Determine requirements
    - Evaluate design options
  - Licensing Strategy
  - Scope of International Cooperation
- **Coolant Technology Objectives**
  - Establish techniques to monitor coolant chemistry & flow
  - Chemistry control
- **Fuel Technology Objectives**
  - Fuel selection
    - Long-life requirements
    - Fabrication & recycle
  - Fuel design
    - Enhance passive response
    - Allow long lifetime
  - Fuel development
    - As needed for application
  - Fuel qualification
    - Irradiation testing
    - Licensing support

LFR Viability R&D Areas

- **System Design & Evaluation**
  - Long-life core design, near unity conversion ratio
  - Thermal hydraulic design for passive safety, natural circulation, and autonomous load following
- **Materials**
  - Material challenges: Pb/LBE, fast neutron fluence, time/temp
  - Corrosion testing & modeling, radiation damage models, material design, interface with fuel (cladding)
- **Coolant Technology**
  - Instrumentation, testing, modeling: flow, chemistry control, …
  - Thermal & hydraulic properties
- **Institutional and Deployment Issues**
  - Deployment analysis: factory production, transportation, economics
  - Non-proliferation requirements & assessment

FY03 ACCOMPLISHMENTS

- **Reactor Design & Coolant Technology**
  - Determined LFR attributes and recommend priorities for LFR R&D
  - Sealed-core reactor issues framework established
  - Prepared LFR R&D plan
- **Materials**
  - Draft a materials needs document
  - Draft an irradiation testing needs document
  - Materials Screening Tests
    - Corrosion testing of MA957 ODS and SiC-SiC at 800°C in lead
    - Corrosion testing of ODS and V-4Cr-4Ti at 550°C in LBE
- **Institutional & Deployment Issues**
  - Started formulating licensing and safety approach
  - Economic studies initiated

FY04 ACCOMPLISHMENTS: Reactor Design

- **LFR design concepts have been focused to the smaller power sizes desired for remote applications:**
  - 10 to 25 MWe; Pb over LBE; T_{out} limited for $T_{clad} \leq 650°C$; SCO_2 Brayton cycle; Nitride fuel pellets;
- **Natural circulation, autonomous feedback design scaled down from larger system to 45 MWth/18 MWe with CO_2 Brayton cycle. Trade-off between burn-up reactivity swing and compensation rod worth points to tight core with large fuel pin diameter, modest flux, nitride fuel.**

Cartridge core change-out conceptual design.

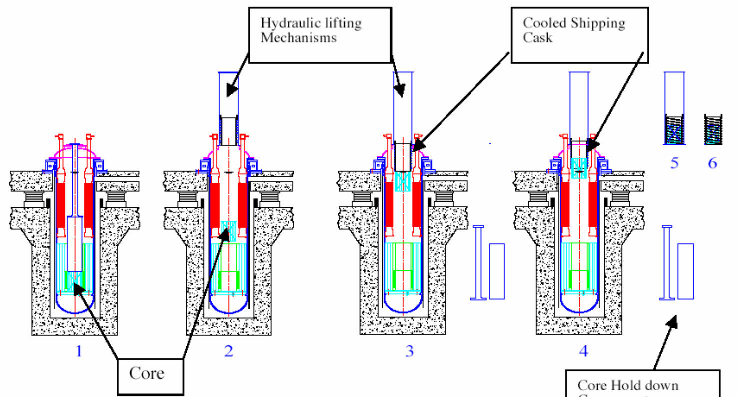


Figure 4 Core Removal Sequence

FY04 ACCOMPLISHMENTS: Materials & Coolant Tech

- **Completion of 1,000-hr 450°C and 400-hr 520°C DELTA tests of 20+ materials, including surface-treated materials**
- **Analysis of 1,000-hr corrosion tested materials (optical and SEM)**
- **Design of a natural convection lead correlation stand**
- **Initial 'LFR Materials & Coolant Technology Plan' drafted.**
- **Materials Screening Tests**
  - 1,000 hr test of MA957 ODS and SiC-SiC to 650°C in lead

- **Materials tested: 20+**
- **Three time intervals (333, 667, 1,000 hrs)**
- **Test temperature: ~ 450°C**
- **LBE flow velocity: 1.5 m/s (?)**
- **Oxygen concentration: varied due to cleaning of excess oxides, target 10^{-6} wt% could be achieved but not maintained for extended periods**

Material	Cr wt%	Si wt%	Ni wt%	C wt%	Mo wt%	Mn wt%
FeCr1	1	-	-	-	-	-
FeCr2	2,25	-	-	-	-	-
FeCr3	9	-	-	-	-	-
FeCr4	12	-	-	-	-	-
FeSiCr1	2,25	0,5	-	-	-	-
FeSiCr2	2,25	1,25	-	-	-	-
FeSiCr3	12	0,5	-	-	-	-
SiFe1	0,09	1,24	0,08	0,01	-	-
SiFe2	0,08	2,55	0,15	0,02	-	-
SiFe3	-	3,82	-	0,01	-	-
pure iron	-	-	-	-	-	-
T91	8,26	0,3	0,13	0,1	0,95	0,38
EP823	12	1,3	0,8	0,18	0,9	0,8
HT-9	11,5	0,4	0,5	-	-	0,6
316L	17,3	0,35	12,1	0,02	2,31	1,8

FY04 ACCOMPLISHMENTS: Institutional & Deployment

- **Economic factors for factory production/modular installation have been evaluated to guide costs and benefits that can be optimized in the system design.**
  - Small size gives up "economy of scale".
  - 30 years of "built-in" fuel add to capital cost.
  - Modular "mass" production, modest field preparation, transportability and rapid installation reduce capital and financing costs. (est. time to field ~1 year)
  - Simplified operating requirements (small staff), no fuel handling facilities, no refueling outages, non-safety secondary all reduce O&M costs.

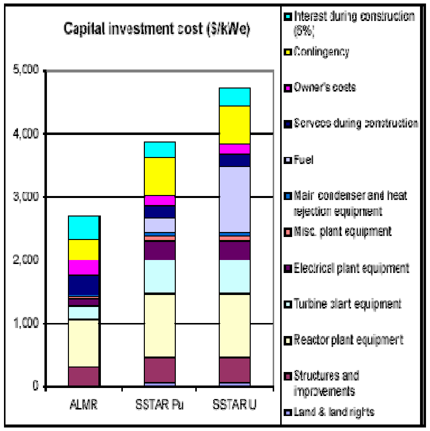


Figure 1. Capital investment cost estimates for 6 percent annual rate of interest

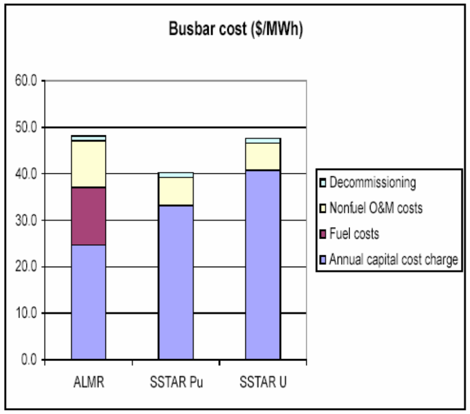
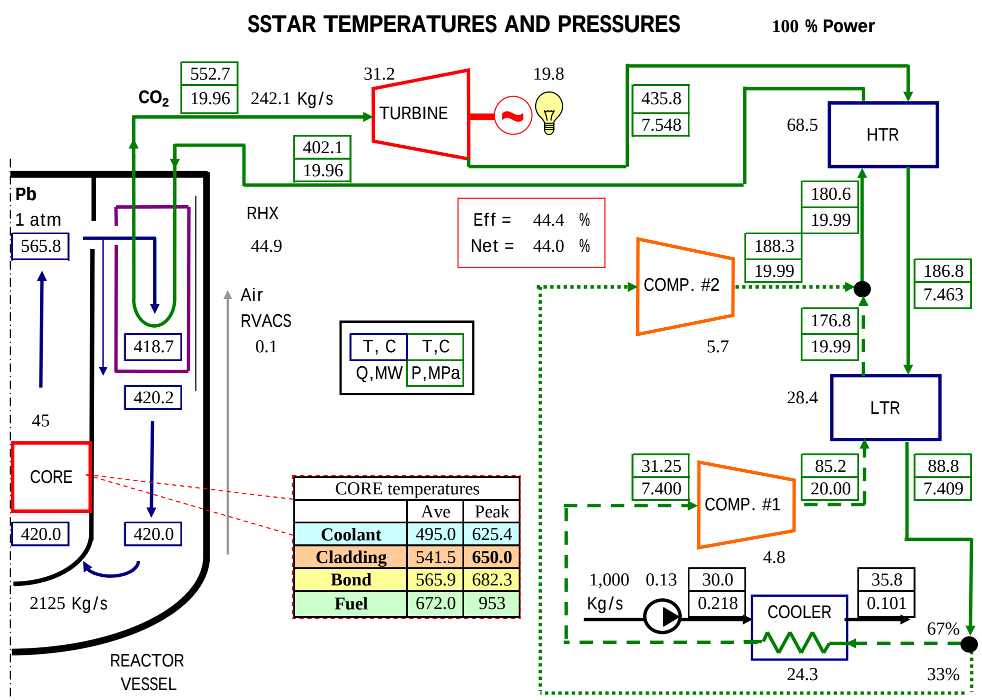


Figure 2. Busbar cost estimates for 6 percent annual rate

WORK IN PROGRESS FOR FY05: Design

- **Development of integrated design to serve as basis for material, fuel, design sensitivity and systems trade studies.**
  - Component evaluations: compact steam generators, secondary systems, seismic issues, safety systems, …

Nominal Operating Conditions for SSTAR withS-CO_2 Brayton Cycle Power Converter



Working Design Highlights

- A 20 MWe (45 MWt) Small Secure Transportable Autonomous Reactor (SSTAR) proliferation-resistant and passively safe fast reactor concept for deployment at remote sites has been developed from synthesis of core neutronics, fuel pin mechanical, and system thermal hydraulics analyses
- Good core, thermal hydraulics, and power conversion performance
  - 20-year core lifetime
  - Average discharge burnup = 72 MWd/Kg HM
  - Compact core (1.0 m active dia X 0.8 m active height)
  - Burnup reactivity swing = 0.96 $
  - Pb void worth = -0.71 $
  - Peak cladding temperature = 650°C
  - Core outlet/inlet temperatures = 566/420°C
  - Peak transuranic nitride fuel temperature = 953°C
  - Small reactor vessel (18 m height X 3.3 m dia)
  - Autonomous load following
  - S-CO_2 Brayton cycle power conversion efficiency = 44.4 %
  - Plant efficiency = 44.0 %

WORK IN PROGRESS FOR FY05: Materials & Coolant

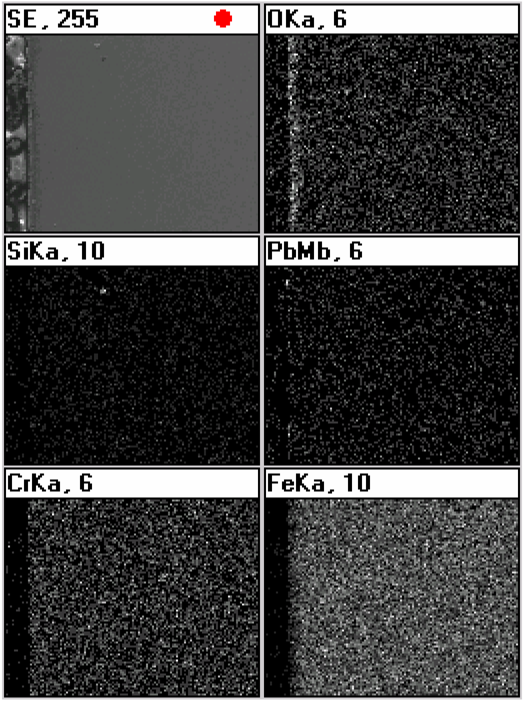
- ♦ **Higher temperature DELTA loop testing (520°C):**
  - Screen new materials (amorphous …), coatings (aluminide...), treatments (laser peening …)
- ♦ **MA 957 ODS alloy exposed to Pb and Pb-Bi eutectic in quartz thermal convection loop: 1,000 hours, 650°C**
- ♦ **Design requirements for Pb/LBE engineering test facility**

520^{o}C DELTA Test

•~20 Alloys (HT-9, T91, EP823, 316L, Fe-Si and Fe-Cr-Si alloys, and aluminum-coated and shot-peened 316L) tested in LBE, flow ~ 1.5m/s, oxygen ~ 10^{-6} wt%, time interval: 133, 267, 400 hrs.

•Protective oxides formed on F/M steels

•Aluminized 316L well protected, shot-peened 316L showed enhanced corrosion resistance (study supported by AFCI)



WORK IN PROGRESS FOR FY05: Institutional, Deployment, International

- **Institutional & Deployment**
  - **Evaluate NRC licensing and safety approach developments**
  - **Deployment cost/benefit**
- **The LFR System Steering Committee has been formed under the GIF. Members include: US, Euratom, Japan, S. Korea**
- **Other International Cooperation**
  - **A coordination meeting was held on small-modular reactor technologies with CRIEPI.**
  - **Continuing I-NERI with ROK KAERI/SNU (jointly supported with AFCI WPs)**
  - **Provided F/M steel specimens (HT-9, T91) for W and Mo surface-coating through STCU collaborations in Ukraine**

PLANS FOR FY06-08 (shown as milestones)

- **FY06**
  - **Preconceptual design viability evaluations including reactivity control, system, heat transport and emergency heat removal.**
  - **Initiate studies of potential alloy modification, surface treatments and advanced materials for LFR environments.**
  - **Complete design & start construction of Pb Engineering Test Facility**
- **FY07**
  - **Complete preliminary selection of primary candidate materials for LFR system, including assessment of mechanical and corrosion properties of primary candidate LFR materials.**
  - **Preconceptual design viability evaluations including structural assessment, containment approach and transient/safety analysis.**
- **FY08**
  - **Establish reference cladding design and material specifications.**
  - **Complete construction of Pb Engineering Test Facility & start testing.**
  - **Preconceptual design viability evaluation including core refueling and transport approach and integrated system viability.**

Suggested subjects for university proposals

- **System Design & Evaluation**
  - **Measurement & modeling of 3-d natural circulation**
    - **Natural circulation flow stability experiments/models.**
    - **Heat transfer and flow pressure drop correlations.**
  - **Core Physics and Thermal-Hydraulics Design**
    - **Conduct core design studies, including neutronics and/or thermal-hydraulics analysis with emphasis on achieving long core life.**
    - **Evaluate core structural design to provide thermostructural reactivity feedback for passive safety and semi-autonomous load following.**
  - **LFR Energy Conversion Technology Studies**
    - **Evaluation of LFR coupling to super-critical CO_2 energy conversion.**
    - **Assess proposed technologies for conversion and utilization of heat from a small modular LFR.**

Suggested subjects for university proposals (cont.)

- **Materials**
  - **LFR Materials Design**
    - Identify new or modified materials for successful application in Pb and Pb-Bi-eutectic systems.
  - **Testing and Modeling**
    - Perform experiments to characterize candidate materials, including bulk and surface considerations, plus corrosion screening tests. Take advantage of collaboration on existing experimental capabilities, such as the LANL DELTA loop and/or the ANL Pb/LBE corrosion equipment.
    - Determine strategies for eliminating or relaxing oxygen control requirements for the compatibility of steels with Pb/LBE.
    - Test/model irradiation effects on LFR candidate materials.

Suggested subjects for university proposals (cont.)

- **Pb/LBE Coolant Technology**
  - **Oxygen monitoring and control strategies**
    - **Development of on-line oxygen sensors.**
    - **Demonstration of coolant chemistry control methods.**
    - **Development of real-time corrosion measurement methods for Pb/LBE.**
  - **Flow measurement techniques**
  - **In-service inspection techniques**

Suggested subjects for university proposals

- **Deployment & Institutional Issues**
  - **Market Assessment and Economic Analysis for Small Modular LFRs**
    - **Assess the market potential and requirements for small modular reactors, with consideration of global energy demand and supply, evaluate economic requirements.**
    - **Evaluation of the potential for economies of mass production, rapid deployment, and simple operation - in lieu of the 'economy of scale'.**
    - **Systems evaluation of how small, modular, long-core-life fast reactors fit within possible future nuclear energy growth scenarios.**
  - **Proliferation-resistant Design and Nonproliferation Assessment of Small Modular LFRs**
    - **Evaluate the proliferation-resistant features of possible LFR designs that are intended to incorporate additional proliferation resistance.**

Generation IV LFR for FY03, FY04 and FY05

FY 2003 Appropriation: **$2,200k**

FY 2004 Appropriation: **1,000k**

FY 2005 Appropriation: **1,000k**

Functional Area	Tasks	Performers	FY 2003	FY 2004	FY 2005
System design and evaluation	Reactor Design	ANL	619	203	200
		LLNL	480	226	170
		UC-Berkeley	70		
		Tx A&M	50		
Materials	DELTA loop operation and experimentation	LANL	396	367	229
	Corrosion screening tests	ANL	215	114	93
	Materials requirements definition	LLNL	70	90	240
	Project management & reserve	LLNL	250		
		LANL	50		
		INL			68
	Total		2200	1000	1000

FY 2006 Planning: **1,300k**

FY 2007-10: **TBD**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Kevan D. Weaver

Gas-Cooled Fast Reactor (GFR)

INL

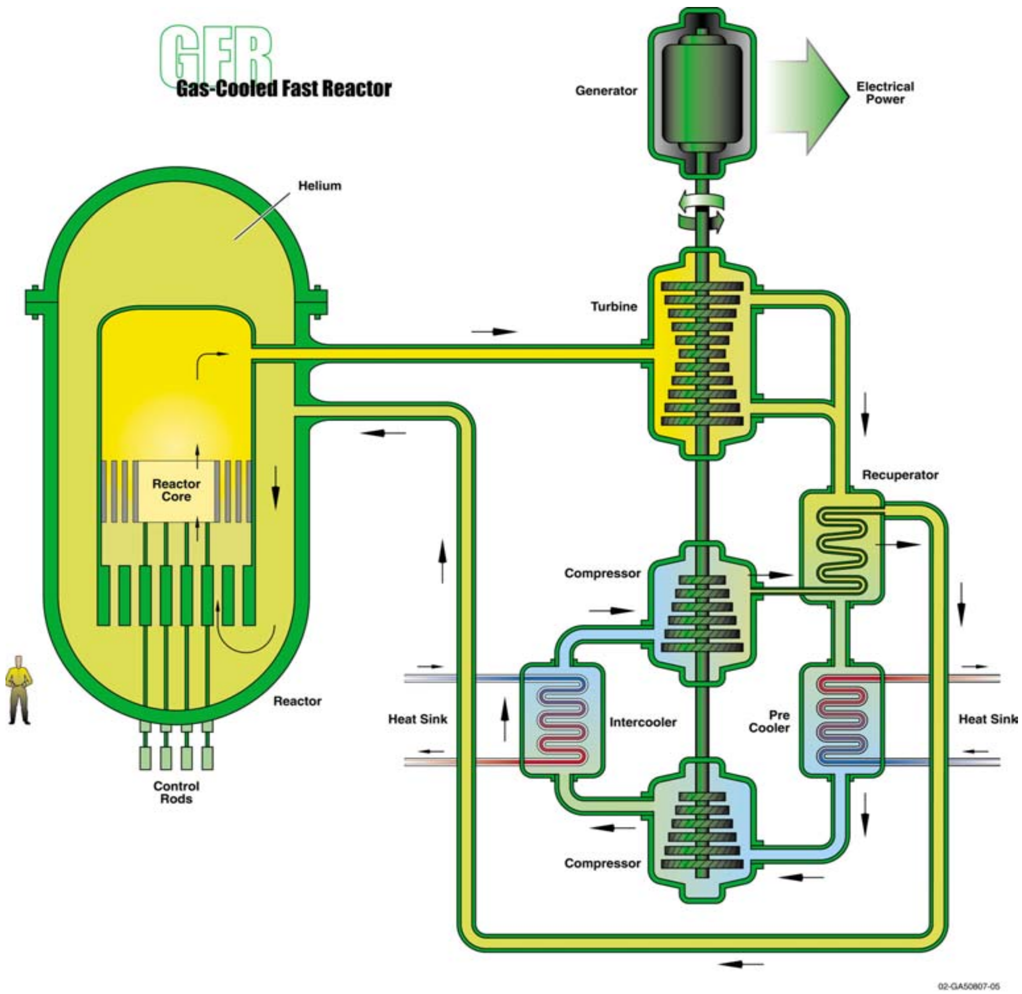
Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Demonstrating technical feasibility of the GFR**
- **The GFR project is part of the Generation-IV program**
- **The Generation-IV program calls for the development of:**
  - the next generation of nuclear systems for production of high-value energy products such as electricity and hydrogen, and
  - development of fast reactor systems for sustainability (including actinide/waste management)

GFR Objectives

- High level of safety
- High sustainability with a closed fuel cycle and full TRU recycle
- Fast- spectrum core
- Direct Brayton cycle, high-efficiency energy conversion
- Production of H_2
- Estimated deployment time: 2025



Near Term GFR Projects

- **GFR Design and Safety**
  - Define GFR reference design features (fuel technology, coolant, unit power) and operating parameters (power density, temperatures)
  - Identify safety systems capable of decay heat removal

- **GFR Fuels, In-Core Materials, and Fuel Cycle Processes**
  - Identify fuels and core materials capable of high temperature operations, high fission product confinement, and reasonable burnup/fluence
  - Identify and test fuel treatment and refabrication processes

GFR FY05 Tasks

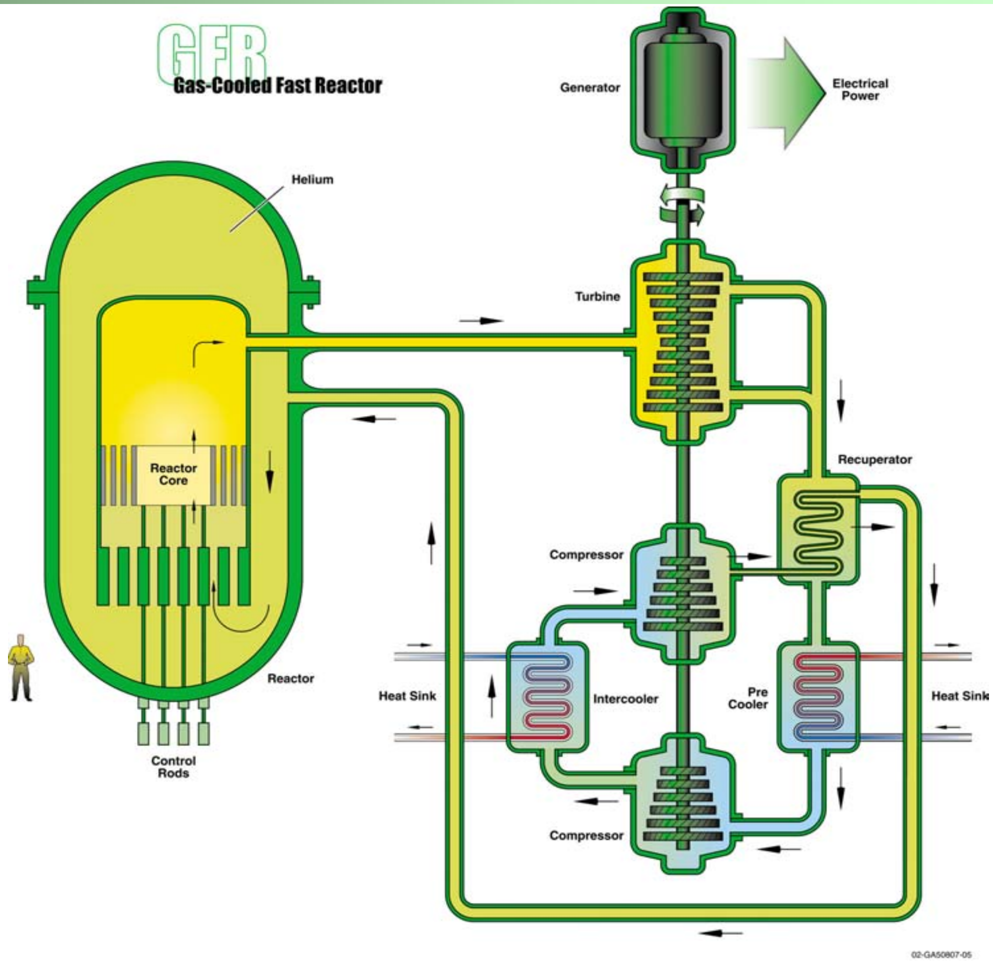
- System modeling and analysis for 600MWt design
- T-H calculations to optimize pressure conditions (low pressure drop core)
- Core neutronic calculations
- Analyze depressurization accident
- Continue ODS joining experiments
- Ion irradiation and characterization of ceramic materials
- Fabricate material samples for FUTURIX-MI irradiations

Relationship to NE Program Priorities

- The gas-cooled fast reactor (GFR) was chosen as one of the Generation IV nuclear reactor systems to be developed based on its excellent potential for sustainability through reduction of the volume and radiotoxicity of both its own fuel and other spent nuclear fuel, and for extending/utilizing uranium resources orders of magnitude beyond what the current open fuel cycle can realize
- Viability research is being performed to determine whether the GFR can meet the Gen IV goals

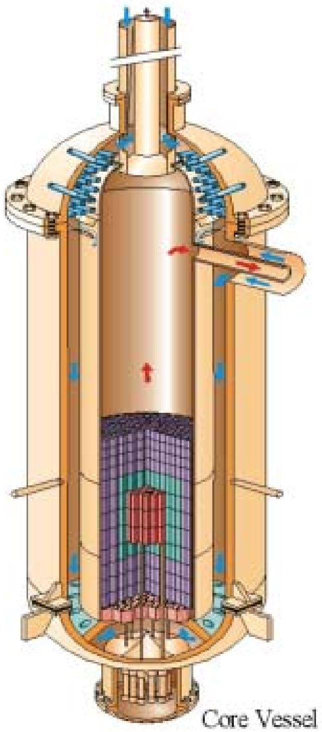
GFR Design Options

- **Reference design features:**
  - Coolant: He (at 850ºC and 7MPa)
  - Direct Brayton cycle
  - Unit power: 600-2400 MW_{th}
  - Power density: 50-100 MW/m^3
- **Option 1 design features:**
  - Coolant: He (at 600-650ºC and 7MPa)
  - Indirect Brayton cycle (with $S\text{-}CO_2$ on the secondary side)
  - Unit power: 600-2400 MW_{th}
  - Power density: 50-100 MW/m^3
- **Option 2 design features:**
  - Coolant: $S\text{-}CO_2$ (at 550ºC and 20MPa)
  - Direct Brayton cycle
  - Unit power: 600-2400 MW_{th}
  - Power density: 50-100 MW/m^3



System Design and Safety

- **Perform thermal-hydraulic and physics studies for candidate designs**
- **Studies for DHR in case of depressurization with loss of offsite power in a GFR with 50 –100 MW/m^3**
  - Heat storage
  - In core conduction and vessel radiation
  - Forced circulation
  - Natural circulation
  - Heavy gas injection
  - Other?

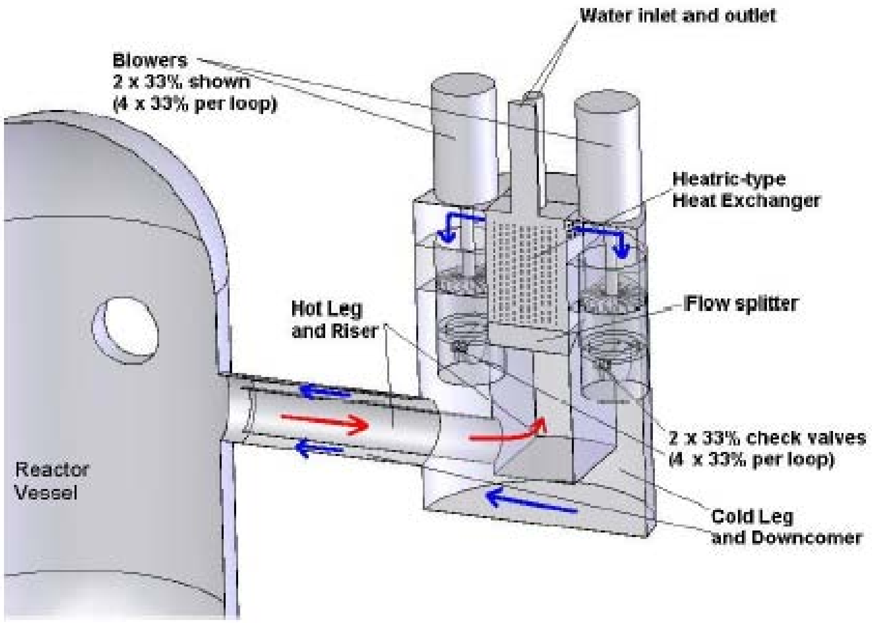


GFR Design and Safety Work

- **Heat storage, in core conduction, and vessel radiation**
  - Combination of these three alone cannot effectively remove decay heat during an accident
- **Forced circulation**
  - Very efficient – 3% nominal flow enables core cooling while fulfilling fuel temperature criteria
  - Circulators of a very limited power (100 KW) meet the requirements
- **Helium natural convection**
  - Sufficient with a top mounted HX at nominal pressure
  - Requires a significant back pressure in case of depressurization (function of MW/m^3, HX elevation, core ΔP,..)
- **Heavy gas injection**
  - Increase of temperature of the injected gas
  - Enhancement of natural circulation when compared with helium

GFR Design and Safety Work

- **Combined safety system**
  - Accident initiates a blower (16KW); only needed for first 24 hours
  - Minimal back pressure needed (≤5 bar)
  - Natural convection takes over after 24 hours
- **Conclusions**
  - Helium natural circulation is not suitable alone for high power density (high back pressure needed)
  - Heavy gas injection presents perspectives for effective cooling at the beginning of the transient, and would provide for easier natural circulation



Proposed University Contributions to GFR System Design and Safety

- **FY06-FY08**
  - Thermal-hydraulic analysis and design, and safety system design (i.e., model development and transient analysis using active, passive, or combined DHR systems)
  - PRA studies to determine the best safety systems (or combination of systems) that satisfy Gen IV goals in safety
  - Neutronic/physics core design, including analysis of reactivity coefficients during accident conditions, reactivity limited burnup, etc.

Material Requirements

- **In-core structures**

Key point: in-service mechanical integrity under prolonged irradiation at high temperature imply development of new innovative material solutions that are, today, not yet proven

Adequate initial & in-pile following characteristics :

- Physical properties (e.g., heat capacity, heat transfer coefficient, thermal expansion), neutronic transparency, chemical compatibility with He (and impurities) & actinide compounds and resistance to gas permeability,
- Tensile, creep, fatigue, and toughness properties.
- Microstructure and phase stability
- Irradiation creep, in-pile creep and swelling resistances,
- Mechanical & chemical stabilities in LOCA transients and air ingress conditions

Ability to weld/join and fabricate components at a reasonable cost

- **Out-of-core structures**

  - Fabricability & welding capabilities on thick products
  - Adequate tensile, creep, fatigue, and toughness properties

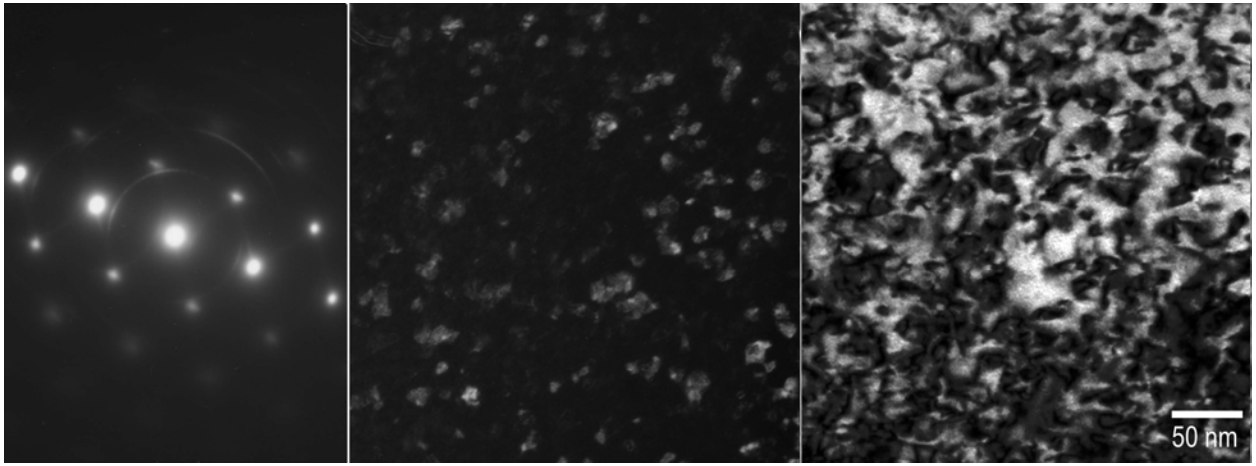
GFR fuel matrix and structural material reference requirements

Requirement	Reference Value
Melting/decomposition temperature	>2000°C
Radiation induced swelling	< 2% over service life
Fracture toughness	> 12 MPa $m^{1/2}$
Thermal conductivity	> 10 W/mK
Neutronic properties	Materials allow low core heavy metal inventory and maintain good safety parameters

Candidate ceramic matrix materials: SiC, ZrC, TiC, ZrN, TiN, and AlN

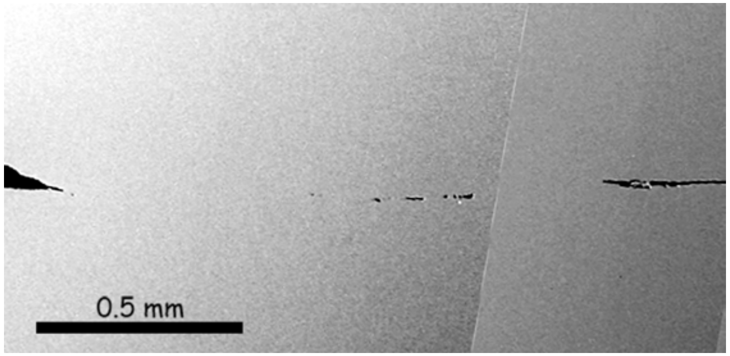
GFR In-Core Materials

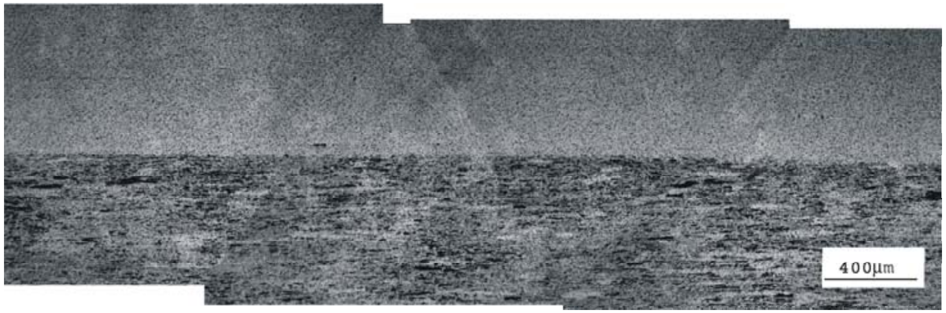
- **The following alloys are being considered:**
  - Alloy 800H, ferritic steel T-122, and oxide dispersion strengthened (ODS) alloys MA957, MA754, and PM2000
- **Neutron irradiation of ceramics**
- **Ion-beam (Kr) irradiation of ceramics**
  - Zr-based ceramics show significant swelling



GFR In-Core Materials

- **Welding/joining studies of ODS materials**
  - Both resistance pressure welding (RPW), and transient liquid phase (TLP) bonding studies were performed
- **RPW studies using fine grained PM2000 indicated full bonding in apparent bonded areas**
- **TLP bonding has been used to successfully join MA956 in the longitudinal direction**

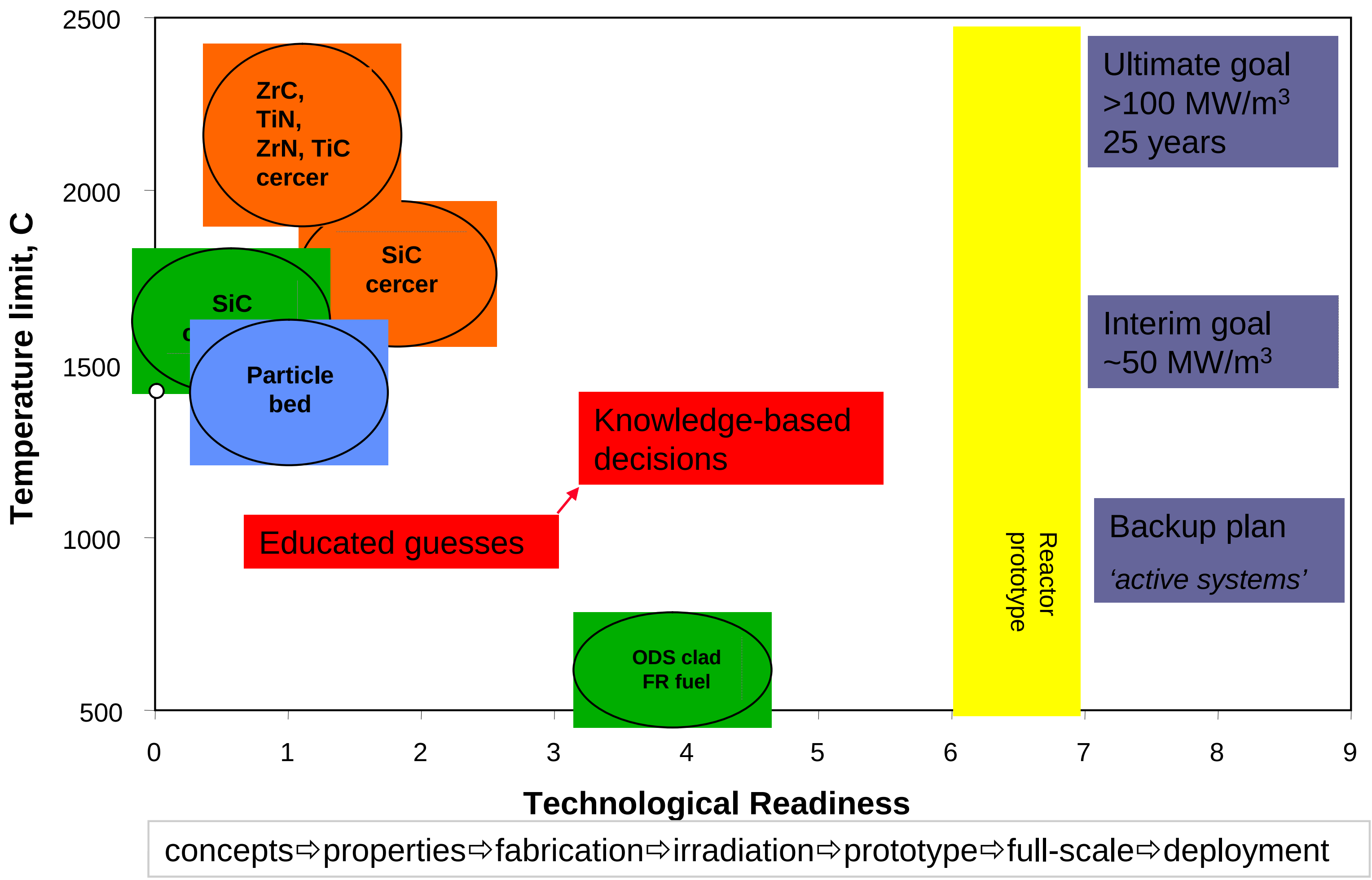




Proposed University Contributions to GFR Materials Development

- **FY06-FY08**
  - Measure missing thermo-mechanical/physical properties for those ceramics of interest (e.g., carbides and nitrides)
  - Joining/welding studies of candidate materials (both ceramic and metallic)
  - Supercritical CO_2 corrosion studies on materials of interest (both ceramic and metallic)
  - In-pile or accelerator irradiations of candidate materials

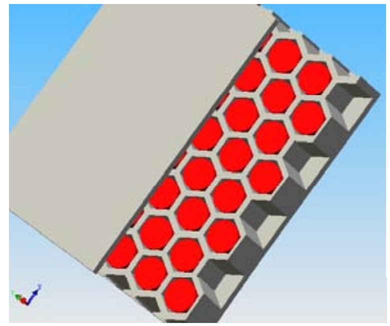
Current GFR Fuel Status

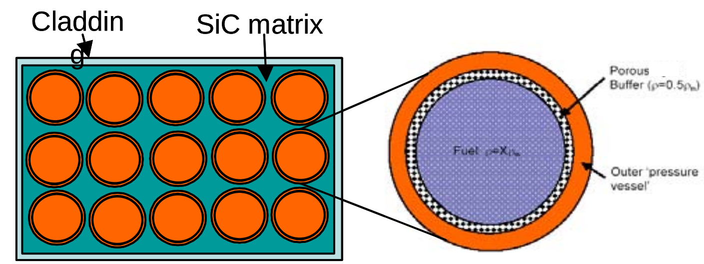


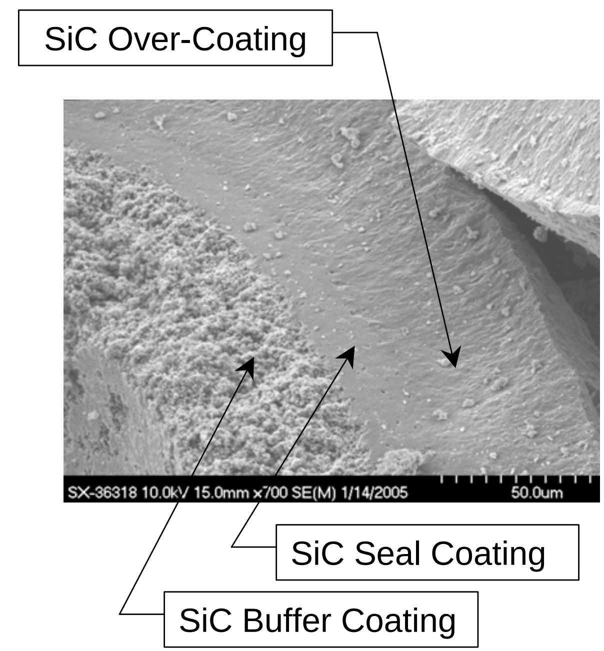
GFR Fuel

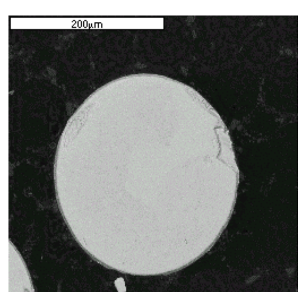
Dispersion fuel

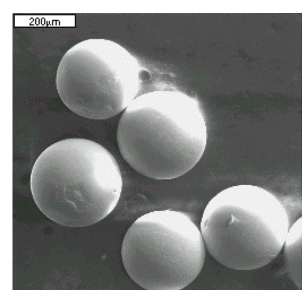
- Fuel form: fibers or particles in plates or inert matrix
- Initial matrix material selection: SiC
- Fuel composition: UC or UN (will also include Pu+MA)
- Initial fuel loading goal: 50% fuel & 50% matrix
- Fuel loading objective 70% fuel & 30% matrix
- Rationale: Best from neutronics standpoint and will withstand high temperatures during accidents





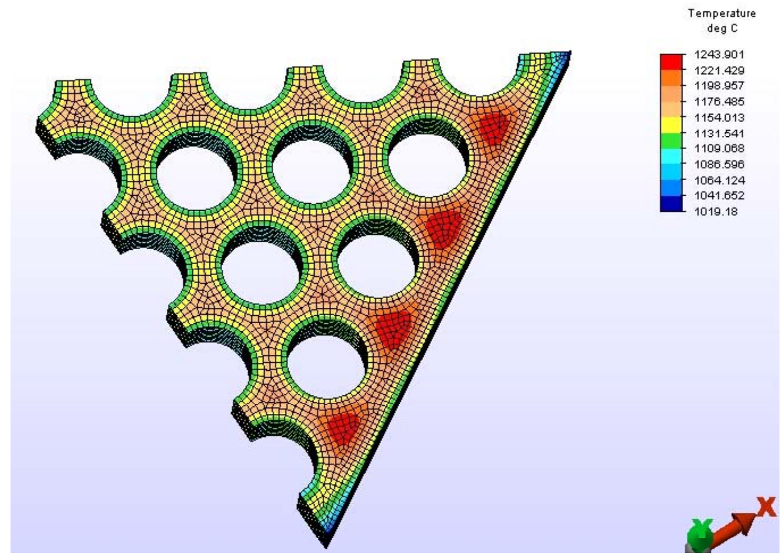


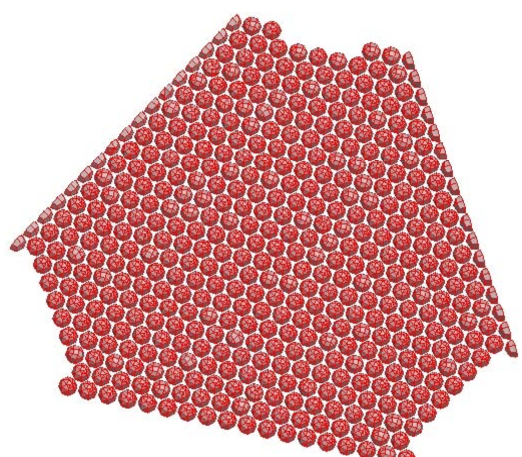


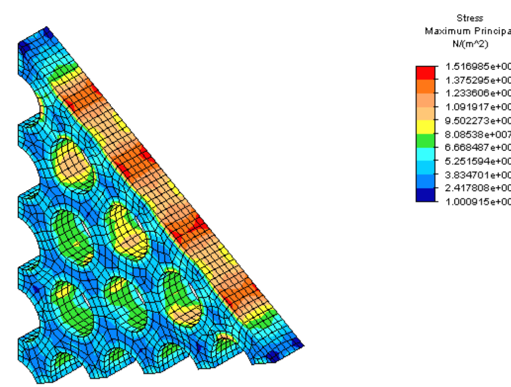


GFR Fuel Modeling

- **Finite element fuel models have been constructed for local and global temperature and stress profiles**
  - Peak fuel temperature of ~1250°C
  - Peak stress of ~150MPa







Load Case: 1 of 1
Maximum Value: 1.51698e+008 N/(m^2)
Minimum Value: 1.00091e+007 N/(m^2)

Proposed University Contributions to GFR Fuels Development

- **FY06-FY08**
  - Innovative matrix material fabrication techniques for ceramics of interest
  - Fuel performance modeling using UC and UN in ceramic matrices (specifically thermo-chemical)
  - Preliminary assessment of the GFR fuel cycle (includes flow sheet development, physics/neutronics analysis of equilibrium cycle, and possible surrogate material experiments)

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

JOHN H. KOLTS

NUCLEAR HYDROGEN

THERMOCHEMICAL CYCLES

IDAHO NATIONAL LABORATORY

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- **Sulfur – Iodine Thermochemical Cycle**
  - Laboratory scale process testing
  - Membrane development
  - Catalyst development
  - Engineering and process optimization
- **Hybrid Sulfur Cycle**
- **Alternative Thermochemical Cycles**
  - Calcium – Bromine Cycle
  - New thermochemical cycles

Sulfur – Iodine Thermochemical Cycle

- $2HI \longrightarrow H_2 + I_2$

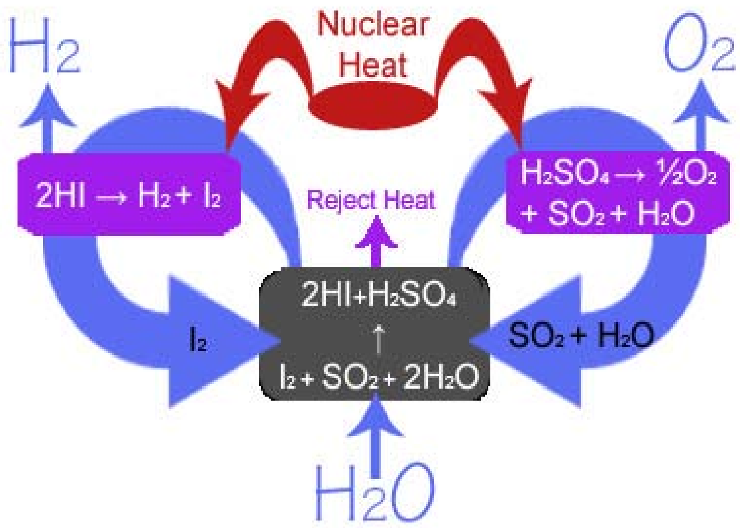
  General Atomics

- $I_2 + SO_2 + 2H_2O \longrightarrow$
  $2HI + H_2SO_4$

  CEA France

- $H_2SO_4 \longrightarrow 0.5O_2 + SO_2 + H_2O$

  Sandia



Sulfur – Iodine Thermochemical Cycle

FY04

- Completed Phase I Nuclear Hydrogen Systems configuration study
- Completed design of the integrated laboratory scale sulfur-iodine process
- Completed design and flowsheet analysis for reactive and extractive decomposition of HI
- Scoping of catalysts, membranes and materials for sulfur-iodine cycle

Sulfur – Iodine Thermochemical Cycle

FY05

- Fabricated and initiated testing of reactive and extractive decomposition of HI to iodine and hydrogen
- Fabricated, in metal components, sulfuric acid decomposition reactor and started testing
- Continued efforts with CEA, France on fabrication and testing of Bunsen reactor ($SO_2 + I_2 + 2H_2O \rightarrow H_2SO_4 + 2HI$)
- Initiated comprehensive materials testing program
- Initiated heat exchanger design, modeling and testing program
- Continued catalyst and initiated membrane test programs

Sulfur – Iodine Thermochemical Cycle

FY06 – FY08

- Complete construction of integrated laboratory scale S-I loop components
- Ship individual S-I laboratory scale loops to INL, assemble and begin testing of fully integrated system
- Complete high temperature, long term, physical property tests on HI and H_2SO_4 compatible materials
- Complete demonstration tests for high temperature membranes to shift SO_3 decomposition equilibrium
- Complete SO_3 decomposition catalyst efforts
- Complete HI decomposition catalyst efforts
- Complete membrane separation efforts for removal of water from iodine solutions

Sulfur – Iodine Thermochemical Cycle

$$2HI \longrightarrow H_2 + I_2 \quad 300 \text{ to } 400° \quad C$$

Research:

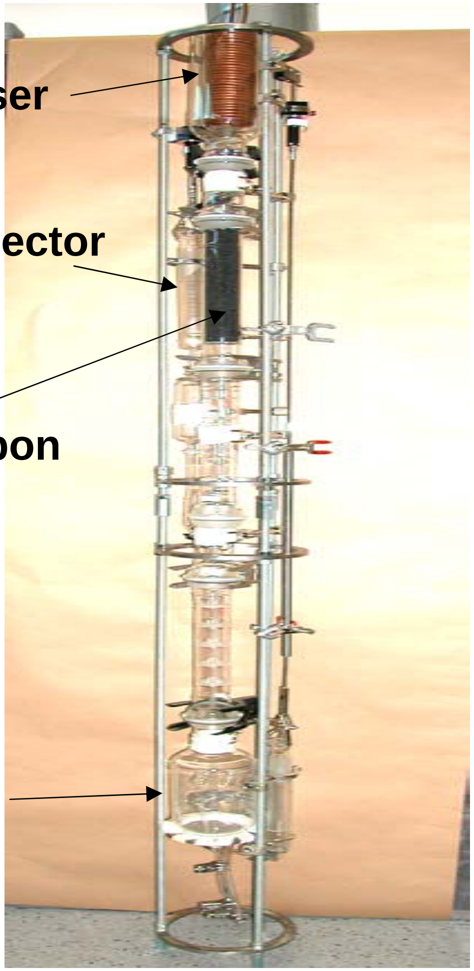
- **Removal of excess water (membrane)**
- **Slow conversion kinetics (better catalyst)**
- **Phase data for $HI/I_2/H_2O$ ternary system**

Sulfur – Iodine Thermochemical Cycle

$$2HI \longrightarrow H_2 + I_2$$

- **Catalyst deactivates rapidly**
- **Unknown data for ternary mixture**

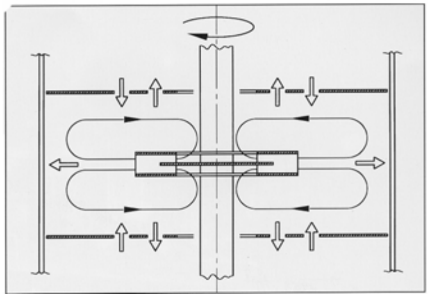
Reactive distillation



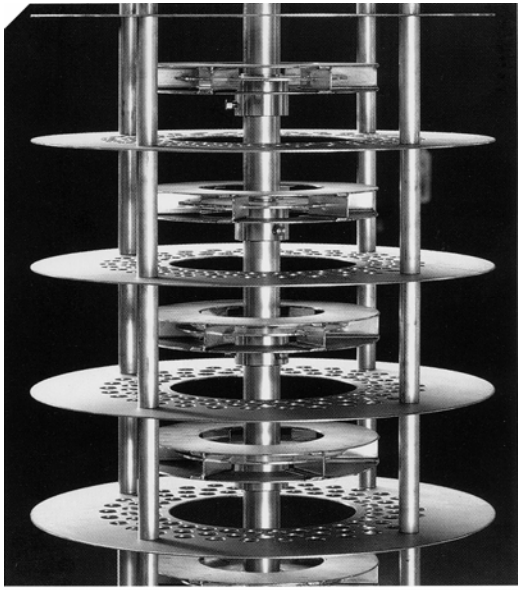
Sulfur – Iodine Thermochemical Cycle

Column part
DN150 and DN30

Bunsen Reactor



Mixing cell

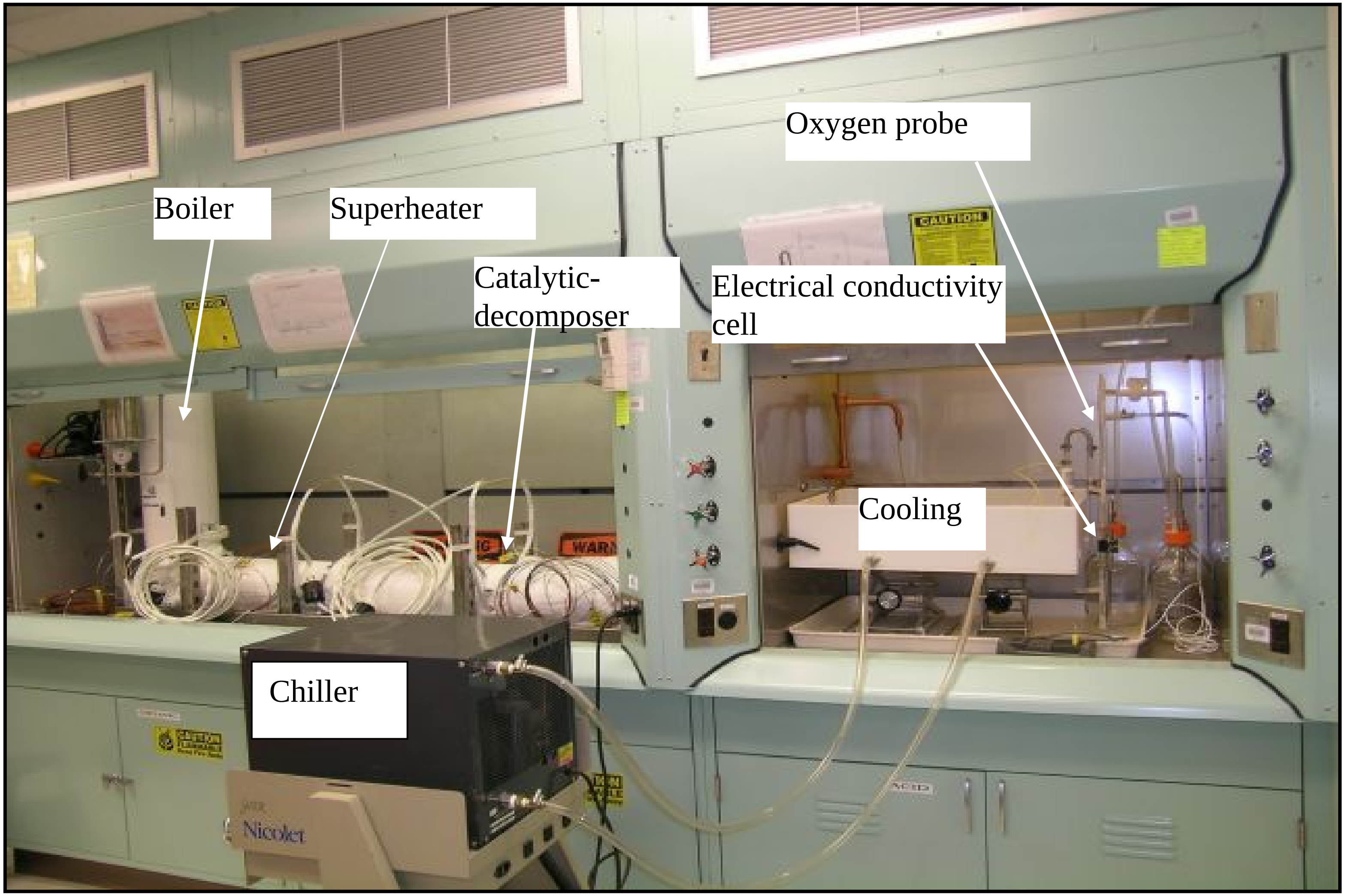


Internals

$$I_2 + SO_2 + 2H_2O \longrightarrow 2HI + H_2SO_4$$

- Temperature control – sulfur
- SO_2, O_2 separation prior to reactor

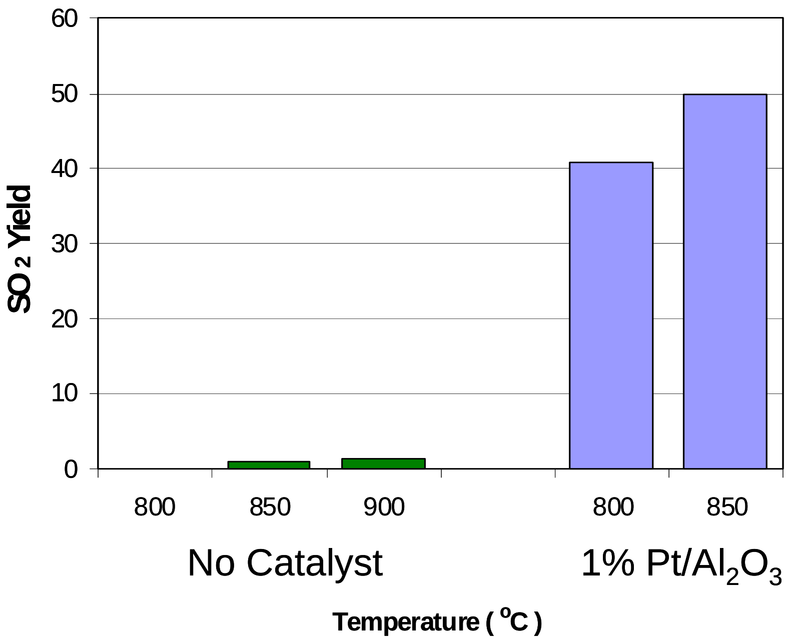
Sulfur – Iodine Thermochemical Cycle – H_2SO_4 Decomposition

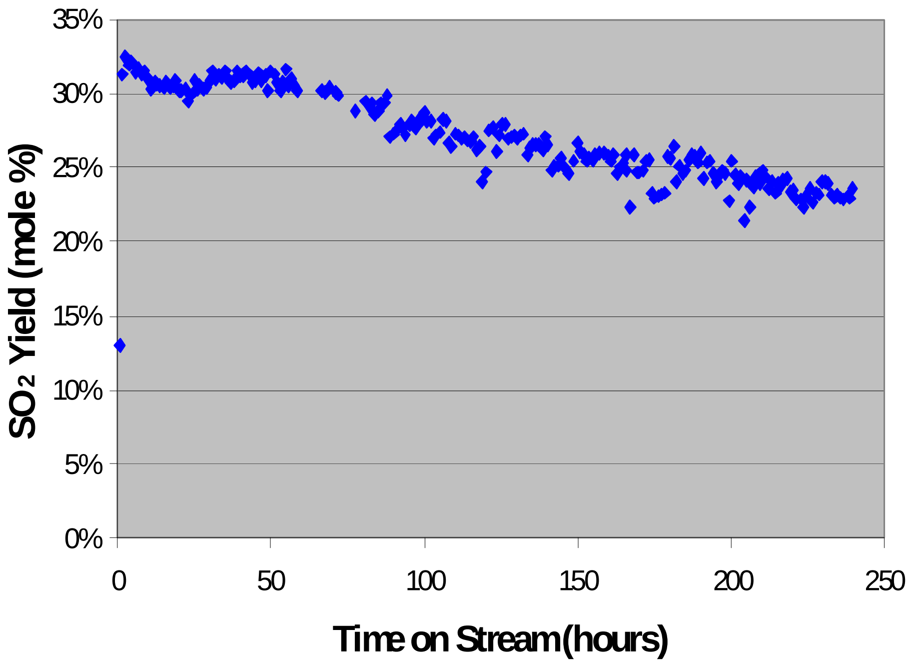


Sulfur-Iodine Thermochemical Cycle

$$H_2SO_4 \longrightarrow 0.5O_2 + SO_2 + H_2O$$

- Slow kinetics – Need active long lived catalyst





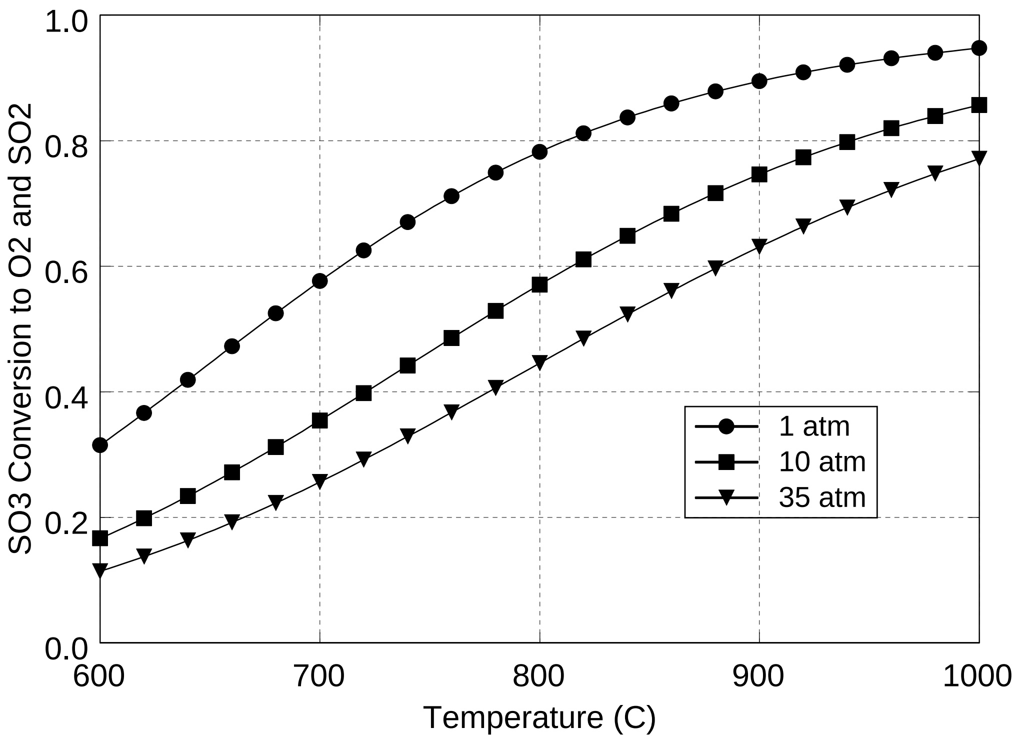
0.1% Pt on Ti02

Sulfur-Iodine Thermochemical Cycle

$$H_2SO_4 \longrightarrow O.5O_2 + SO_2 + H_2O$$

High temperature reaction

- **Use high temperature membrane to remove O_2 and shift equilibrium**

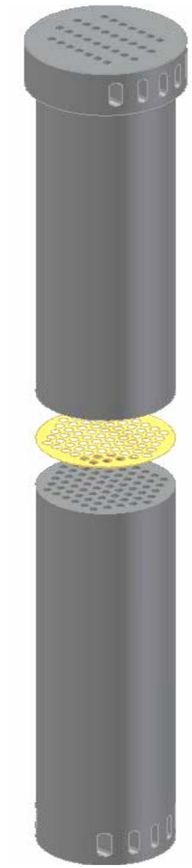


Sulfur-Iodine Thermochemical Cycle

$H_2SO_4 \longrightarrow 0.5O_2 + SO_2 + H_2O$

Materials: Must have stable high temperature materials

Sandia Reactor after testing



Gold seal

SiC

Japan is using SiC heat exchanger/ reactor for H_2SO_4 decomposition

Hybrid Sulfur Cycle

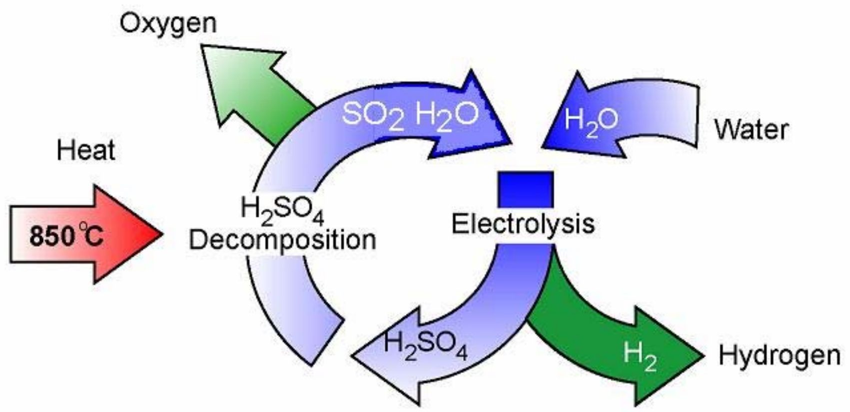
FY05

- **Complete conceptual design of hybrid sulfur system**
- **Conducting ambient pressure testing of H_2SO_3 electrolysis**

FY06-08

- **Conduct optimization tests with single cell electrolyzer for hybrid sulfur cycle**
- **Conduct and test lab-scale multi-cell electrolyzer**

Hybrid Sulfur Cycle



- Same H_2SO_4 decomposition step as sulfur- iodine cycle
- Eliminates all iodine reactions
- Does require both electricity and heat

- **Optimization of anode and cathode life**
- **Very stable long lived diffusion membrane**
- **Optimum cell design for high performance**

Alternative Cycles – Calcium-Bromine Cycle

FY04

- Completed initial analysis of calcium-bromine cycle

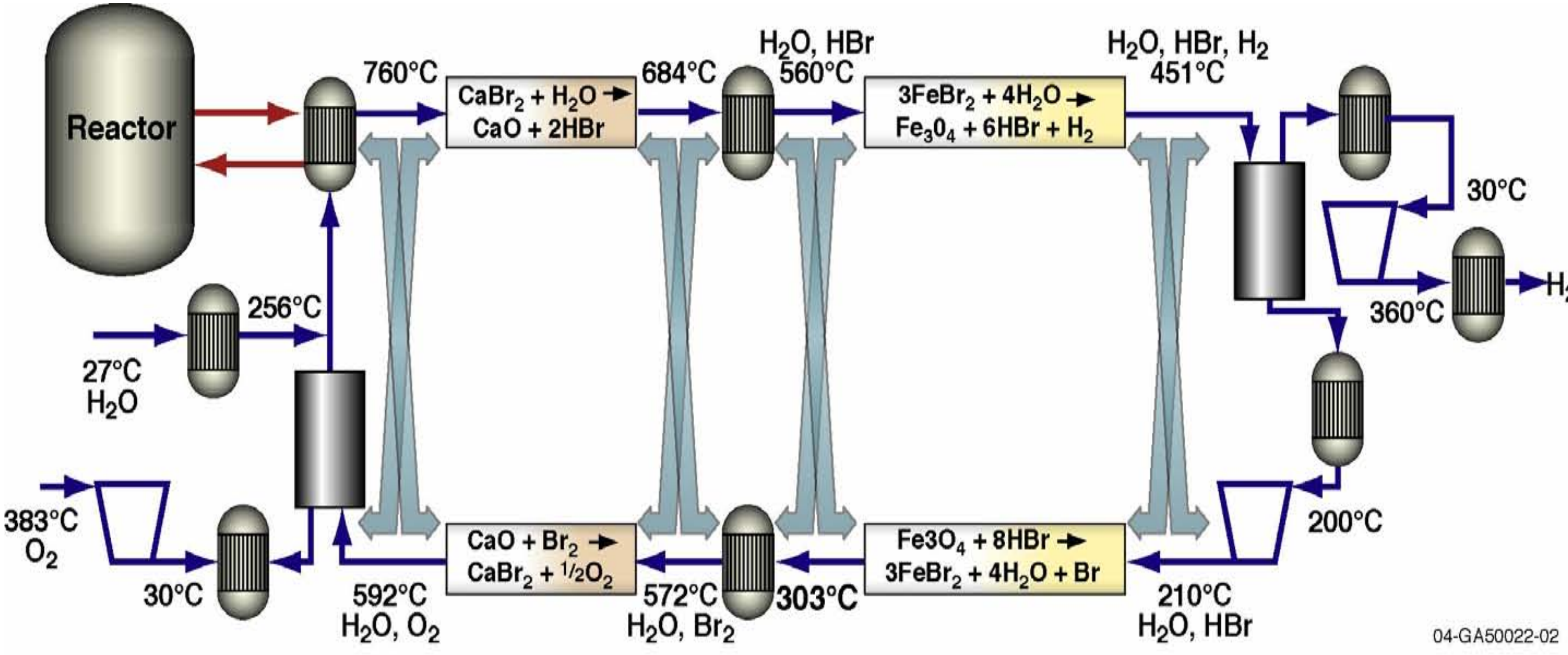
FY05

- Conducting analysis and design of cold plasma dissociation of HBr
- Evaluating feasibility of using molten spray contactors to stabilize Ca on support surface

FY06-08

- Complete testing required to determine feasibility and performance of Ca-Br cycle

Alternative Cycles – Calcium/Bromine Cycle



- **Decomposition of HBr**
- **Stabilization of Ca when cycling between $CaBr_2$ and CaO (79% volume change)**
- **Efficient separation of water from HBr**

Alternative Thermochemical Cycles

Needs:

- Best possible confirmation that no other cycle is more efficient than sulfur-iodine
- Sound analytic methods of evaluating competing thermochemical cycles
- Physical parameters: temperature, materials, phase changes, volume changes, competing reactions
- Laboratory tests to support new thermochemical cycles

BUDGET INFORMATION (1)

FY 2004 Funded Projects

♦ Sandia	H_2SO_4 decomposition	301K
♦ Argonne	Ca-Br cycle	87K
♦ General Atomics	HI decomposition	426K

BUDGET INFORMATION (2)

- **FY05 funded projects**
  - **Total funding: $2,840K**

Sandia	H2SO4 decomposition	650K
Argonne	SO3 electrolysis	40K
Argonne	Ca-Br cycles	180K
Argonne	Alternative TC cycles	150K
General Atomics	HI Decomposition	700K
INL	Acid conc Membranes	190K
INL	Acid decomp catalysts	220K
ORNL	High temp membranes	170K
ORNL	Materials support	50K
SRNL	Hybrid Sulfur	300K

- **2 U-NERI Projects** — Clemson University; Johns Hopkins

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Steve Herring

Nuclear Hydrogen

High Temperature Electrolysis

Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

Structure of the Work

1. **HTE System Definition**

   Plant conceptual design [INL]

   CFD and Plant Modeling [INL and ANL]

   Athabasca oil-sand upgrading [I-NERI, INL,ANL with AECL]

2. **HTE Experiments**

   Button Cell and Stack fabrication [Ceramatec, Inc., SLC]

   Advanced electrodes and electrolytes [ANL]

   HTE test stand operation [INL]

   Plasma deposition of cells [INL]

   High temperature H_2/H_2O membrane separations [ORNL]

High Temperature Electrolysis Overview

Technical Area Objectives (FY05)

- ♦ **Develop and demonstrate energy-efficient, high-temperature, solid-oxide electrolysis cells (SOECs) and stacks for hydrogen production from steam**
- ♦ **Demonstrate technology at progressively larger scales**
- ♦ **Perform flowsheet analyses of systems-level HTE processes to support planned scale-up to Integrated Laboratory Scale, Pilot-Scale and Engineering Demonstration Scale**
- ♦ **Develop detailed CFD models of operating SOECs; validate with experimental data**
- ♦ **Investigate alternate cell materials (e.g., alternate electrode materials), alternate cell configurations (e.g., porous-metal substrates, Tuff Cell), and applications of inorganic membranes**

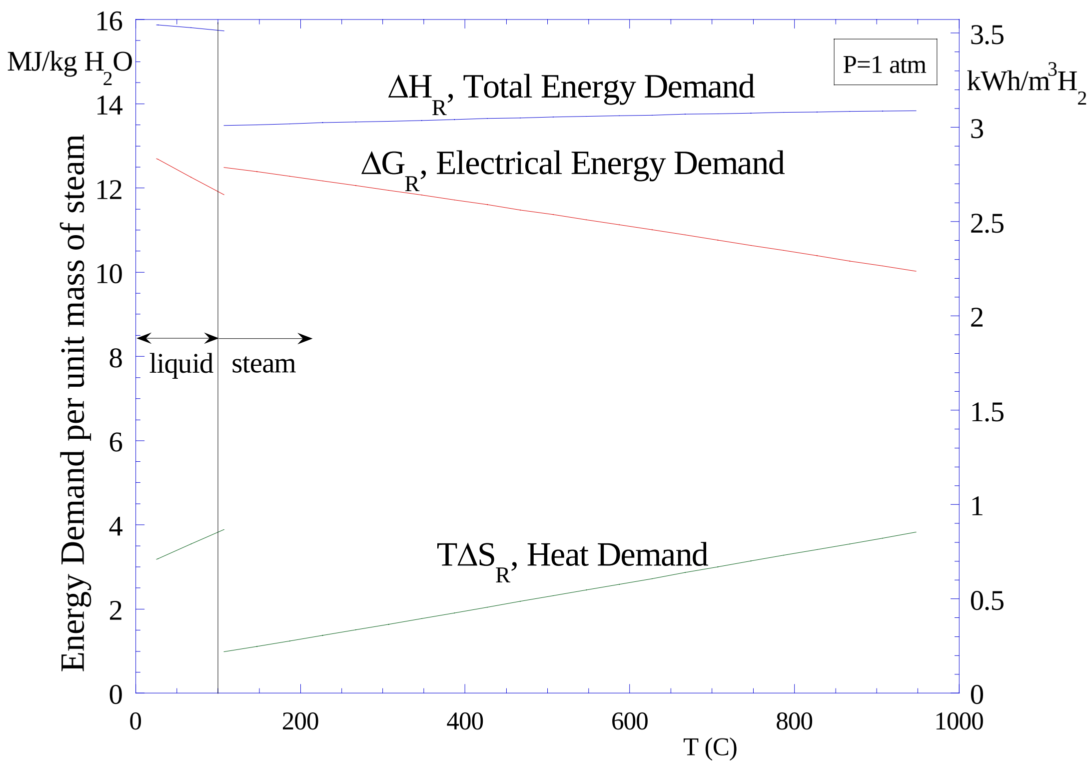
High Temperature Electrolysis Overview

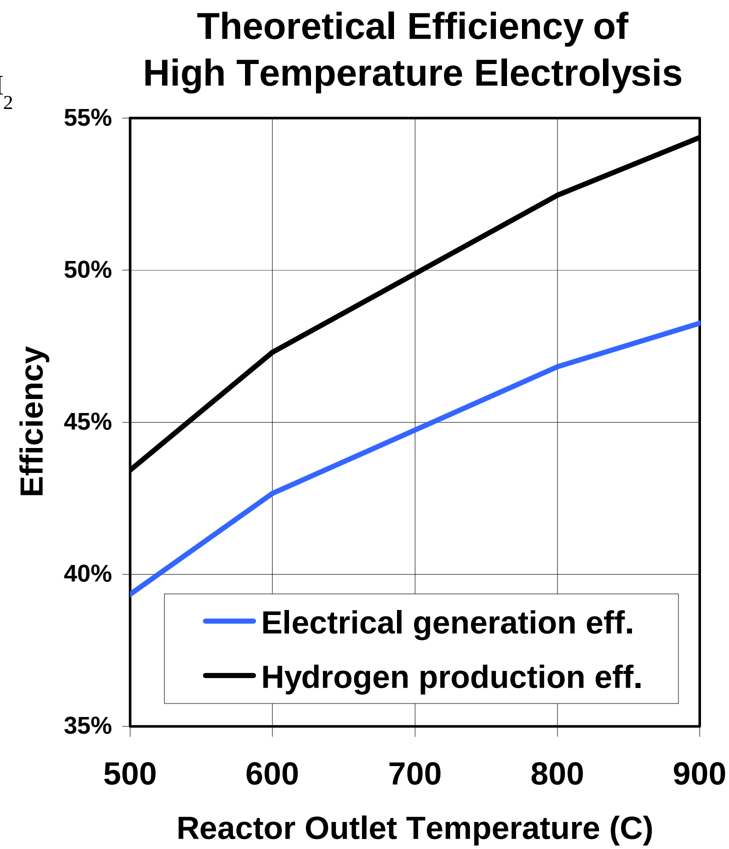
HTE

Key Milestones (Level 2)

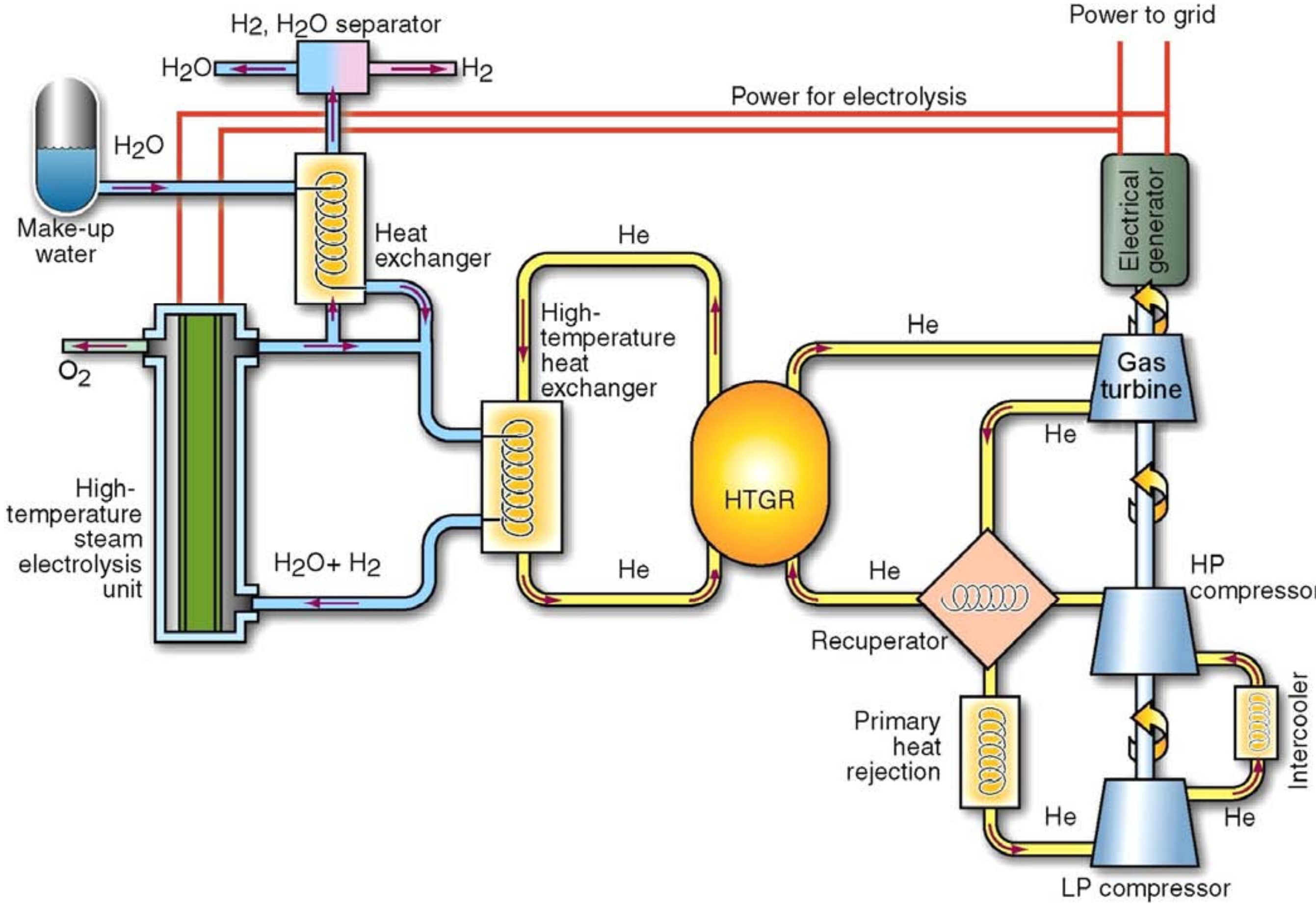
- Demonstrate high-temperature electrolysis stack testing at a production rate of 50 normal liters per hour of hydrogen. [INL, 12/31/04]
- Develop engineering process model for HTE system performance evaluation. [ID, 5/17/05]
- Demonstrate high-temperature electrolysis stack testing at a production rate of 100 normal liters per hour of hydrogen. [INL, 8/1/05]
- Develop conceptual design documentation for the 200 kW high-temperature electrolysis pilot-scale experimen.t [INL, 8/15/05]
- Complete Annual Report of CFD and Flowsheet Analyses of High Temperature Electrolysis Plant. [ANL, 9/15/05]
- Complete analyses of membrane applications to High Temperature Electrolysis. [ORNL, 9/1/05]
- Successful fabrication of 15 button cells based on the INL porous-metal substrate design. [INL, 1/15/05]

Energy Input to Electrolyser

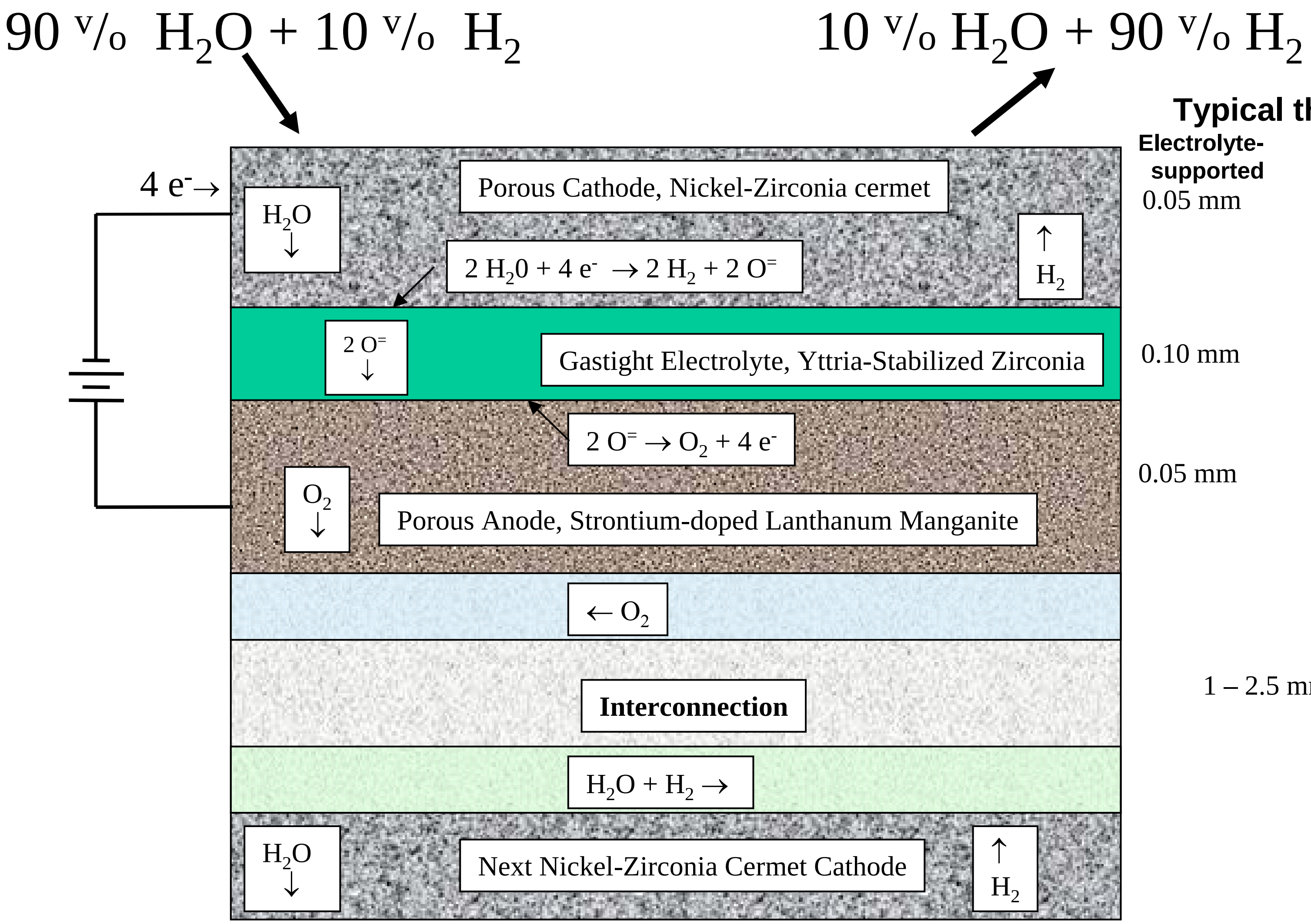




High Temperature Electrolysis Plant



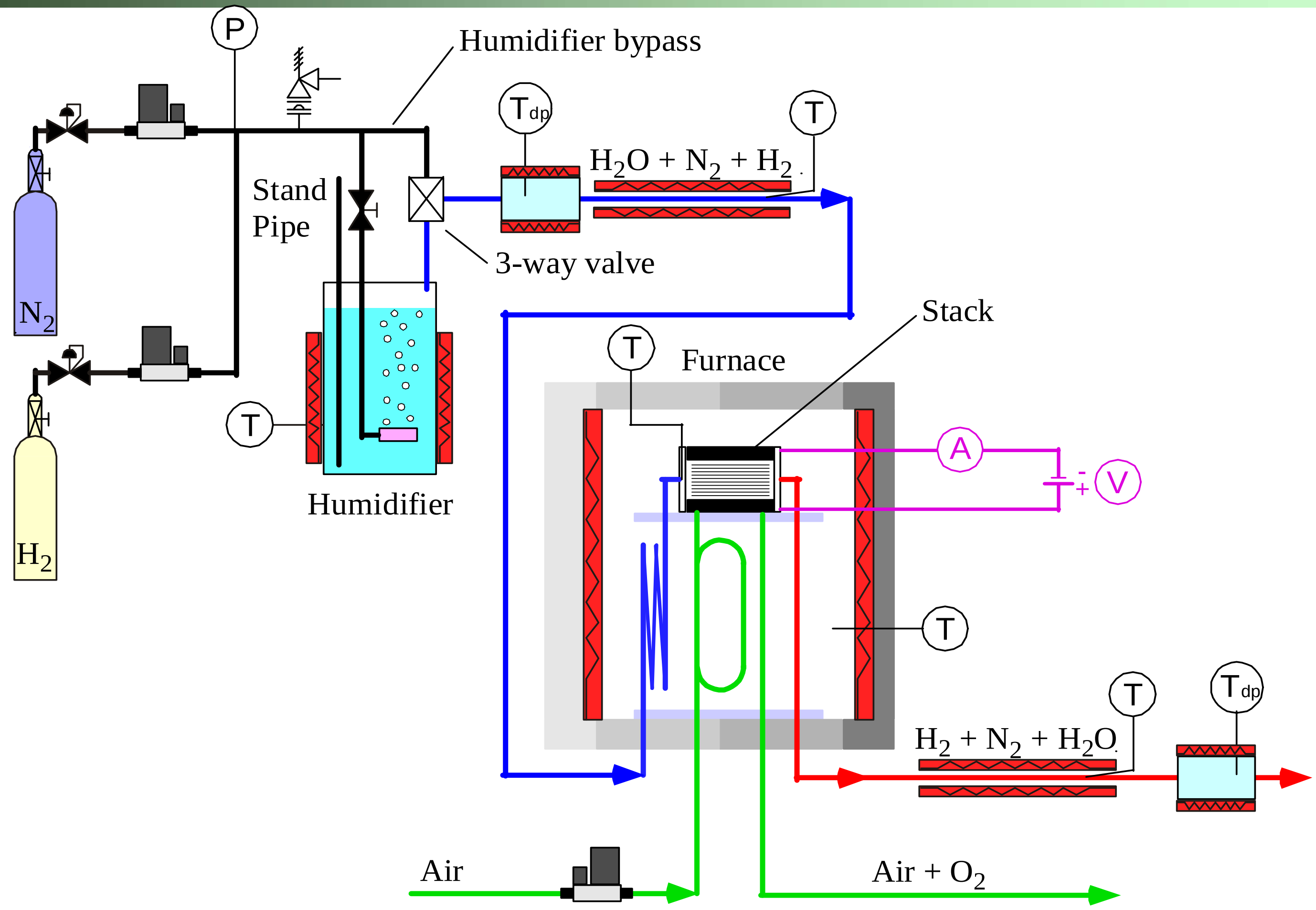
Electrolysis Cell Cross Section



Typical thicknesses

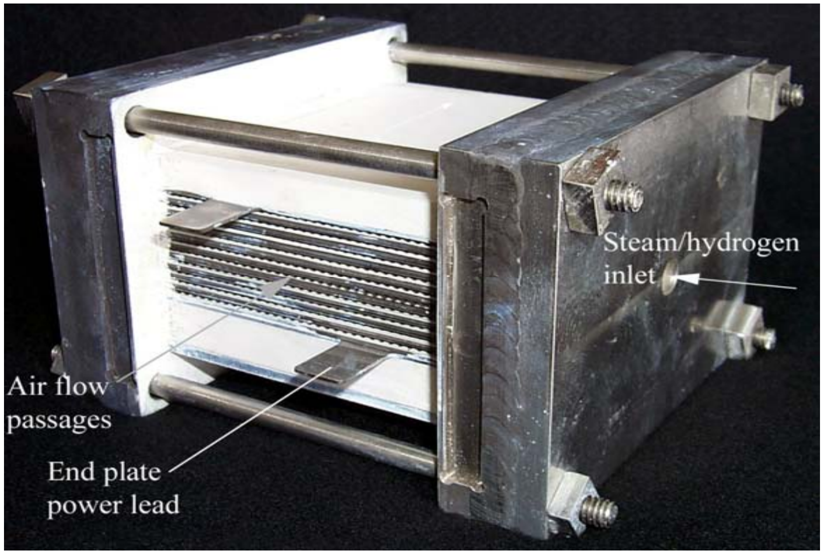
Electrolyte-supported	Cathode-supported
0.05 mm	1.500 mm
0.10 mm	0.01 mm
0.05 mm	0.05 mm
1 – 2.5 mm	

Schematic of Stack Testing Apparatus



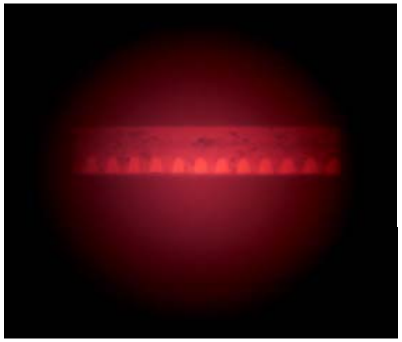
Electrolysis Stack Performance Testing Hardware

10-cell electrolysis stack; has produced up to 100 SLPH H_2

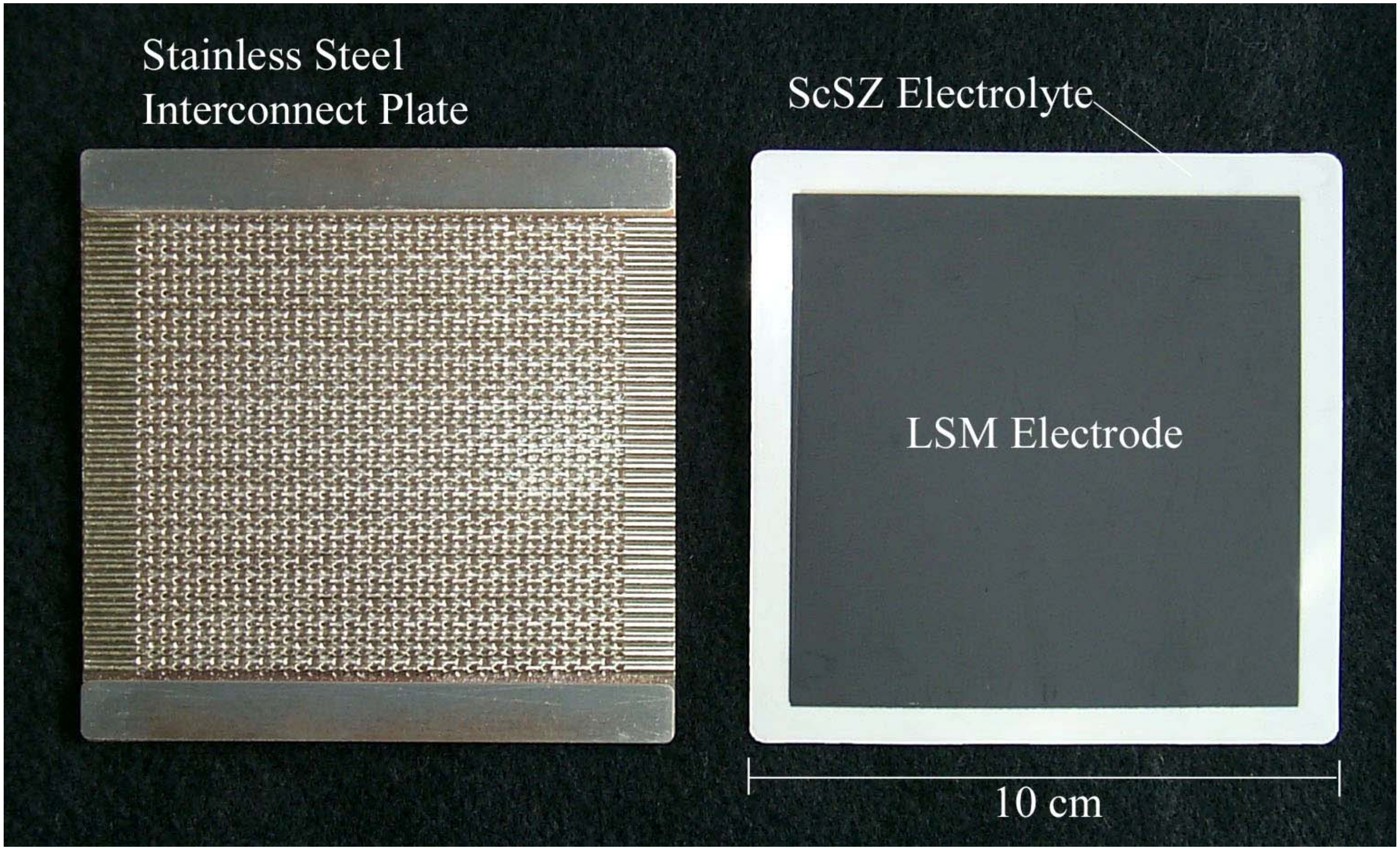


Stack mounted on test fixture

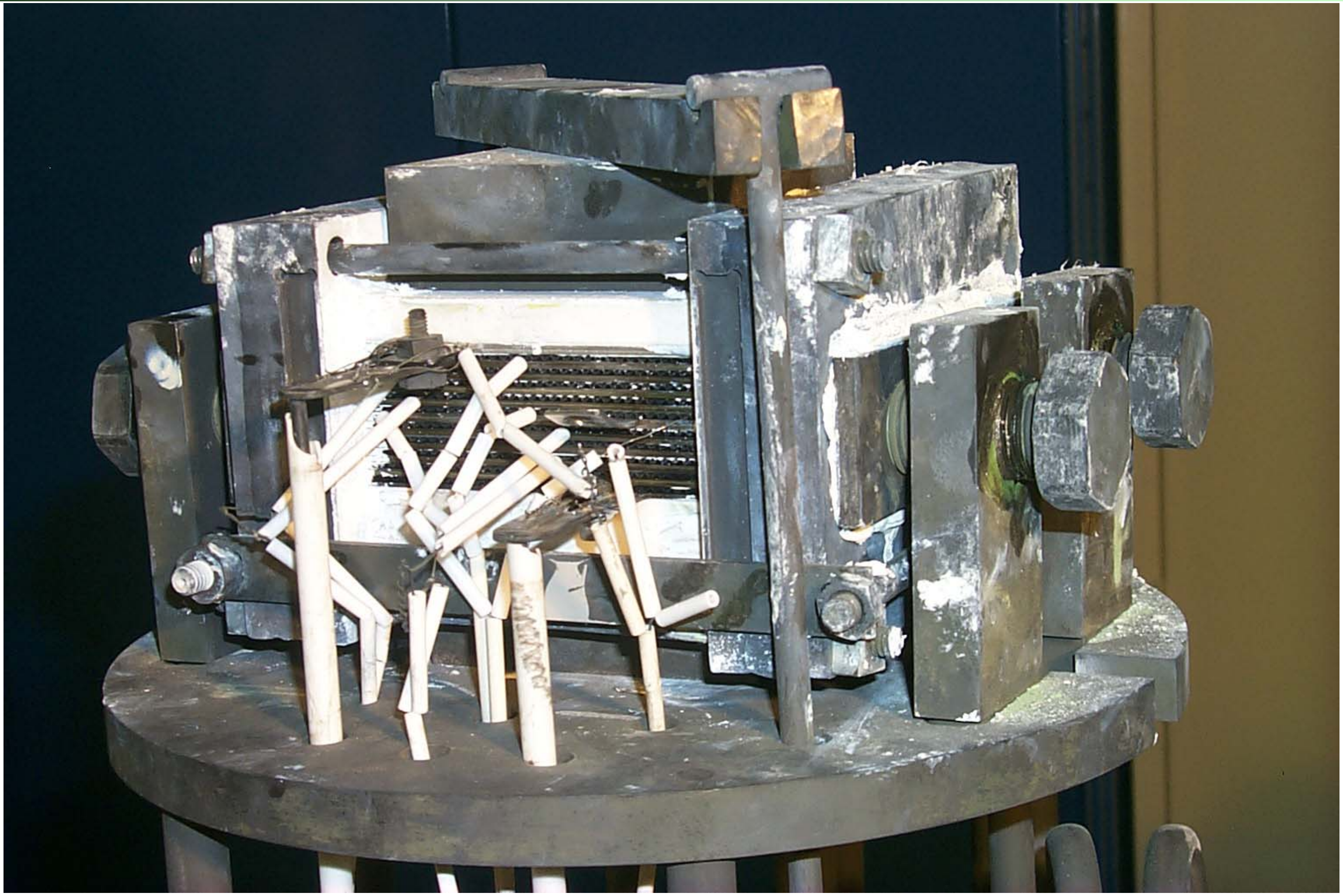
View of air flow passages, inside furnace at 800°C



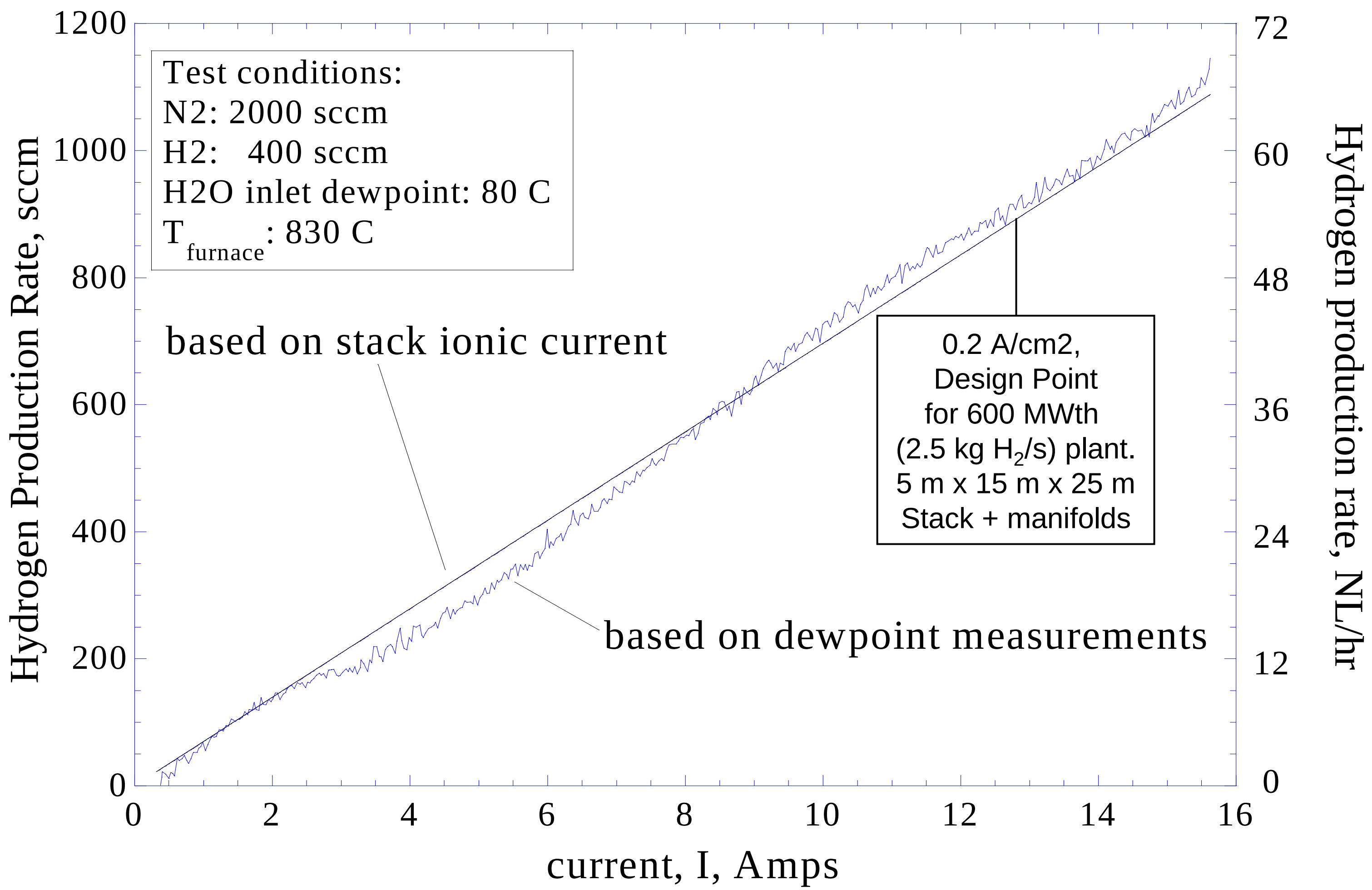
Stack Internal Components



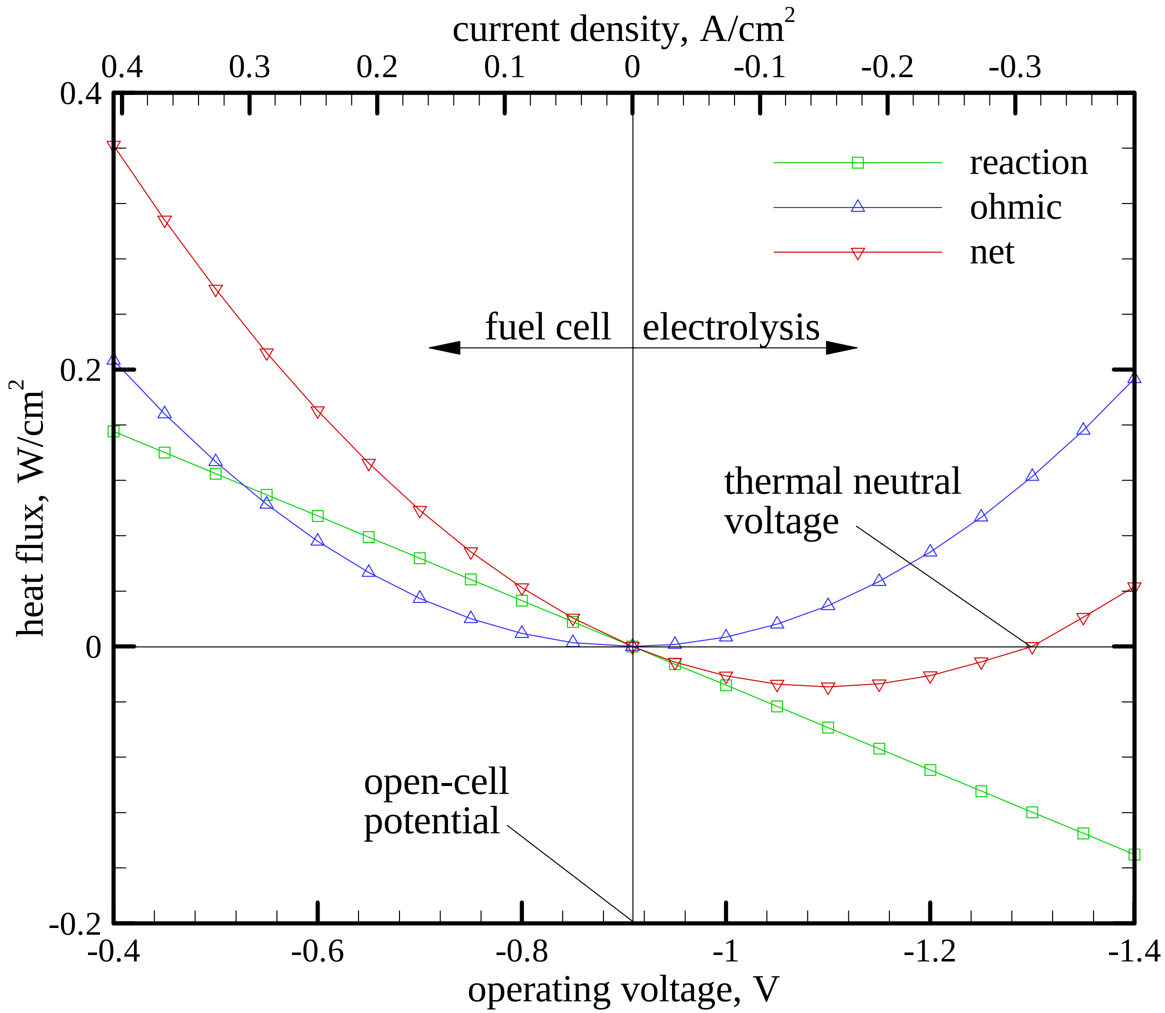
Fully instrumented stack prior to testing



Hydrogen Production at 830° C



Energy Budgets in fuel-cell and electrolysis modes



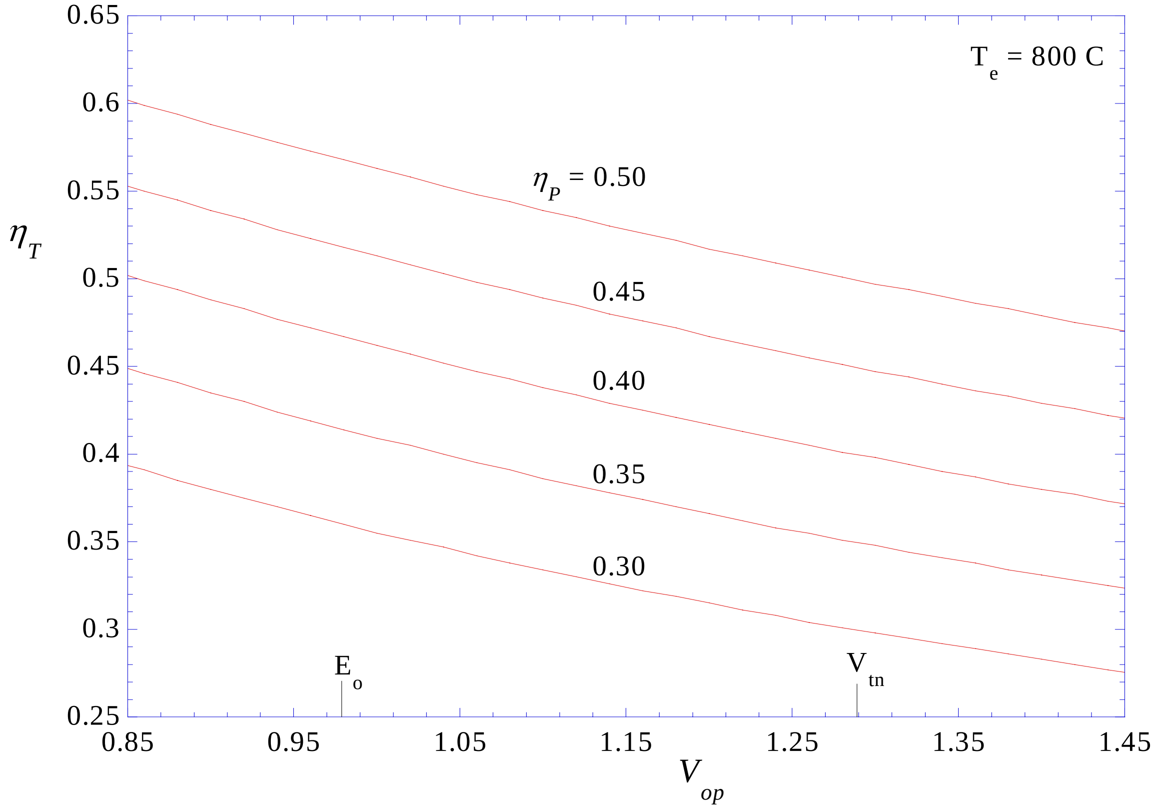
Stack ASR = 1.25,
T = 927 C,
$y_{H2,i}$ = 0.1,
$y_{H2,o}$ = 0.95

$$V_{tn} = \frac{-\Delta h_R}{2F}$$

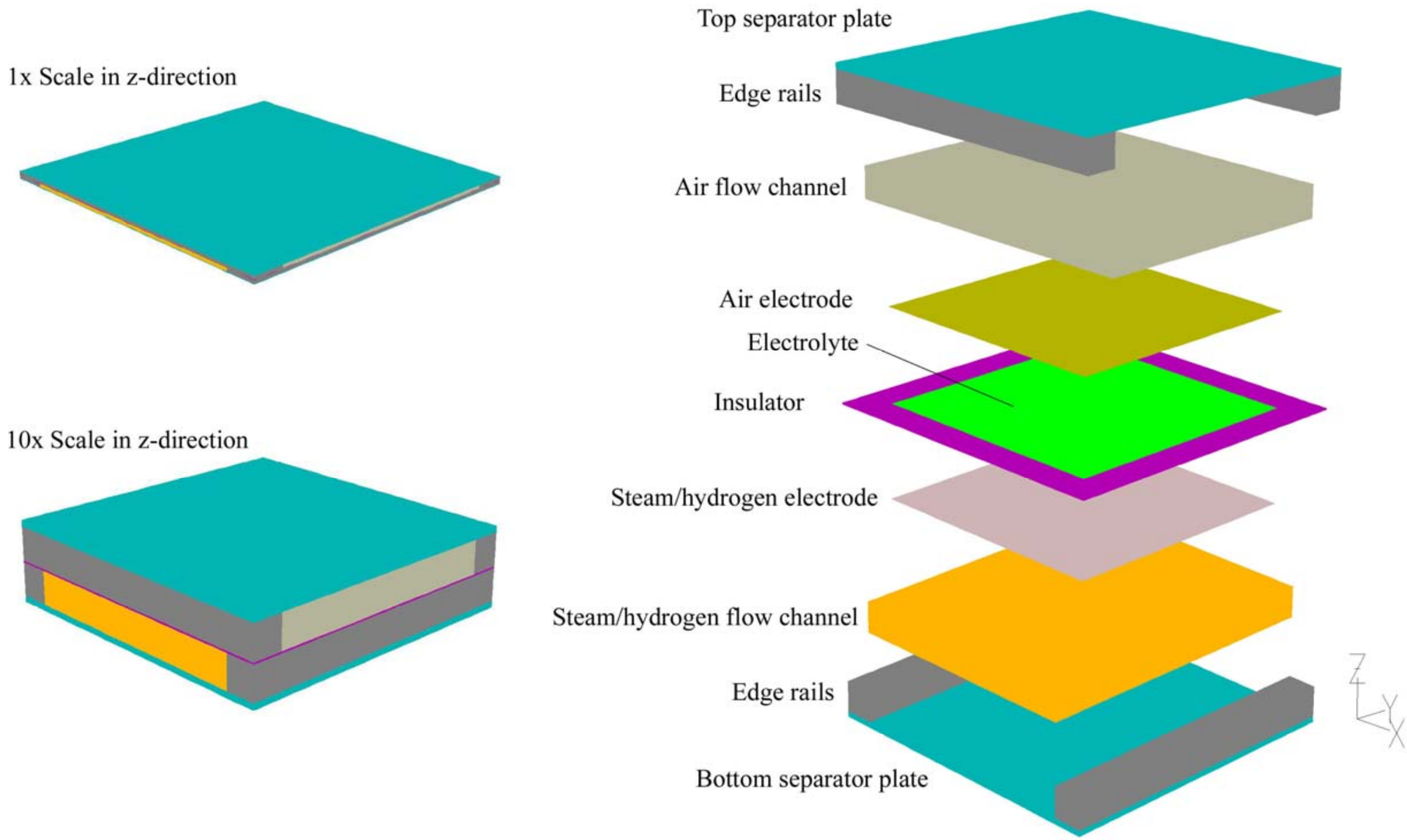
(1.291 V at 1,200 K)

Overall Hydrogen Production Efficiencies As a Function of Power Production Thermal Efficiency

and Electrolyzer Per-cell Operating Voltage

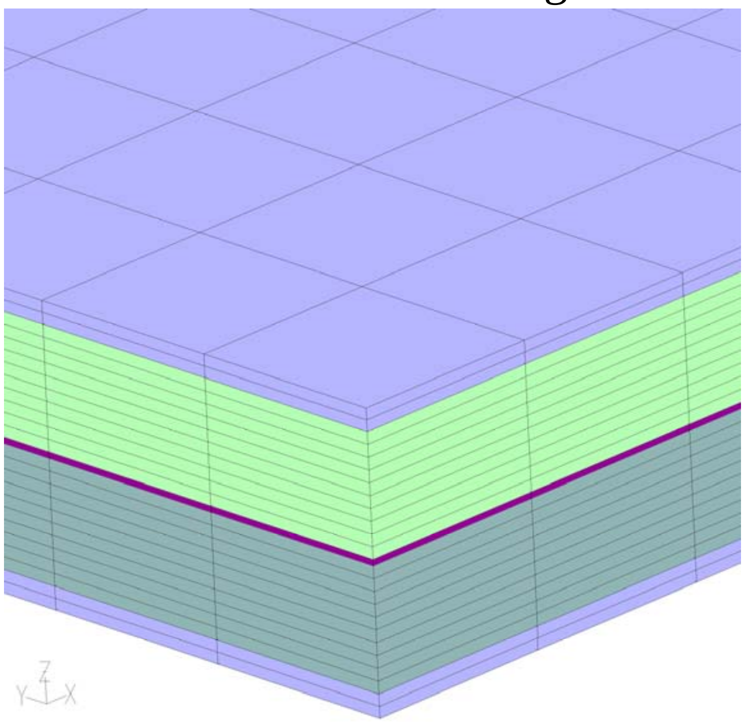


FLUENT Single-Cell SOEC Model

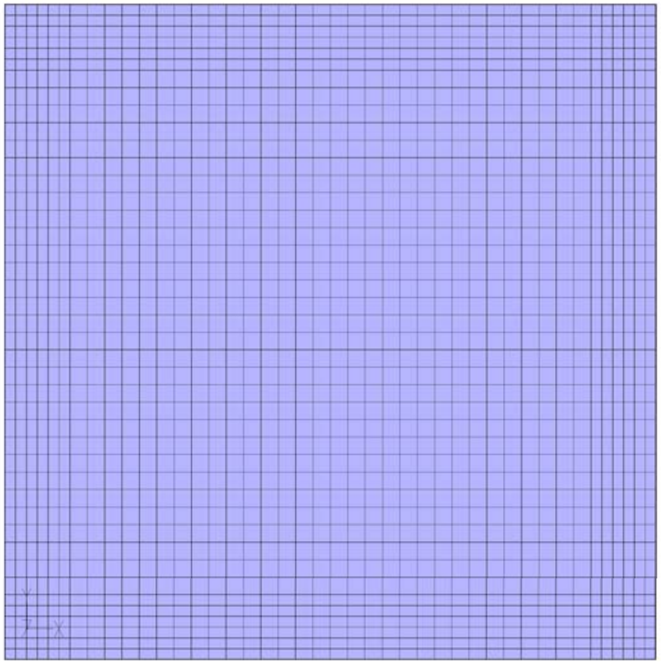


Details of 3D numerical mesh

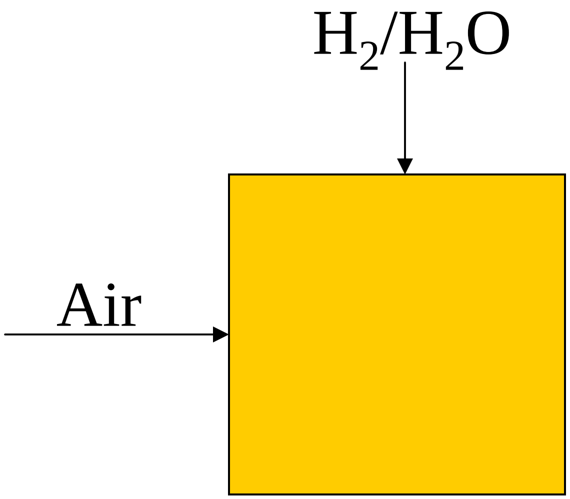
Closeup of corner, showing vertical element stacking



Top view, showing 42 x 42 element grid

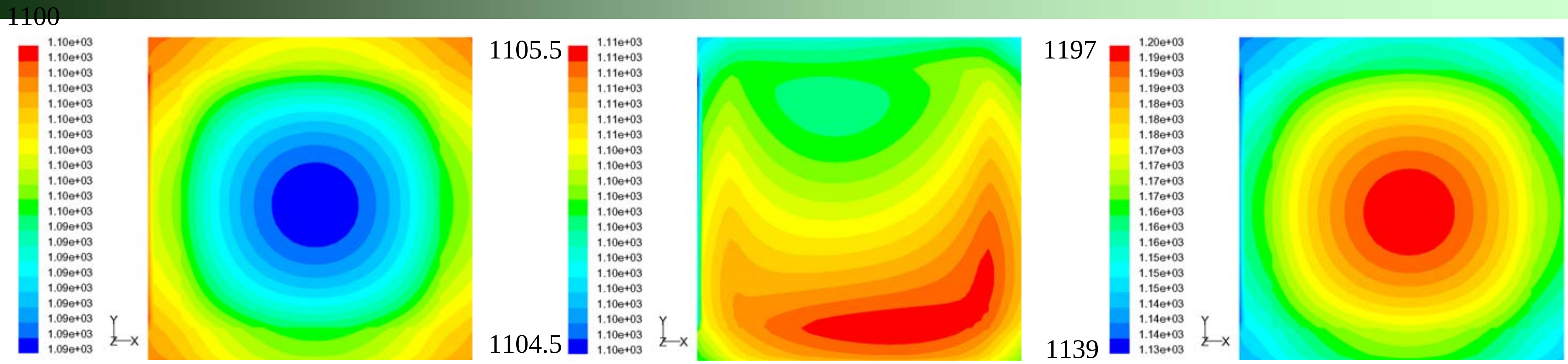


CFD Contour Plots



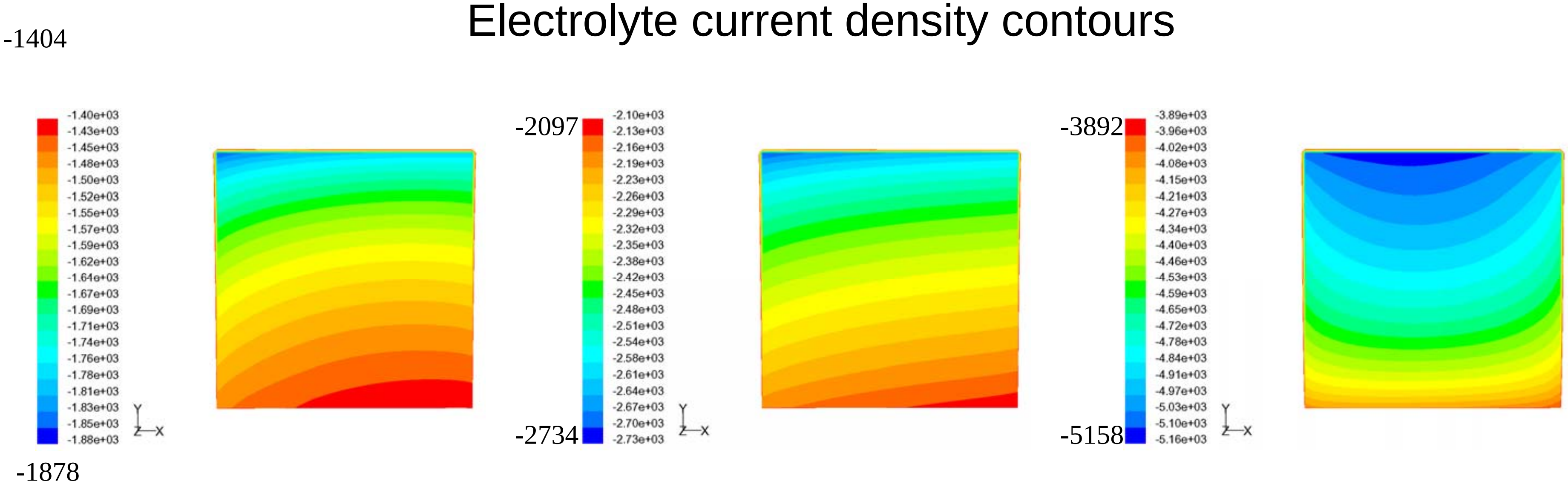
Electrolyte/insulator temperature contours

0.156 A/cm²; 1.164 V | **0.2344 A/cm²; 1.306 V** | **0.4688 A/cm²; 1.640 V**



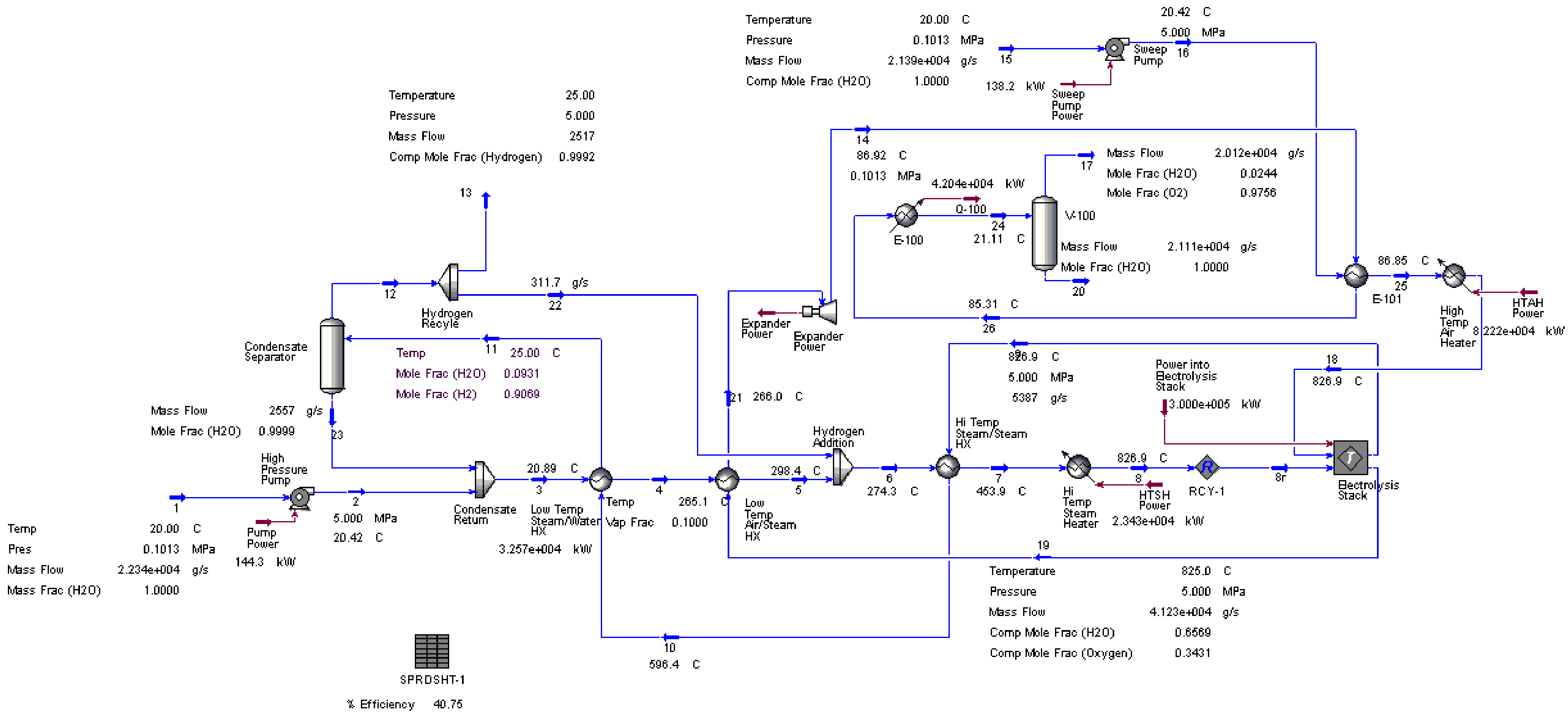
near thermal minimum | near thermal neutral | above thermal neutral

Electrolyte current density contours



HYSYS Model of a 300 MW Electrolysis Plant

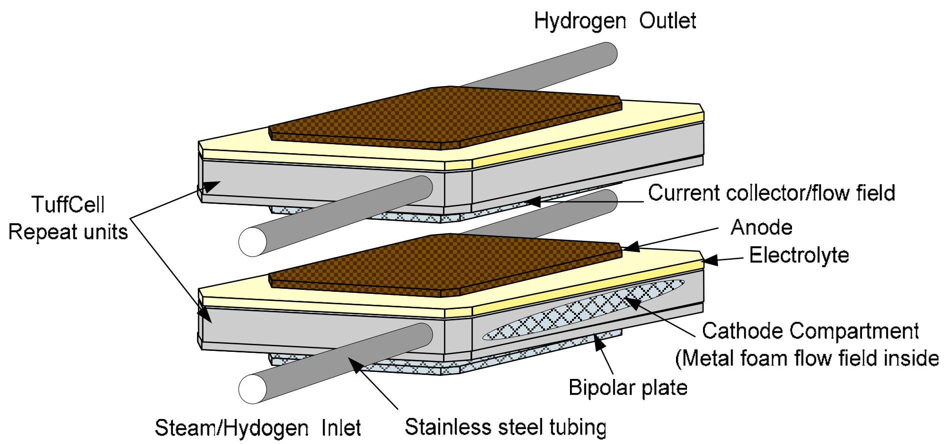
HTE



HTE System Definition
3.1.3 System Configuration
ANL, PIs: Petri, Myers, Carter, Yildiz

ANL's new cell design addresses the cost and durability issues facing SOECs (and SOFCs)

- **Cell is supported on a metallic bipolar plate; layers are sintered together in one high-temperature process.**
  - More robust to thermal cycles and mechanical shock that ceramic-supported cells.
  - Thickness of expensive ceramic-containing layers is minimized.
  - Manufacturing cost is reduced through elimination of several high-temperature processes.
- **Hydrogen electrode chamber is self-sealed during sintering.**
  - Simplifies stacking of cells and sealing the stack to the gas manifolds.
  - Replaces the traditional brittle glass, glass-ceramic, or mica seals with robust sintered ceramic-to-metal bonds.



HTE Research Priorities
Laboratory Scaling Phase (10 Year Plan)

Key technical issues:

- Cell Sealing – For planar electrolysis stacks, edge and manifold sealing is a critical issue, both for stack performance and to enable efficient collection of the hydrogen product.
  - **Glass ceramic seals**
  - **Compression seals (e.g., mica)**
  - **Ceramic pastes**
  - **Significant research has been performed on stack sealing under the DOE SECA program for the fuel cell mode of operation.**
  - **Design studies and laboratory tests are needed to address these issues.**
- Interconnections – The use of metallic interconnection between planar cells would result in lower ohmic losses, improved resistance to thermal and mechanical shock, and reduced manufacturing costs.
  - **Metallic interconnections must operate at lower temperatures than ceramic interconnections.**
  - **Chromium mobilization**
  - **Contact resistance (bond layers)**

Key Technical Issues (cont)

- Electrolyte Performance – Methods for increasing electrolyte performance are under investigation
  - **higher ionic conductivity materials with comparable cost are being developed and will be examined for this application.**
  - **thin electrolytes with very high ion mobility may be produced using Thermal Spraying and/or Chemical Vapor Deposition (CVD) techniques.**
  - **methods for the production larger cells may also reduce the overall cost of cells and stacks.**
- Cathode and Anode Materials – Electrodes optimized for larger cells and for more economical production techniques will reduce the capital cost of the electrolyzer.
  - **Graded porosity electrodes**
  - **Thermal spray techniques**
- Materials costs – The use lower cost materials and the use of reduced amounts of intrinsically costly materials will reduce the overall capital cost of the cells

Milestones
HTE Systems Analysis and Experiments

FY2005
- Complete HTE conceptual design and system cost assessment
- Define HTE cell/module options, develop cell / module test plan for FY05-07
- Develop engineering model for HTE system performance evaluation
- Complete button cell experiments
- Continue stack experiments

FY2006
- Design HTE integrated laboratory-scale experiments
- Develop conceptual HTE pilot-scale experiment design
- Construct stack /module arrays for integrated laboratory-scale experiments
- Develop conceptual pilot scale module design

FY2007
- Begin HTE integrated lab-scale experimental operations
- HTE Pilot-scale experiment preliminary design
- Complete HTE cell testing
- Conduct HTE stack / module tests
- Candidate pilot scale module tests

FY2008
- Pilot-scale experiment final design
- Complete HTE integrated lab-scale experimental operations
- Implement cell/module technology improvements

FY2009
- Pilot scale experiment decision
  (Milestone sequence after FY2009 decision as shown in Table 5.2)

Major Issues in HTE Materials Needs

- **Cost of materials and cell fabrication**
- **Lifetime of the module**
  - Performance – lifetime tradeoff
  - Limiting number of thermal cycles/transients
- **Uniformity and quality of cell manufacturing**
- **Maximum temperature of interconnects**
- **Sealing, especially in planar configuration**
- **Manufacture of thin electrolytes**
- **Matching coefficients of thermal expansion**
- **Shrinkage during manufacture**

ADVANCED REACTOR, FUEL CYCLE, AND ENERGY PRODUCTS WORKSHOP FOR UNIVERSITIES

Steven R. Sherman, Ph.D.

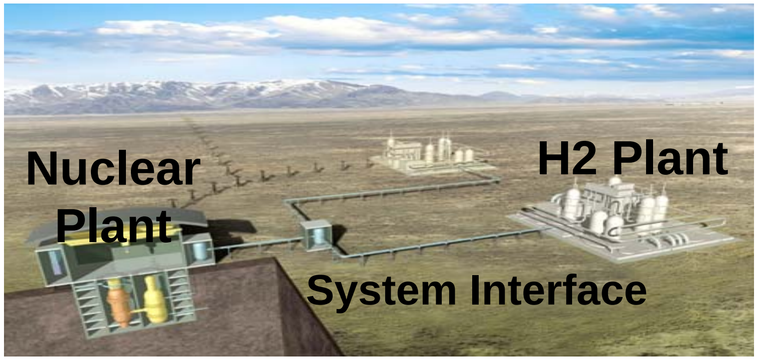
Reactor-Hydrogen Production Process Interface

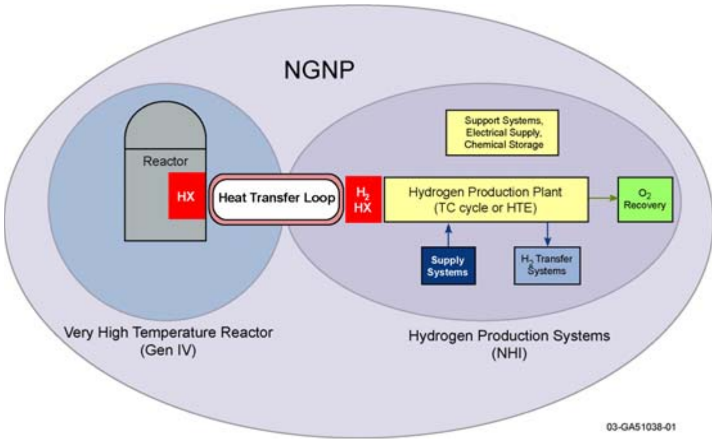
Idaho National Laboratory

Workshop for Universities
DoubleTree Hotel, Rockville, MD
June 16-17, 2005

THIS RESEARCH AREA INCLUDES

- Development of high-temperature heat transfer network to enable linkage of nuclear plant to hydrogen production plant
  - **Materials (structural and fluids)**
  - **Heat exchanger design and development**
  - **Modeling and simulation**
  - **Experimental testing**
- Development of hydrogen plant ancillary systems and infrastructure for pilot-scale and engineering-scale nuclear hydrogen production plants





RELATIONSHIP TO NE PROGRAM PRIORITIES

- Technical area directly supports NHI objectives
  - **Will develop enabling technologies to provide thermal energy to hydrogen production plant**
  - **Will support design, construction and operation of pilot-scale and engineering scale hydrogen plants**
- Impacts Gen IV initiative
  - **Interface performance and capabilities may affect nuclear reactor power conversion configuration and performance**
  - **Reactor/process isolation decisions affect plant spacing and regulatory boundaries**

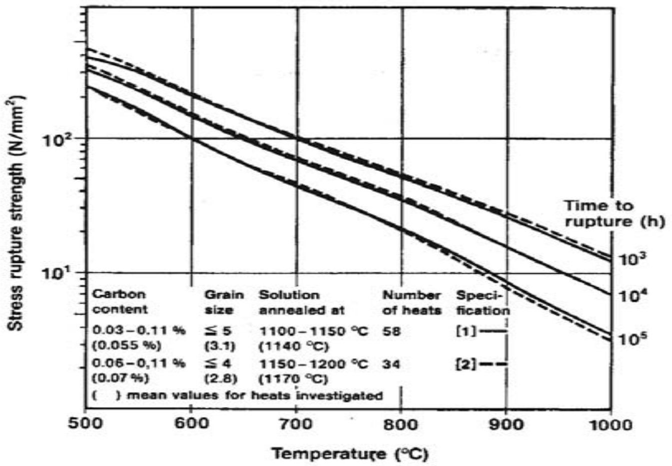
FY04 ACCOMPLISHMENTS (1)

- Definition of System Interface & Support Systems project scope (ANL-W and INEEL)
  - **Defined technical issues and barriers for high-temperature system interface**
    - **Materials**
    - **Mechanical Construction**
    - **System Interface Operation**
    - **Safety**
  - **Defined initial infrastructure requirements for pilot-scale hydrogen plant (thermochemical or HTE)**
    - **S-I, 4,200-16,000 ft^2 (500 kW to 5 MW)**
    - **HTE, 1,700-2,000 ft^2 (200-500 kW)**
  - **Defined initial balance-of-plant requirements for hydrogen plant**

FY04 ACCOMPLISHMENTS (2)

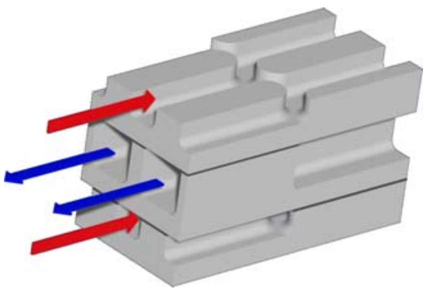
- Performed materials properties measurements on high-temperature metallic alloys (UNLV)
  - **Waspaloy, C-276, Alloy C-22 to 600 °C tested**
  - **SEM also performed**
  - **Examined stress corrosion cracking behavior of these alloys in H_2SO_4 and NaI at 90 °C**

- Developed 2-D and 3-D FLUENT models of compact heat exchangers (UNLV)
  - **Model validation**
  - **Mechanical and thermal stresses**



Total power= 50 MW
Total dimensions= 1m x 1m x 1m

Hot side (helium)
T_{in}= 1273 K
T_{out}= 905 K
D_h= 2.28 mm
P = 7.06 Mpa



Cold side (molten-salt)
T_{in}= 835 K
T_{out}= 1248 K
D_h= 1.35 mm
P = 0.1Mpa

FY04 ACCOMPLISHMENTS (3)

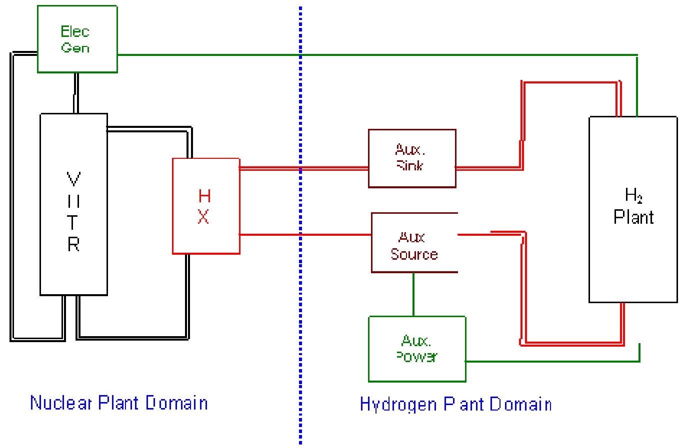
- Helium permeation testing of C-C/Si-C composite samples for application to small size prototype heat exchanger plates (UC-Berkeley)
  - **Melt-infiltrated composites tested**
  - **Initial concepts of plate designs developed**
- Corrosion testing of materials candidates in $HI+I_2+H_2O$ liquid mixture at 310°C was initiated (General Atomics)
- LESSONS LEARNED:
  - **Most challenging issues lie with materials**
    - **Structural materials**
    - **Heat transfer fluids**
  - **Heat exchanger designs must be developed in parallel with materials choices**

CURRENT WORK IN PROGRESS – FY05 (1)

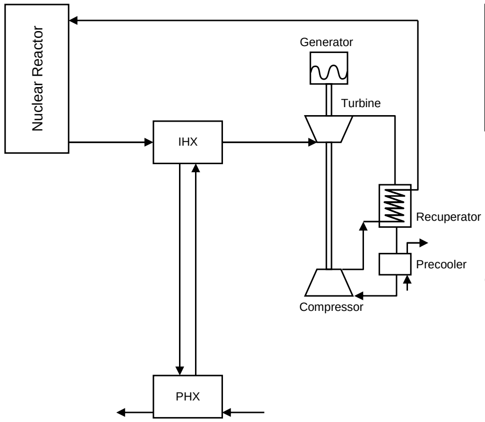
- In FY05, project funding was provided by two different pathways in the System Interface and Support Systems area
  - **Direct funding from DOE ($790K – FY05)**
    - **INL**
  - **Indirect funding from DOE through the UNLV Research Foundation ($1.8M – FY04 carryover)**
    - **Ceramatec, General Atomics, MIT, UC-Berkeley, UNLV**
- Since this area is relatively new, no NERI projects were requested in this area last year

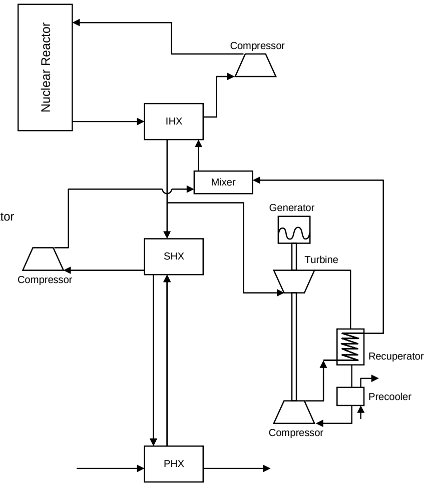
CURRENT WORK IN PROGRESS – FY05 (2)

- Examination of spacing requirements between nuclear plant and hydrogen plant based on probabilistic risk assessment tools and consideration of the probability of damage to nuclear plant core (INL)
  - **Initial results indicate 60-120 m, but further work must be done to refine the arguments**



- Thermal-hydraulic studies of interface configurations and effects on materials and fluids choices (INL)





CURRENT WORK IN PROGRESS – FY05 (3)

- Examination of materials and operational requirements for individual heat exchangers in system interface (INL)
- Continuation of 2-D and 3-D modeling work of compact heat exchanger designs (UNLV, UC-Berkeley, Ceramatec)
- Extension of mechanical properties measurements of metallic alloys to higher temperatures (1000 °C) and other materials (Incoloy 800H, Zr-705, Nb-12Zr, Nb-7.5Ta, AL 610 Stainless Steel) (UNLV)
- Study of Incoloy 800+Pt and Inconel 617+Pt for application to H_2SO_4 decomposition (MIT)
- Continuation of corrosion testing of materials for application to HI-I_2-H_2O ternary system at elevated temperature (GA)
- Studies of non-metallic alloy characterization, heat exchanger design, manufacturing techniques (UC-Berkeley, Ceramatec)

FUTURE PROJECTIONS FY06

- Materials
- HX Concepts
- Modeling
- Codes/Safety

→

- HX Designs
- System models

→

- Lab Testing
- System designs
- Improved models

→

- Support pilot-scale and engineering-scale decisions

♦ Develop baseline interface heat exchanger design and perform high temperature materials testing

♦ Construct and initiate testing of lab-scale high temperature hydrogen process materials, heat exchangers, and test loops

  - **He-He, He-molten salt, H_2SO_4 loop**
  - **Test valves, valve stems, piping, components, HXs**

♦ Complete initial design studies for H_2-plant BOP

♦ Complete initial reactor-process isolation assessment

♦ Complete assessment of applicable codes and standards

♦ Develop system models (steady state and transient response)

FUTURE PROJECTIONS FY07

- Coordinate pilot-scale component designs and testing
  - **Work must support integrated laboratory-scale thermochemical and HTE testing**
- Perform lab-scale experiments on heat exchangers & components
- Complete pilot-scale BOP design for baseline H_2 processes
- Initiate permitting activities for pilot-scale experiments

FUTURE PROJECTIONS FY08, FY09

- Complete design and testing activities to support pilot-scale decisions
- Complete design and testing for required BOP components and systems
- Complete documentation of systems interface and BOP technologies for pilot-scale experiment decisions

LIKELY AREAS FOR UNIVERSITY SUPPORT (1)

- National laboratories are product-driven – must deliver technologies according to schedule and budget demands
  - **Critical path**
  - **Larger-scale or higher safety risk**
  - **Work originating from national labs (e.g., INL work packages) will be focused and will have monthly, quarterly, yearly goals**
- Universities have resources and academic depth to explore topics in more detail
  - **Support technology development by providing basic scientific knowledge, new concepts, prototype equipment designs, analyses and mathematical modeling**
  - **Usually longer time frame is allowed to perform the work to coincide with student schedules**

LIKELY AREAS FOR UNIVERSITY SUPPORT (2)

- Materials
  - **Expansion of molten salt knowledge base**
    - **Properties, chemistry, corrosion interactions, corrosion product solubilities, redox control methods**
  - **High-temperature metallic alloys and non-metallic materials (SiC, C-C, Cordierite, others) for application to heat exchangers, pipes, valves, valve stems, pumps, etc.**
    - **Properties, joining techniques, manufacturing methods for complex structures, corrosion behaviors, hydrogen permeation**
    - **Construction of code cases for non-ASME approved materials**
  - **Other**
    - **Metallic/non-metallic sealing and joining techniques**

LIKELY AREAS FOR UNIVERSITY SUPPORT (3)

- Mechanical Construction
  - **High-temperature heat exchanger and high-temperature components conceptual designs**
  - **Improved modeling and simulation tools for steady-state and transient behavior (thermal, mechanical, effects of materials aging and creep, etc.)**
  - **Possible application of advanced assembly and construction techniques to improve system interface costs (e.g., modularity, etc.)**

LIKELY AREAS FOR UNIVERSITY SUPPORT (4)

- System Interface Operation
  - **Development/refinement of overall system steady-state and transient models and simulations that include both the nuclear plant and the hydrogen production plant**
  - **Operational control strategies**
  - **Loop contaminant effects and mitigation strategies**
- Safety
  - **Refinement of nuclear plant/hydrogen plant spacing based on regulatory frameworks, nuclear plant risks, chemical plant risks, security considerations, community safety, engineering limitations and hydrogen process demands, etc.**

BUDGET INFORMATION (1)

- FY 2004
  - **New area in FY 2004**
  - **Total: $2.355M**
    - **National Laboratories: $455K**
      - **ANL-W: $245K**
      - **INEEL: $210K**
    - **UNLV Research Foundation: $1.9M**
      - **Work performed at UNLV in FY04 used FY03 carryover money**
      - **FY04 money received, but held back for FY05**

BUDGET INFORMATION (2)

- FY 2005
  - **Total funding: $2.7M**
  - **National Laboratories: $790K**
    - **INL (ANL-W, INEEL) $790K** ***(3 projects)***
  - **UNLV Research Foundation: $1.9M**
    - **$1.9M awarded in FY05 will be used for FY06 projects**
    - **Ceramatec: $166K** ***(2 projects)***
    - **General Atomics: $300K** ***(1 project)***
    - **MIT: $150K** ***(1 project)***
    - **UC-Berkeley: $181K** ***(1 project)***
    - **UNLV: $958K** ***(3 projects)***
    - **UNLV RF: $105K** ***(project administration)***

Advanced Reactor, Fuel Cycle, and Energy Products Workshop for Universities

June 16-17, 2005 • Doubletree Hotel and Executive Meeting Center • Rockville, Maryland

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