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DOE Advanced M&S Program for Fast Reactor Applications

Tanju Sofu
NEAMS RPL Lead
Argonne National Laboratory

FAST REACTOR WORKING GROUP
& DOE FAST REACTOR WORKSHOP
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Case for Advanced Modeling & Simulation

- M&S played a role in design and licensing of existing nuclear reactors, but current LWR fleet also benefited from substantial support of large experimental programs
  - In absence of extensive experimental data, more mechanistic predictive M&S capabilities are essential for advanced reactors
- Differences in, and interdependence of, neutronic, fuel response, and thermo-structural-fluids phenomena poses unique and multiphysics M&S challenges for fast reactors
  - Range of coolants, fuel forms and reactivity feedback mechanisms
  - Micro-reactors with heat-pipe cooling
  - MSRs with moving fuel and need for chemistry modeling
- A comprehensive cross-cutting M&S program is aimed
  - Sound software development and SQA practices
  - V&V of physics models and numerical methods
  - Development of best practices for use of capabilities in different applications

DOE M&S Program Vision and Mission

- Vision: Transform, through advanced modeling and simulation, the nuclear system design and regulation landscape from reliance primarily on empirical models to predictive (closer-to-first-principles) solutions supported by limited experimental data

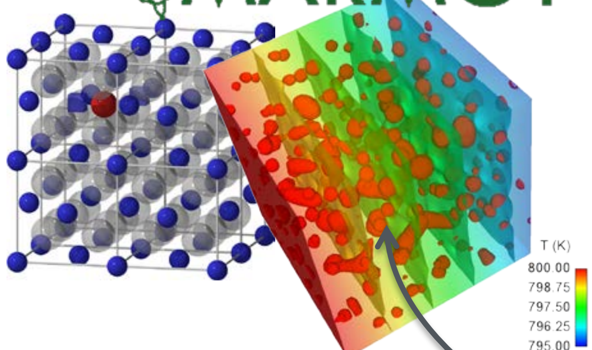
- Mission: Develop, demonstrate, and deploy usable advanced modeling and simulation capabilities to enable RD&D of innovations that align with DOE-NE missions for the existing fleet, advanced reactors, and fuel cycles

- Context: Early stage R&D relevant to industry needs, coordinated with NRC

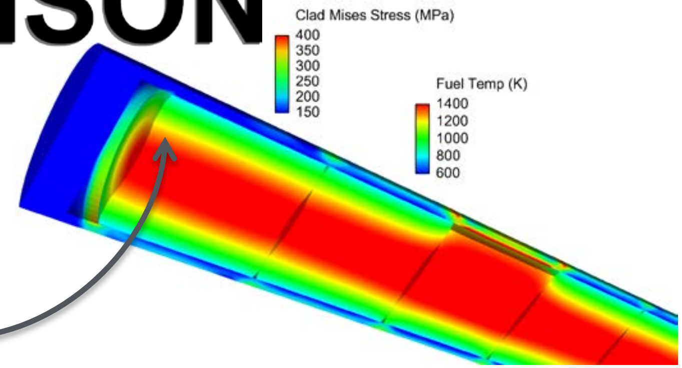
Fuel Modeling Options

MOOSE-based BISON and MARMOT codes provide a multiscale fuel performance capability

MARMOT



BISON



Atomistic/Mesoscale Material Model Development

- Predicts microstructure evolution in fuel and cladding
- Used with atomistic methods to develop multiscale materials models

MOOSE

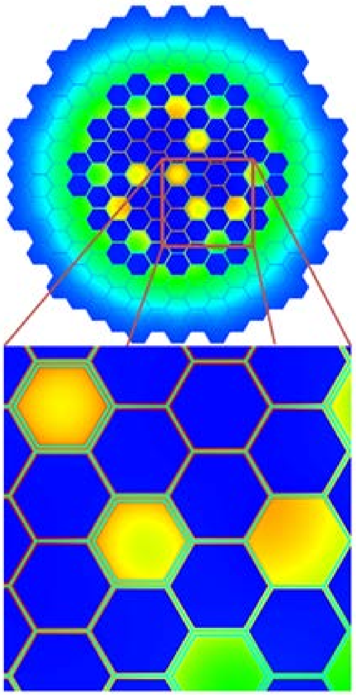
Multiphysics Object-Oriented Simulation Environment

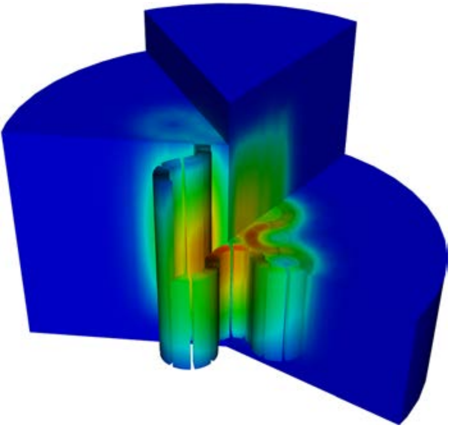
- Simulation framework allowing rapid development of FEM-based applications

Advanced 3D Engineering-Scale Fuel Performance Code

- Models LWR, TRISO and metallic fuels in 1D, 2D and 3D
- Steady and transient reactor operations

Neutronic Modeling Options



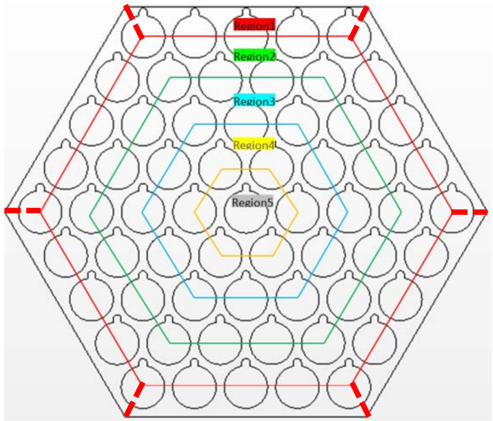


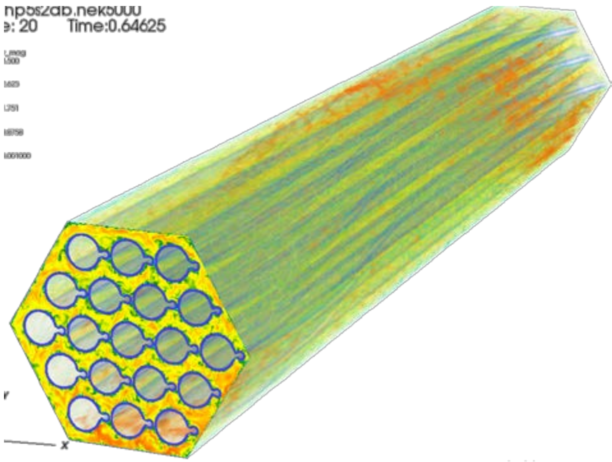
- MCC-3 and Cross Section API: Generate high quality multi-group cross sections with spatial heterogeneities
- DIF3D/REBUS: Legacy deterministic neutronics codes for fast reactors with ability to analyze entire fuel cycle
- PROTEUS: A deterministic, transient, finite-element neutron-transport solver suite with a nodal and two high-fidelity, massively-parallel neutron transport solver options (SN and MOC) with ability model complex and deformable geometries
  - Meshing tools for generation of unstructured finite element girds for Cartesian and hexagonal lattices
- PERSENT: Perturbation and sensitivity analyses based on the variational nodal method
- MAMMOTH: A MOOSE-based neutronic solver being modified for fast reactor applications for NRC's CRAB

NEAMS Workbench

Thermal-Fluid Dynamics Modeling Options

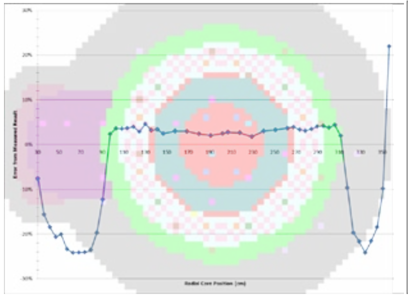
- SAM - System Analysis Module
  - MOOSE-based transient system analysis capability with a robust high-order FEM model of single-phase fluid flow and heat transfer
  - Flexible component modeling using single- or multi-channel representation of fuel assemblies
- Nek5000 - Computational Fluid Dynamics
  - An open source software with DNS, LES, and URANS modeling options for reference solutions
  - Capabilities for moving mesh, adaptive mesh refinement, overlapping multi-domain simulations
- PRONGHORN: MOOSE-based medium-fidelity conjugate heat transfer solver for pebble-bed reactors
- SOCKEYE- Heat pipe modeling tool
- YELLOWJACKET– Molten-salt chemistry and corrosion modeling tool

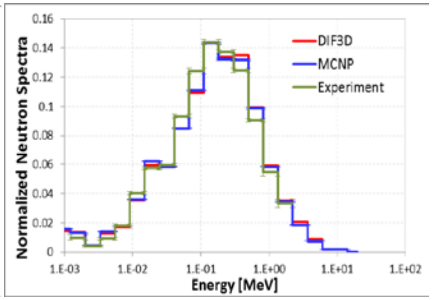


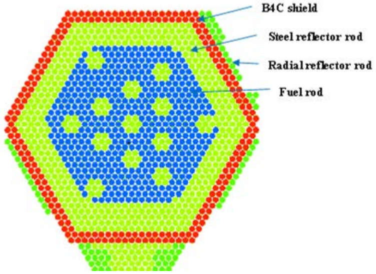


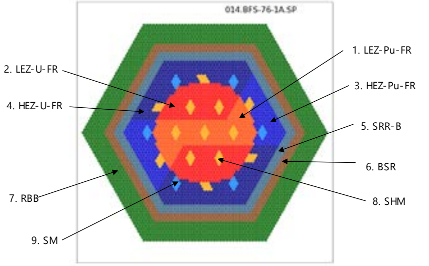
Examples for Applications to Fast Reactors (1/4)

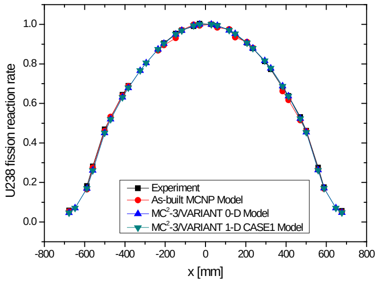
- Validation of Doppler and axial expansion worth, foil reaction rate, gamma dose, neutron spectrum predictions against ZPPR-15 tests
- Comparisons with BFS experiments for sodium and control-rod worth
  - BFS-109-2A for uniform 18.5 wt% enriched core with metallic uranium fuel for a 100 MWe long life SFR
  - BFS-76-1A for mixed Pu/U core with metallic fuel for a 300 MWe TRU burner
  - BFS-73-1 Axially heterogeneous unit fuel cell configuration with metallic uranium fuel

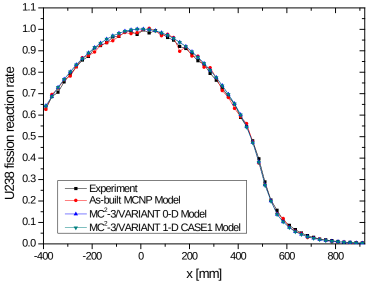






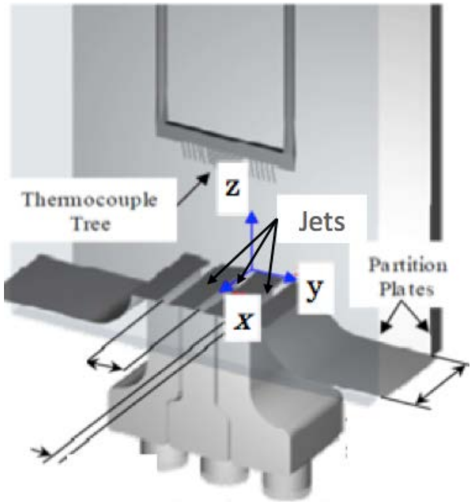


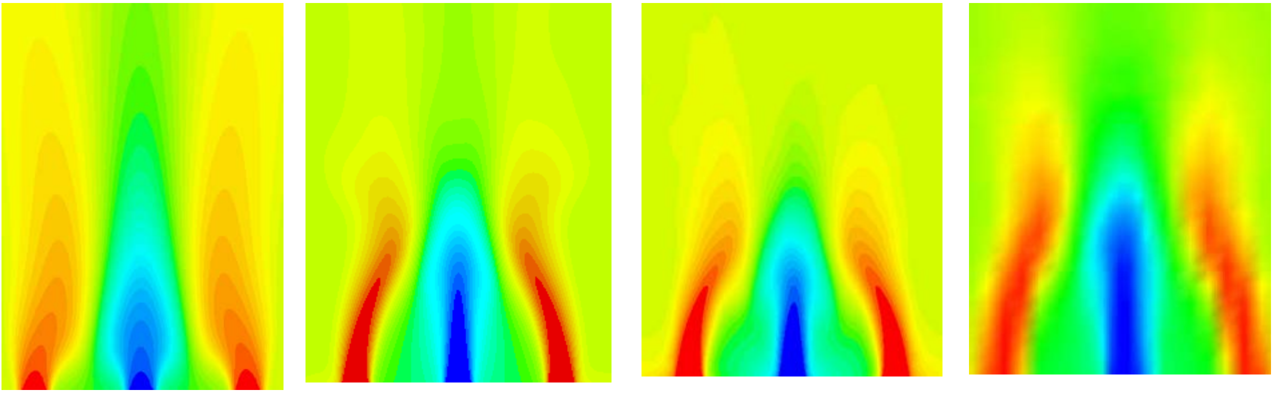




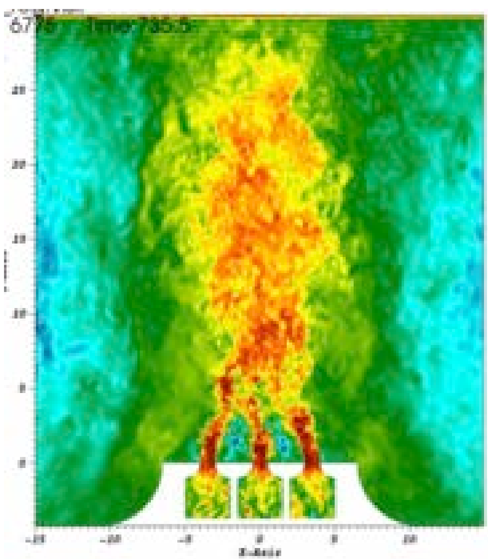
Examples for Applications to Fast Reactors (2/4)

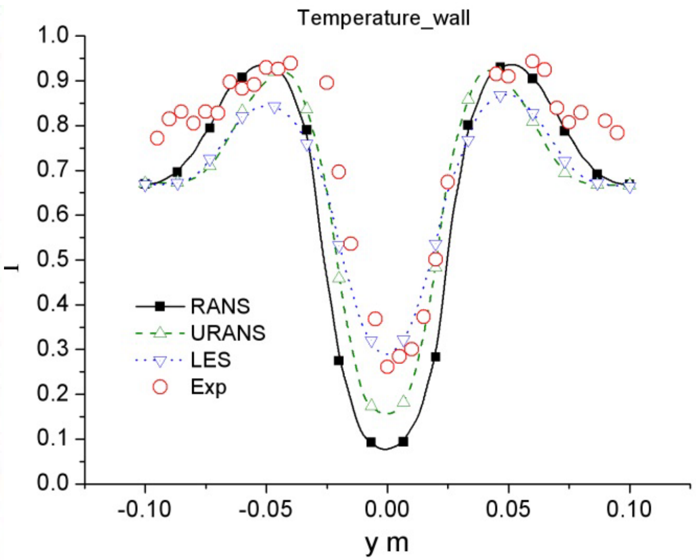
Thermal striping: Analysis of JAEA's PLAJEST sodium mixing test

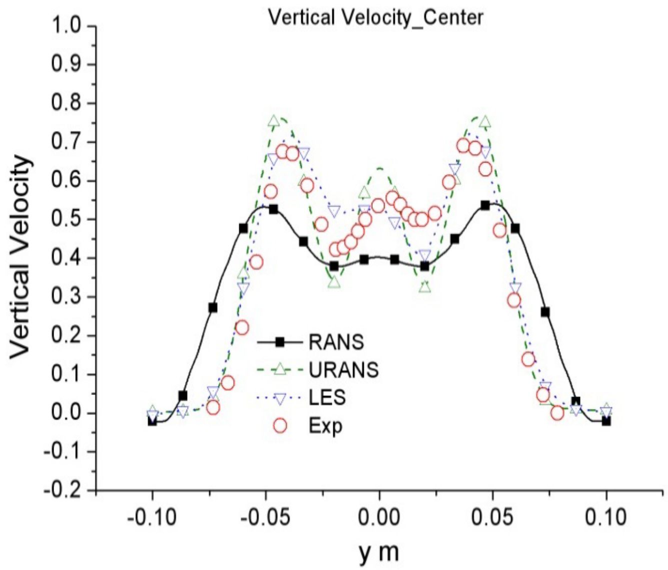




RANS *URANS* *LES* *Experiment*

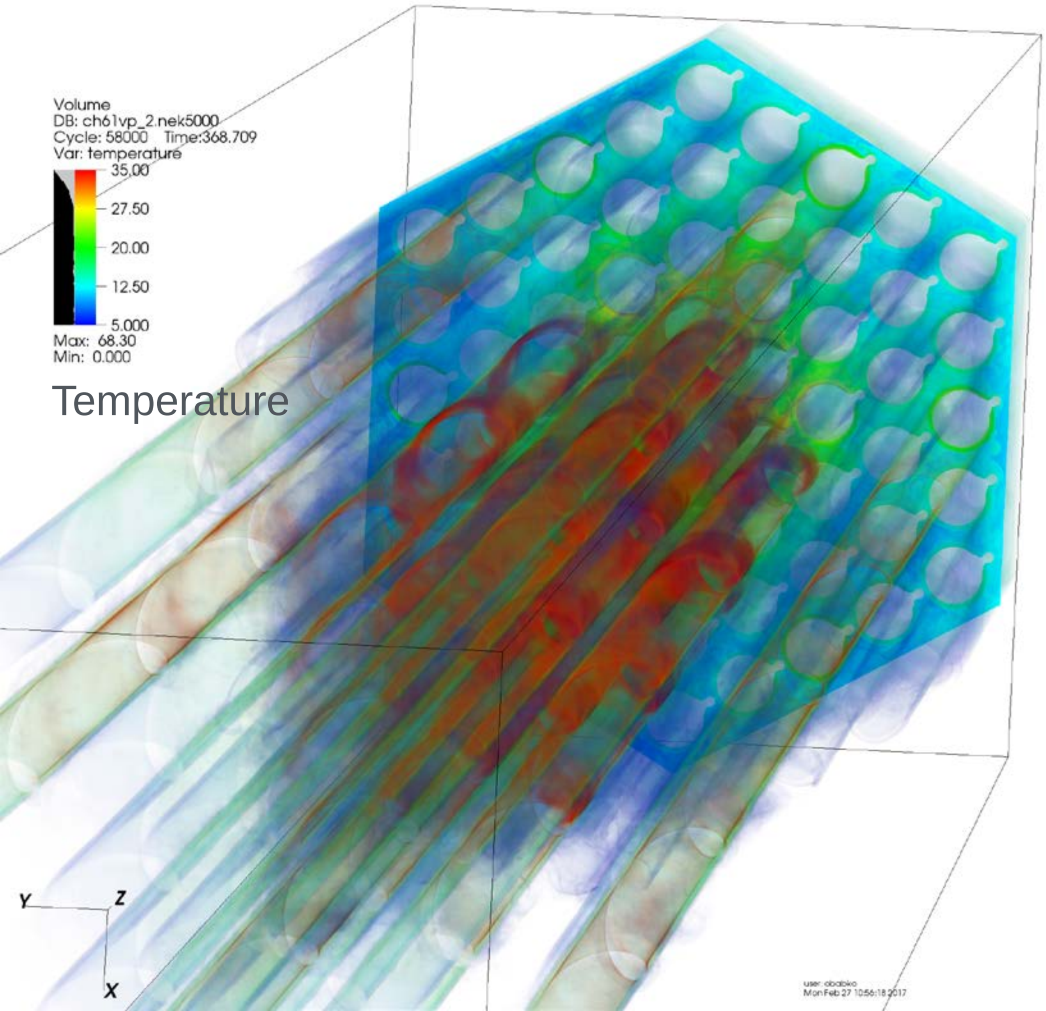




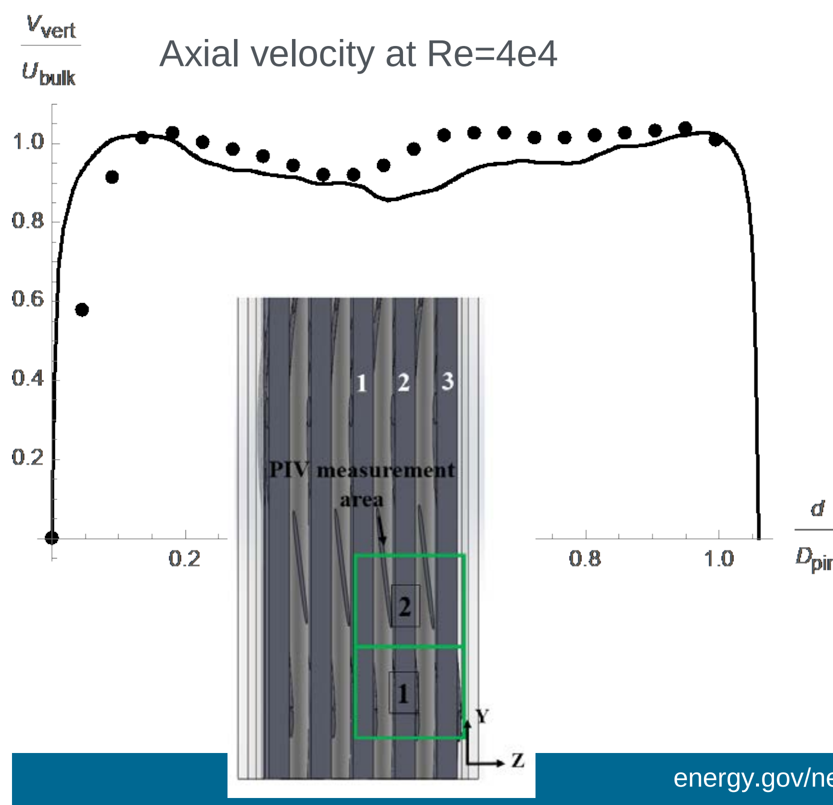


Examples for Applications to Fast Reactors (3/4)

- Simulations for undeformed 61-pin 7-pitch SFR fuel assembly to support Areva/TerraPower/TAMU/ANL collaboration
  - Largest conjugate-heat-transfer case (E=15 M, 1.9 B DOF) in production runs up to 1 M MPI ranks
  - Also leveraged for SESAME INERI w/ EU
  - Good agreement with TAMU/isotheral experiment:



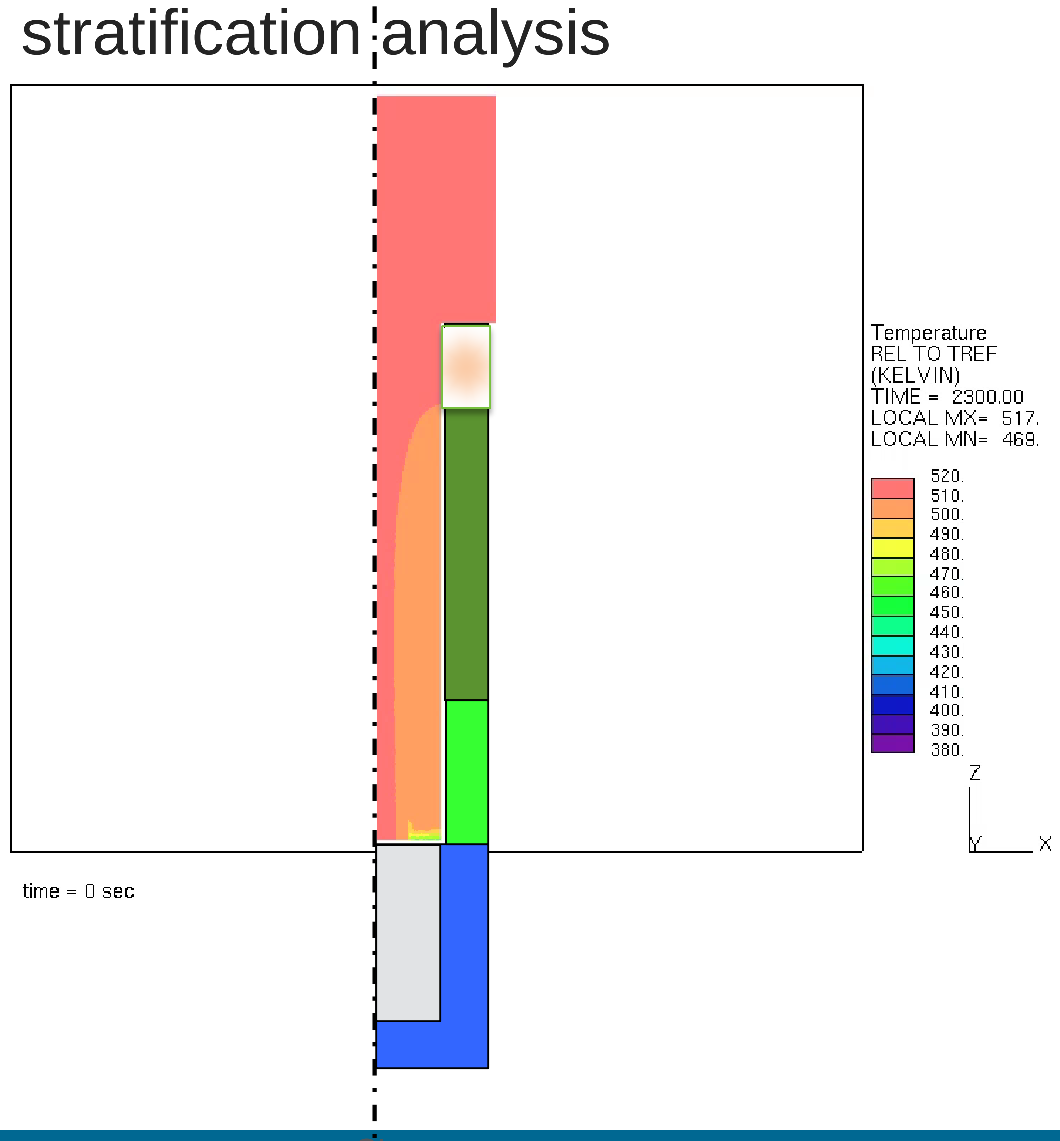
Temperature



Axial velocity at Re=4e4

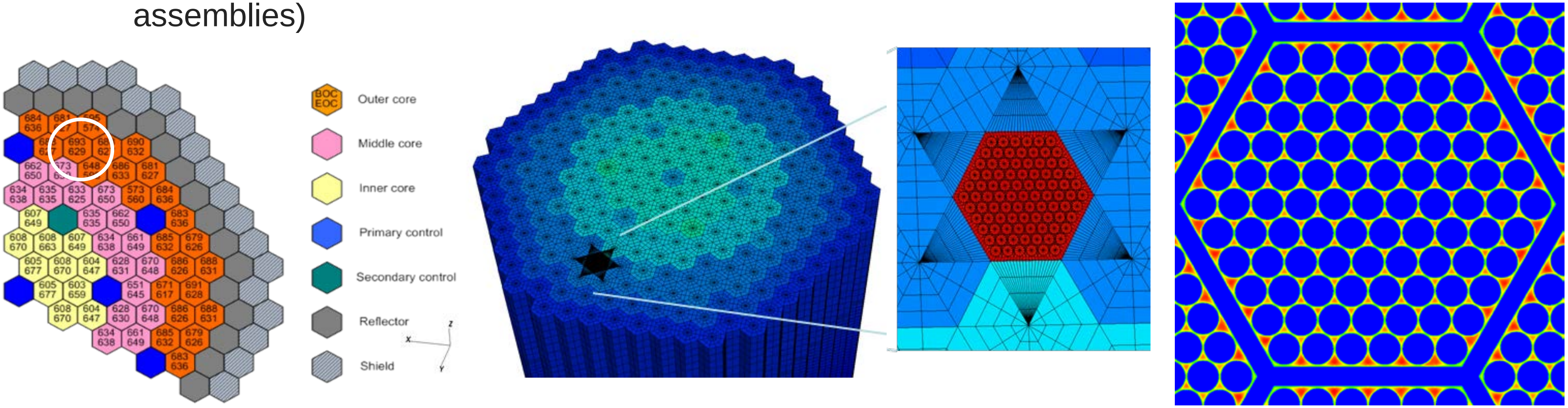
Examples for Applications to Fast Reactors (4/4)

- Thermal stratification analysis

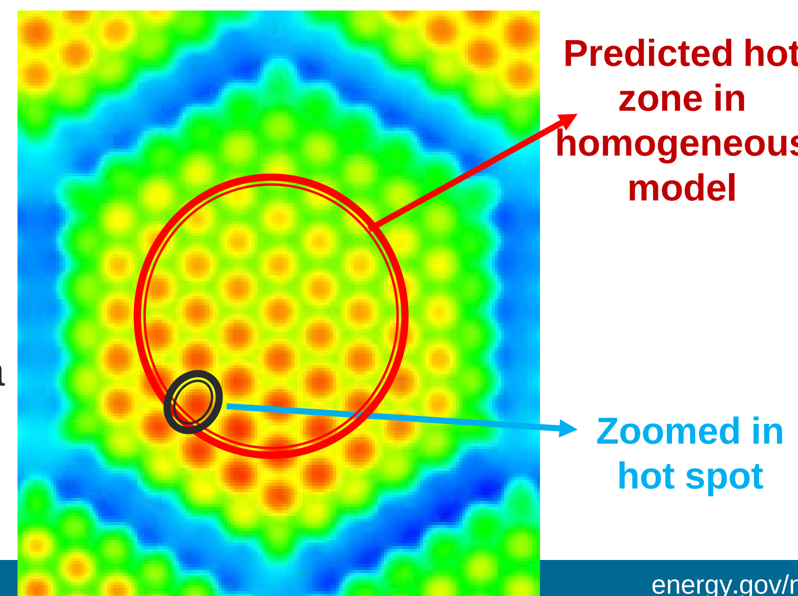


Multiscale Example: SHARP Zoom

- Assembly with peak cladding temperature is modeled as the focal area
  - Full core analyses for two zoomed-in cases (single focal assembly, focal and surrounding six assemblies)



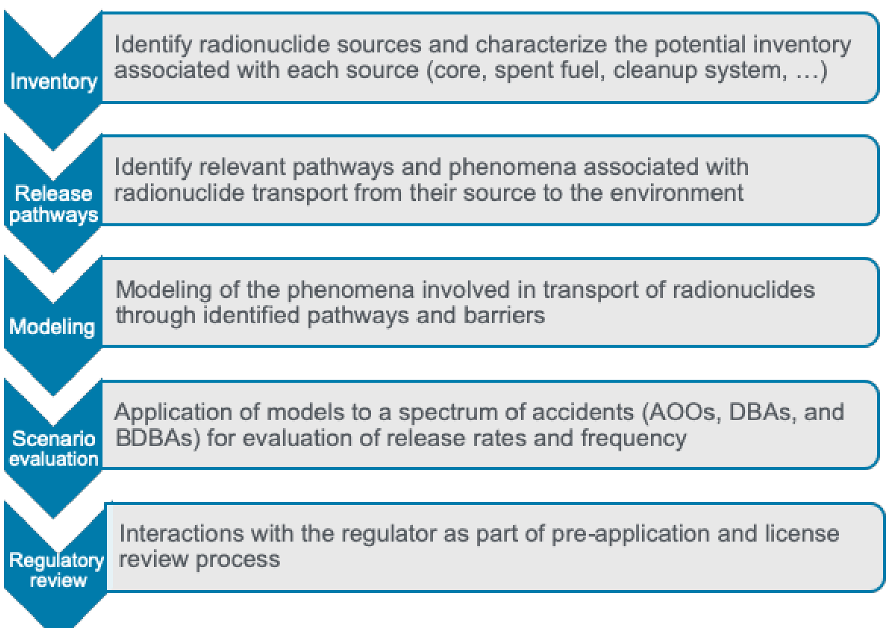
  - Coupled PROTEUS+Nek5000 solutions are compared to demonstrate proof of concept for localized high fidelity calculations
  - Use of accurate pin powers can significantly change the prediction of local hot spot location and maximum cladding/fuel temperatures
  - Offers a capability to zoom anywhere in the core, making detailed information available at a fraction of the cost of a fully heterogeneous multiphysics core calculation



Multiphysics Example: Mechanistic Source Term

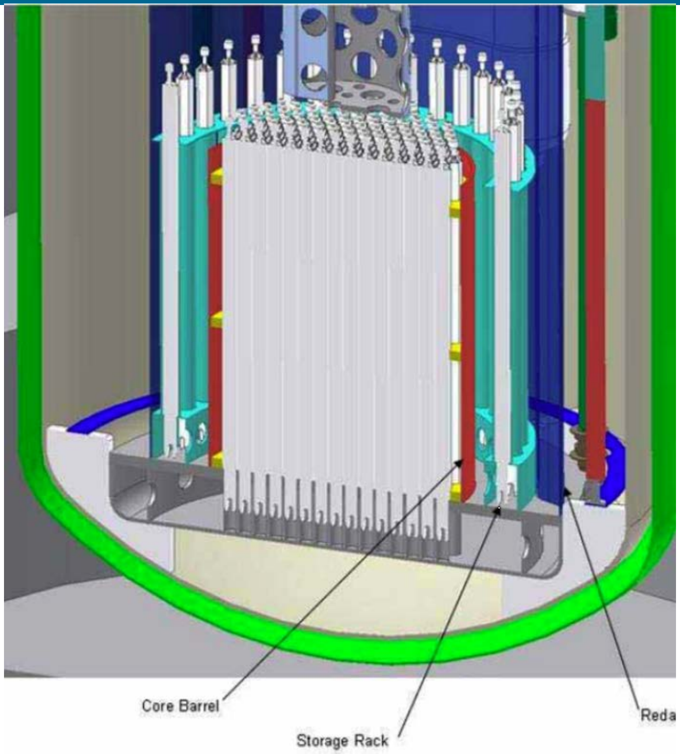
- Need for MST assessments for broad spectrum of accidents:
  - Burnup level of fuel batches
  - Timing and dynamics of accident progres
  - Conditions during fuel pin failures
  - Conditions of the primary sodium, cover gas region, and containment
  - Leakage from reactor vessel head and containment
  - “Frequent but small” vs. “infrequent but large”, early vs. large releases with different radionuclide discharge and emergency response implications
  - Radionuclide sources other than the fuel in the reactor core such as uel storage, coolant/cover-gas cleanup and chemical processing systems

- Trial LMR MST calculations:

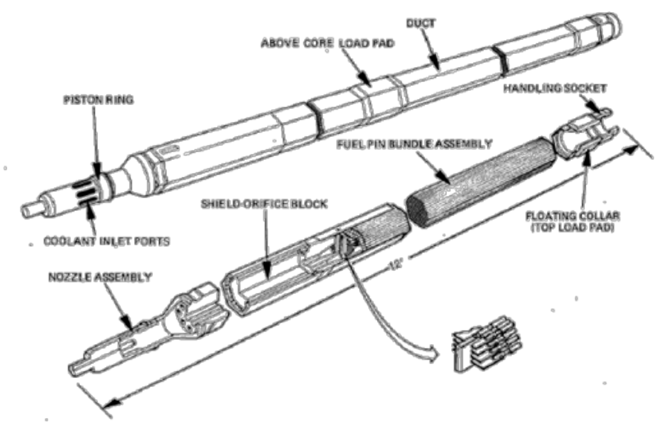


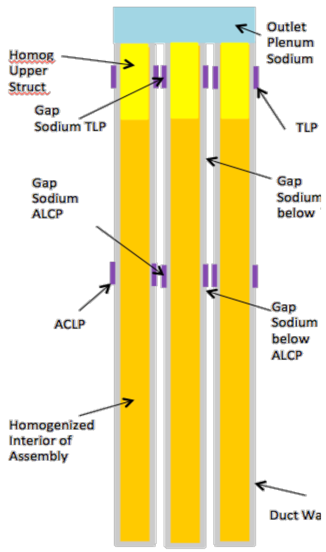
  - AFR-100 design (ANL-ART-49: http://www.ipd.anl.gov/anlpubs/2016/11/131283.pdf)
  - TerraPower: Company-funded work to repeat trial MST calculation for TWR design
  - Korean Atomic Energy Research Institute (KAERI): Source term assessments and experiments to support PGSFR licensing
  - GE-Hitachi: MST as part of PRISM PRA update/modernization effort
  - Fauske & Associates and Westinghouse: SAS4A-FATE coupling for LMR source term assessments and initial application to W-LFR
  - Two NEUP awards to UWM and UNM for radionuclide retention tests in liquid sodium and lead

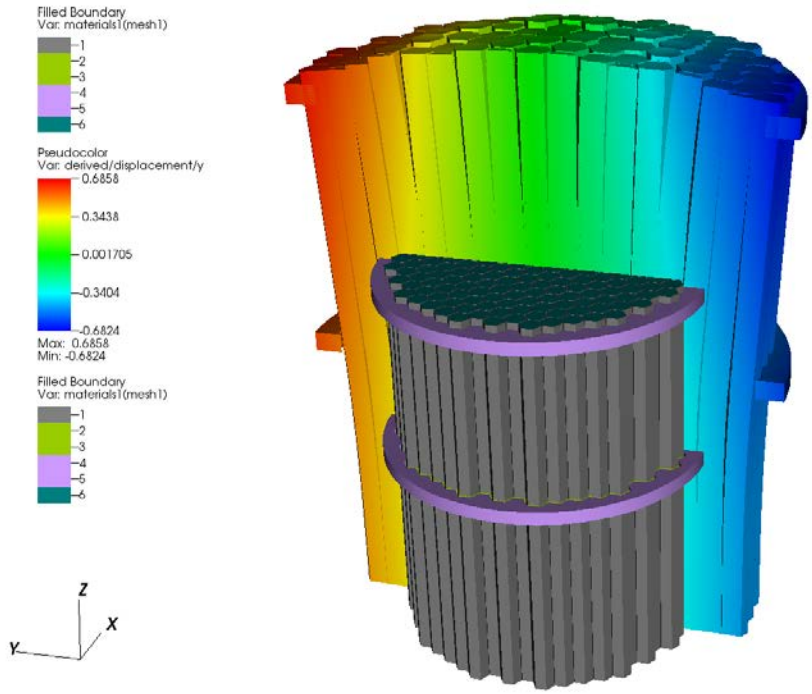
Multiphysics Example: Core Radial Expansion



- Core thermal expansion is one of the primary reactivity feedback mechanisms for FR safety
- Geometry deforming due to temperature gradients in the presence of restraining contacts
- By appropriate design of the core restraint system, neutron leakage is enhanced
- Demonstration for ABTR design



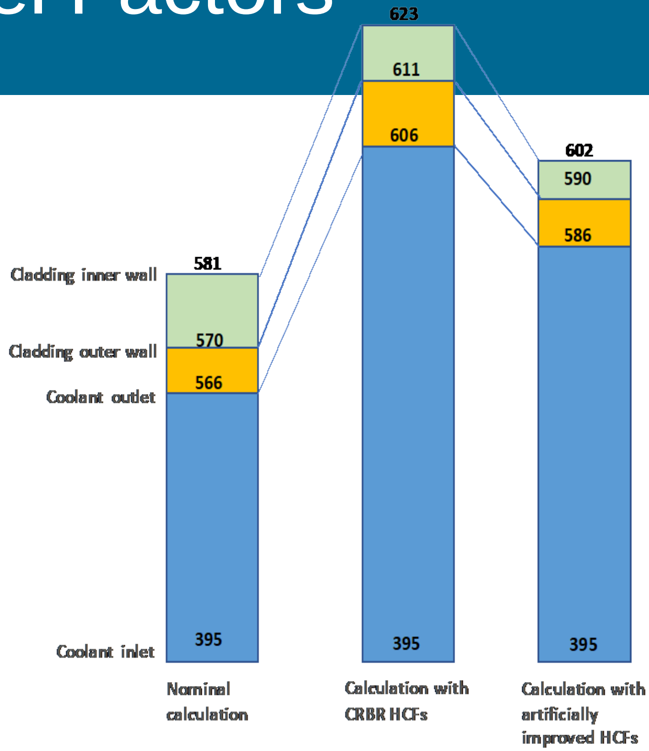


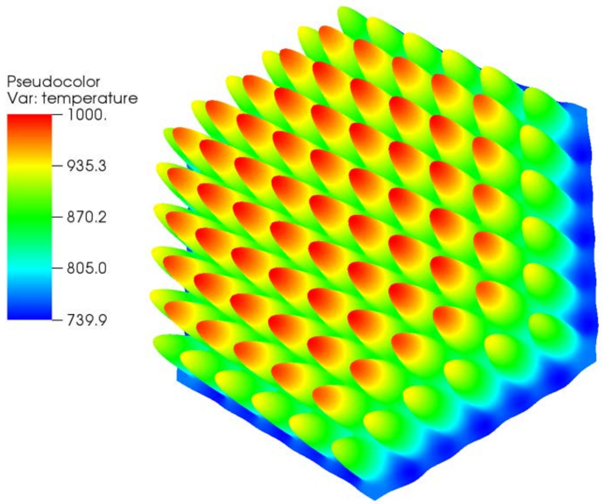


Multiphysics Example: Hot-Channel Factors

- Model various uncertainties involved in the predictions of reactor design parameters:
  - Theoretical and experimental uncertainties, instrumentation uncertainties, manufacturing tolerances, correlations…
  - To assure that fuel, cladding, and coolant temperatures do not exceed the design limits with sufficient margins
- HCFs induced by manufacturing tolerance and property uncertainties are evaluated for AFR-100

ΔT	Parameter	Uncertainties	Legacy HCF	Comp. HCF
Coolant HCF	C_p	±3%	1.017	1.016
	ρ	±0.5%	1.016	1.001
	$\Delta T_{CircClad}$		1.024	1.010
	Wire orientation		1.010	1.003
Cladding HCF	Thickness	±3%	1.030-1.050	1.018
	k	±7%	1.088	1.082
	Fissile maldist.	±6%	1.060	1.036
Fuel HCF	k	±25%	1.250	1.226
	Fissile maldist.	±6%	1.060	1.016





Results with PROTEUS+Nek5000

Questions?

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Evaluation of Hot Channel Factors for Sodium-Cooled Fast Reactors using DOE-NEAMS Tools

Emily R. Shemon
Nuclear Science & Engineering Division
Argonne National Laboratory

Fast Reactor Working Group (FRWG) / DOE Fast Reactor Workshop
July 9-10, 2019
Argonne National Laboratory

NEAMS
NUCLEAR ENERGY ADVANCED MODELING & SIMULATION PROGRAM

Motivation

- **High fidelity solvers developed under the DOE-NEAMS program target single/multi-physics advanced reactor applications**
  - Neutronics, thermal-hydraulics & structural mechanics codes form the "SHARP" toolkit
  - Systems analysis, fuels modeling, and additional tools
  - User interface / code integration tool Workbench

- **Calculation of <u>hot channel factors (HCF)</u> identified as potential area to benefit from advanced modeling and simulation**
  - Applied NEAMS tools to calculation of AFR-100 sodium-cooled fast reactor HCF

- **Goal: Demonstrate reduction or elimination of geometrical and/or physics uncertainties through the use of high fidelity solvers**

Hot Channel Factors (HCF)

Hot channel factors (HCF)

- Direct and statistical types
- Introduced to account for impact of uncertainties on peak temperatures (needed to determine safety margins)

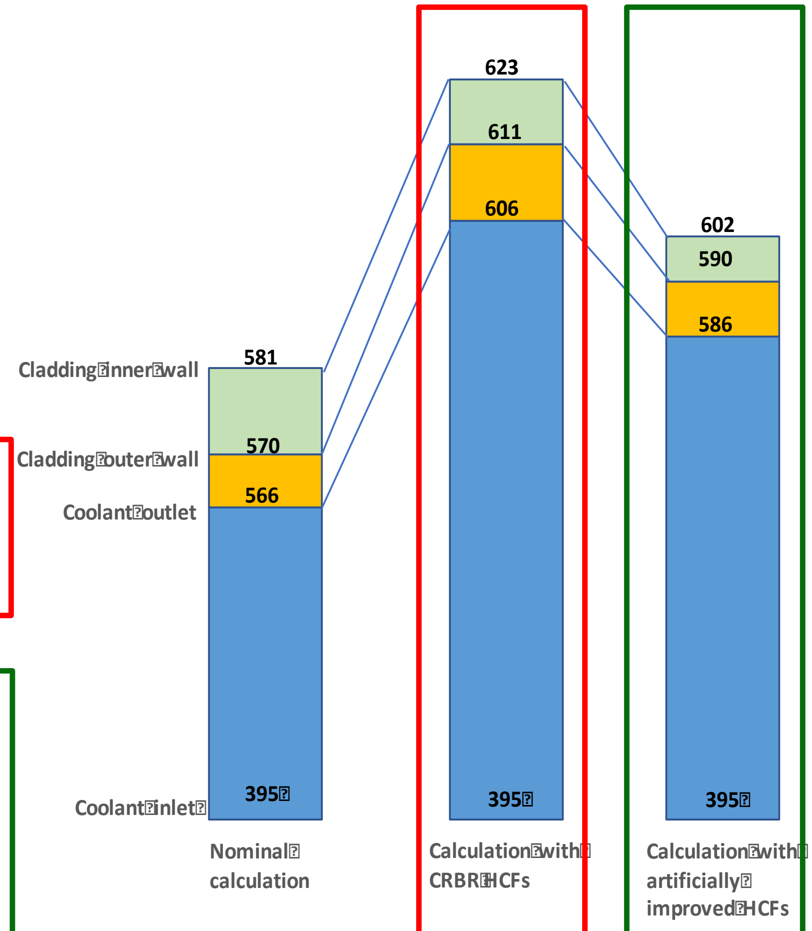
$$\Delta T_{cool} = (\Delta T_{cool}^{nom}) \prod_{i=HCFs} DH_{cool}^{i} + \sqrt{\sum_{i=HCFs} \left(\Delta T_{cool}^{nom} \times SH_{cool}^{i}\right)^2}$$

Legacy estimation

- Low-fidelity calculations and costly mockup experiments

Modern evaluation

- High-fidelity multi-physics capabilities + modern computing power allow **calculation of HCFs without mockups**, and **reduction or elimination of HCFs** that are associated in M&S uncertainties



Hypothetical values

Progress in Fast Reactor Computation Capabilities

1970 — NRC ('75) — 1980 — 1990 — 2000 — 2010 — 2020

HCFs	FFTF ('76)	CRBR ('80)	FFTF-Rev. ('90)	Mark-V, EBR-II ('95)		
Fastest HPC power	Megaflop (10^6)	Gigaflop (10^9)	Teraflop (10^{12})	Petaflop (10^{15})		
ENDF/B	IV	V	VI	VII.0	VII.1	
Cross sections	XSRES-WIDX	MC²	MC²-2	MC²-3		
Flux solver	DOT (XY)	XSRES-WIDX	DIF3D-FDM/Nodal	DIF3D-Variant	PROTEUS	
Steady-state T/H	COBRA	ENERGY	SuperEnergy-2	SE2-ANL	Nek500	
Fuel	LIFE-2	LIFE-3	LIFE-4C,N	LIFE-Metal	BISON	
Structure	NUBOW	NUBOW-3D	Diablo			

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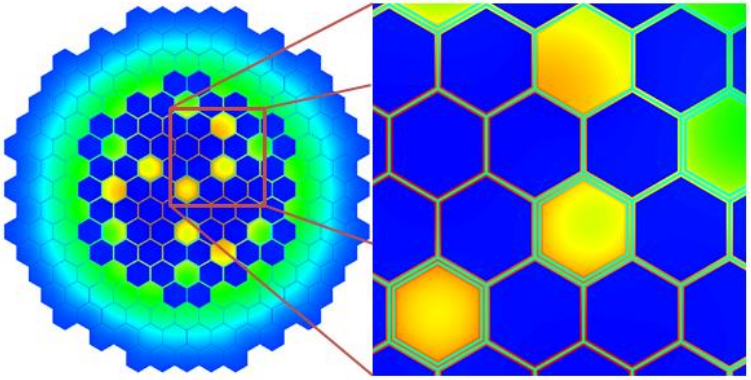
Modern HCF Calculation Capabilities (DOE-NEAMS)

Neutronics (MC^2-3, PROTEUS)

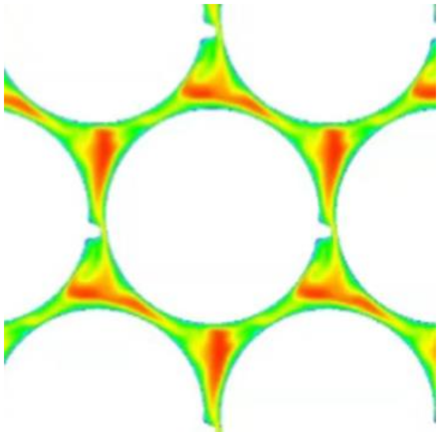
- MC^2-3: fast spectrum XS processing tool accounting for global / local heterogeneity effects
- PROTEUS (general reactor types and spectrums)
  - Deterministic transport
  - Arbitrary geometry complexity (mesh deformation, irregular lattice, unstructured mesh)
  - Transport and full kinetics capability
  - Fidelity and parallelism adjust to user needs
  - Scalable to HPC platforms



Absorption in SFR design with explicit ducts

T/H (Nek5000)

- Spectral element CFD code
- Explicit modeling of detailed geometry (wire wrap)
- Verified and validated for many regimes
- Single phase flow and 2-phase flow modeling
- Scalable to HPC platforms



Coolant velocity in wire wrap SFR design

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Modern HCF Evaluation of SFR HCF with NEAMS Tools

Identified HCFs from metal fuel SFR HCF datasets (EBR-II, FFTF, CRBR)

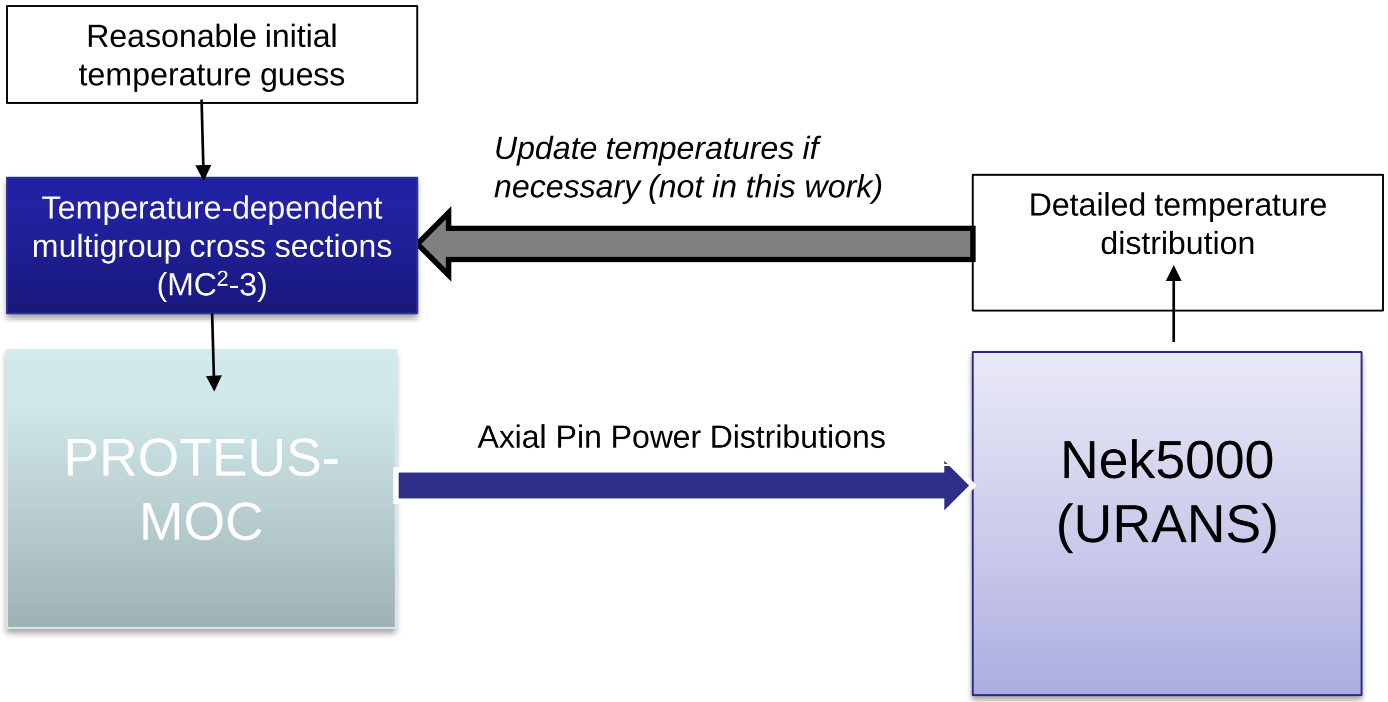
	HCFs	Major source of uncertainties
Direct HCF	Cladding circumferential temp.	Approximation in axial coolant velocity and azimuthal temperature distribution around a fuel pin (bare bundle approximation of wire wrapped assembly)
Statistical HCF	Wire-wrap orientation	Manufacturing mistake
	Sub-channel flow area	Clad fabrication tolerance, bowing, etc.
	Cladding properties	Manufacturing tolerance and empirical correlations, etc.
	Coolant properties	Material impurity and empirical correlations etc.
	Fissile fuel maldistribution	Manufacturing tolerance
	Fuel thermal conductivity	Manufacturing tolerance, uncertainties in irradiated fuels.

Selected AFR-100 sodium-cooled fast reactor design for HCF analysis

- Fuel assembly – 91 wire wrapped pins arranged in tight lattice
- Axial and radial enrichment zoning (18% bottom / 8.8% middle / 18% top) in inner fuel assembly
- Similar fuel form to EBR-II (metal fuel, U-10Zr) for good comparison with legacy data

High Fidelity Simulation Model

- **Performed sensitivity studies to develop efficient coupling procedure and physics models**
  - One-way vs two way iterations
  - T/H: Bare bundle (+/- momentum source) vs explicit wire wrap (12X computational cost)



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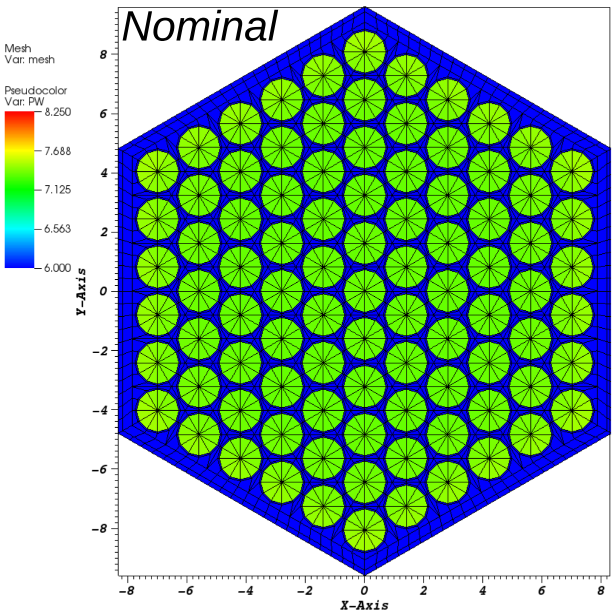
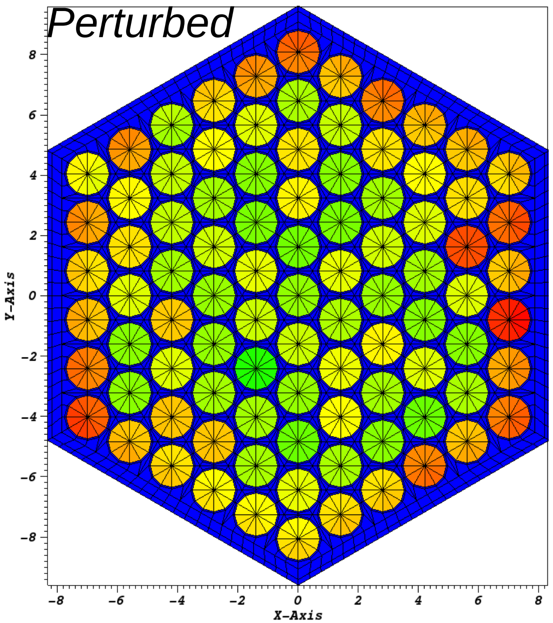
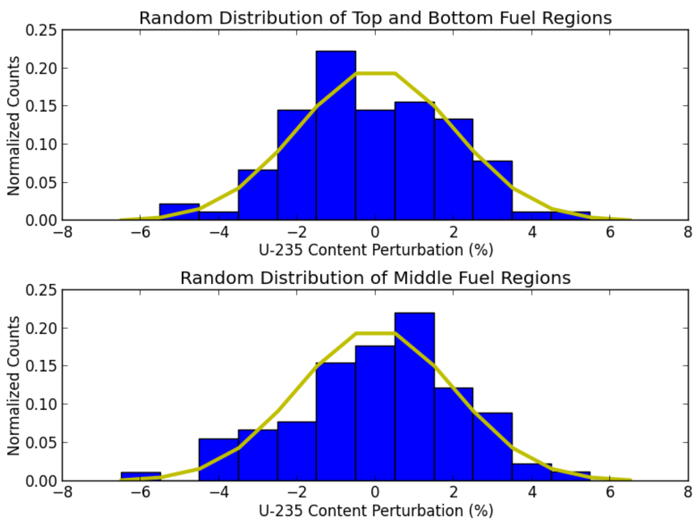
Stochastic Modeling of Fissile Content Maldistribution

- **Generated thirty assembly models w/ randomly perturbed (+/- 6% enrichments**
- **Selected bounding case with maximum pin power for Nek5000 analysis**
  - Peak = 19.082 kW (Pin 41, Ring 5), Avg = 18.276 kW





Example Perturbation Dist.



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Computed Hot Channel Factors

Metal fuel, legacy codes + mockup experiments

Metal fuel, high fidelity modeling

Coolant HCF ΔT = Coolant Outlet – Coolant Inlet	**Uncertainties (3σ) %**	**EBR-II Legacy**	**AFR-100 SHARP**
Coolant Specific Heat	±3	1.017	1.016
Coolant Density	±0.5	1.016	1.001
Cladding Circumferential Temperature Variation	Approximation of wire wrap using bare bundle model	1.024	1.010
Wire Orientation	Reversed wire orientation in center pin	1.01	1.003*

** 7-pin bundle simulation*

NEAMS

NUCLEAR ENERGY ADVANCED MODELING & SIMULATION PROGRAM

Computed Hot Channel Factors

Cladding HCF ΔT = Clad Outerwall – Clad Innerwall	Uncertainties (3σ) %	EBR-II Legacy	AFR-100 SHARP
Cladding Thickness	±3	1.03-1.05	1.018
Cladding Thermal Conductivity	±7	1.088	1.082
Fissile Maldistribution	±6	1.06	1.036

Fuel HCF ΔT = Fuel Outerwall – Fuel Centerline	Uncertainties (3σ) %	EBR-II Legacy	AFR-100 SHARP
Fuel Thermal Conductivity	±25	1.25	1.226
Fissile Maldistribution	±6	1.06	1.016

Axial enrichment zoning in AFR-100 is likely to yield different HCF from EBR-II

Conclusions on HCF Evaluation

- **Successfully evaluated HCFs for AFR-100 with advanced NEAMS tools**
  - One way coupling sufficient (Neutronics to T/H)
  - Bare bundle models are conservative, save cost, and more accurate for fuel temperature
- **Advanced HCFs are generally smaller than the legacy HCFs**
  - Uncertainties involved in M&S were reduced or eliminated, leader to greater confidence in the SHARP result and removal of over-conservatism
  - However, differences can/should appear for different reactors
- **M&S capability developed in this work is applicable to other metallic fuel SFRs (i.e., VTR) and LFRs for evaluation of various HCFs**

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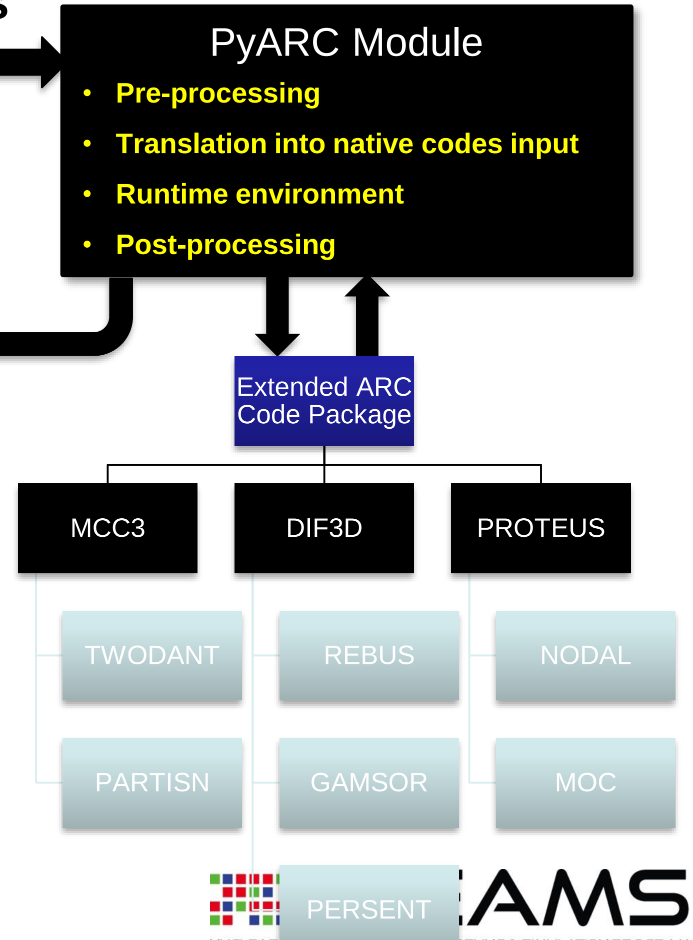
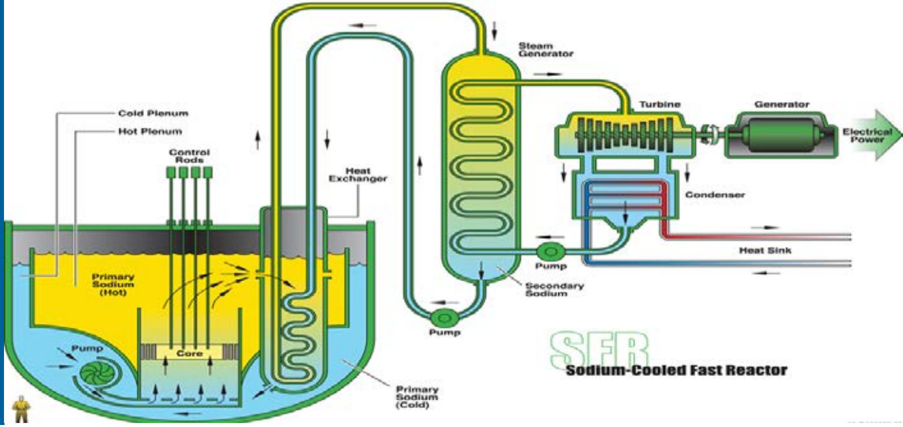
Other Notable FR Capabilities

- **High fidelity tools (PROTEUS, Nek5000) scale to full core for computation of targeted local quantities (demonstrated for SFR – ducted assembly)**
- **Workbench GUI / analysis tool for fast reactors**



Materials Development for Sodium-Cooled Fast Reactors (SFRs)



Meimei Li
Nuclear Materials Group
Nuclear Science and Engineering Division
Argonne National Laboratory

Fast Reactor Working Group (FRWG)/DOE Fast Reactor Workshop,
ANL, July 9-10, 2019

SFR Operating Condition and Material Requirement

- **550°C outlet temperature**
  - Materials must have adequate high temperature strength and ductility
- **Sodium coolant**
  - Materials must be compatible with sodium environments
    - Alloying element dissolution
    - Oxidation
    - Carburization/decarburization

"Go/no-Go"

- **60-yr design life (500,000 h)**
  - Materials must have long-term stability
  - Lifetime irradiation dose 10-15 dpa: radiation resistance for core internals
- Other considerations: manufacturing, welding, etc.

Increased temperature

Materials	Characteristics
Ferritic/Martensitic (F/M) Steels	• Well-established and proven fast reactor material. Low swelling and better thermal properties compared to austenitic SSs. • Lower high temperature strength.
Austenitic Stainless Steels	• Well-established and proven structural material. • Greater swelling rates than F/M steels at high fluences. May not be a critical factor for most structural applications.
Superalloys	• Superior high temperature strength and creep performance over traditional SSs and good heat transfer properties. • Phase instability, swelling, and irradiation embrittlement, and high cost (due to high Ni contents).
Refractory Alloys	• Very high temperature performance, good liquid metal compatibility, commercially available. • Difficulties in joining, sensitivity to impurities, irradiation embrittlement, and high cost. • Not code qualified
Ceramics	• Very high temperature performance and good thermal properties. • Difficulties in joining, manufacturing, compatibility and high cost. • Not code qualified

Increased cost

Materials Used in SFRs

- Austenitic stainless steels: 304, 316
- Ferritic steels: 2.25Cr-1Mo, Mod. 9Cr-1Mo

	Vessel	Piping	IHX	Steam Generator	
				Evaporator	Superheater
EBR-II	304 SS	304 SS	304 SS	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
Fermi-I	304 SS	304 SS	316 SS	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
FFTF	304 SS	316 SS	304 SS	-	-
BN-600	304 SS	304 SS	304 SS	Fe-2¼Cr-1Mo	304 SS
SPhenix	316L(N) SS	304L(N) SS	316L(N) SS	Alloy 800	
PFR	321 SS	321 SS	316 SS	Fe-2¼Cr-1Mo	316SS/9Cr-1Mo
PFBR	316 SS	316 SS	316 SS	Modified 9Cr-1Mo	

SFR New Alloy Development – Downselection and Performance Verification

2008 Established Alloy Development Priority List	2009-2012 Alloy Downselection	2013-2015 Verification of Enhanced Properties
• Considered a large class of structural materials for further development • Involved 5 U.S. national Laboratories and 5 U.S. universities • Considered experience from Fusion, Gen IV, Space Reactor, and development activities in Fossil Energy • Established alloy development priority list: – Ferritic-Martensitic steels • Grade 92 (NF616) • Grade 92 with thermo-mechanical treatment (TMT) – Austenitic stainless steels • HT-UPS • NF-709	• Established comprehensive downselection metrics • Considered tensile properties, creep, creep-fatigue, toughness, weldability, thermal aging, sodium compatibility, mechanical and TMT processes • Integrated R&D activities by DOE Labs – Oak Ridge National Laboratory – Argonne National Laboratory – Idaho National Laboratory • Materials considered include – Optimized-Gr92, Ta/Ti/V-modified 9Cr, Gr92, Gr91 (baseline material) – HT-UPS (Fe-14Cr-16Ni), Modified HT-UPS, A709 (Fe-22Cr-25Ni), 316H (baseline material) • Based on overall performance w/ comprehensive metrics (and accelerated test data), Optimized-Gr92 with TMT and A709 were downselected for further assessment	• Further optimize mechanical and TMT processes • Procure larger heats • Validate performance gains • Longer-term testing of base metals and weldments • Irradiation campaign planning • Development of roadmap for ASME nuclear code cases

SFR New Alloy Development – Qualification

- Alloy 709 has nearly doubled the creep strength of 316SS and overall better performance in SFR environments.
- Next step is to qualify A709 for its use at various design phases of a demonstration plant, and eventually a commercial plant.
  - Generate up to 100,000 h property data aimed for a 500,000 h design life
  - Use a staged approach

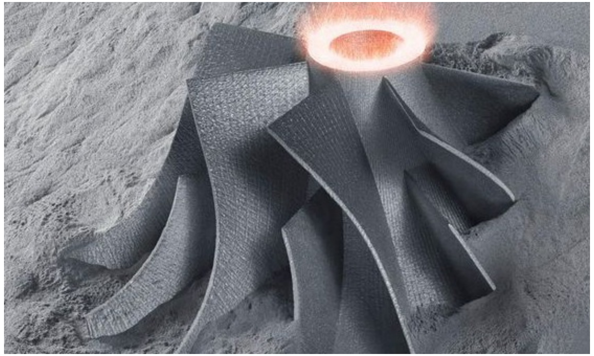
1st commercial heat of A709

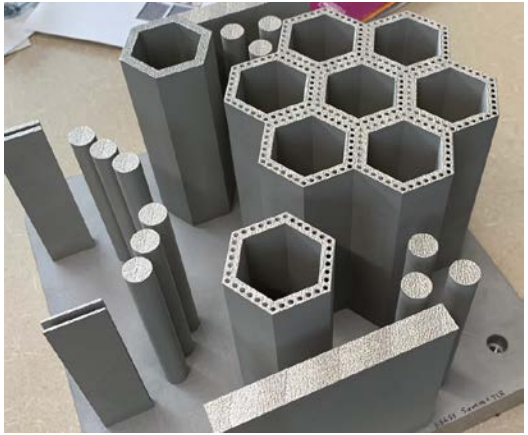
Material Code Qualification and NRC Licensing Need

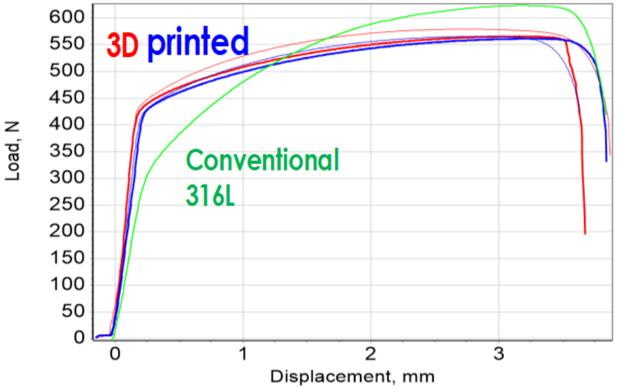
- **ASME Section III Division 5**: Rules for Construction of High Temperature Reactors, including gas-, metal and salt-cooled reactors
  - Rules for metallic components
  - Rules for graphite and ceramic composites (SiC-SiC)
- **Five Qualified High-temperature Materials**
  - 304SS, 316SS, 2.25Cr-1Mo, Mod.9Cr-1Mo, Alloy 800H
- **Code Qualify New High-Temperature Alloy**
  - Alloy 709
- **NRC Licensing Need**
  - ASME Code does not address environmental effects (corrosion, radiation)
  - Understand and predict environmental effects in G91 and A709

Metal Additive Manufacturing

- Advanced manufacturing has the potential to reduce cost and deployment timelines
  - o Advanced designs
  - o High-performance materials







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Molten Salt Corrosion

Jinsuo Zhang, Virginia Tech

Induction

Progress in Materials Science 97 (2018) 448–487

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Corrosion in the molten fluoride and chloride salts and materials development for nuclear applications

Shaoqiang Guo[a,*], Jinsuo Zhang[a,b,*], Wei Wu[b,c], Wentao Zhou[b]

[a] Nuclear Engineering Program, Department of Mechanical Engineering, Virginia Polytechnic Institute and State University, Blacksburg, VA 24061, United States
[b] Nuclear Engineering Program, Department of Mechanical and Aerospace Engineering, The Ohio State University, Columbus, OH 43210, United States
[c] School of Chemical Engineering and Technology, Xi'an Jiaotong University, Xi'an 710049, PR China

ARTICLE INFO

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Chloride salts
Molten salt reactor
Spent fuel reprocessing

ABSTRACT

Next-generation nuclear reactor concepts and advanced techniques for reprocessing spent nuclear fuel (SNF) are drawing great attention in the nuclear field. Molten halide salts have been proposed as the fuel solvent and coolants for many molten salt reactor (MSR) concepts, and the electrolyte for the electrochemical separation of the SNF. The major concern of using molten salts is the corrosion of the structural materials imposed by these extreme environments. Materials corrosion is more challenging in the molten salt nuclear systems than in the traditional water reactors as the formation of the passivating oxide layer on the corrosion resistant alloys becomes thermodynamically unfavorable in molten salts and the use of many corrosion resistant alloys is restricted. This review takes a comprehensive approach covering all relevant work in the field: corrosion data accumulated since the 1950s to date, major corrosion problems and corresponding mechanisms, metallurgical factors, historical development of corrosion resistant alloys and recent attempts. The key environmental factors influencing corrosion in various nuclear systems, electrode kinetics, thermodynamic properties, and corrosion prevention techniques are also reviewed. Finally, current progress and challenges are summarized with an attempt at identifying knowledge gaps and future research directions.

Corrosion reaction

- Corrosion is an electrochemical process involving the anodic metal dissolution and cathodic reduction of oxidants

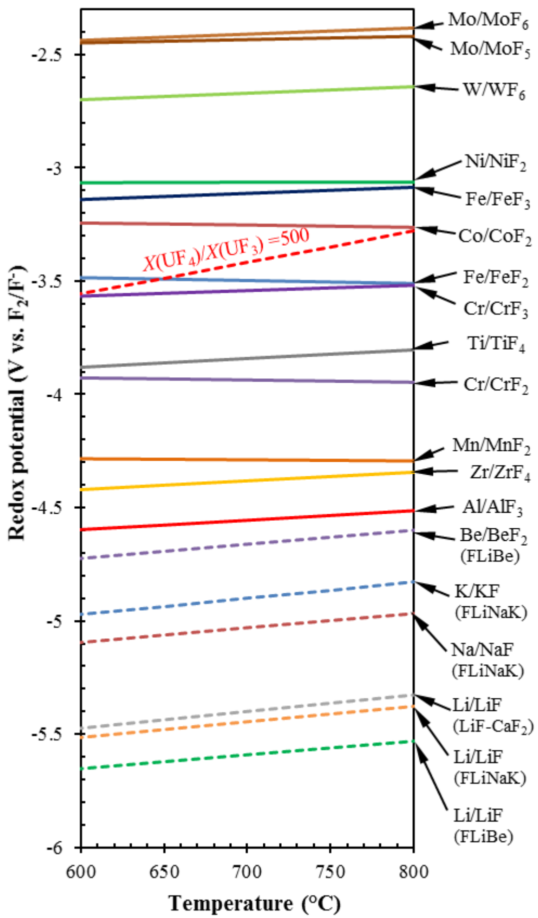
$$\mathrm{M} \rightarrow \mathrm{M}^{\mathrm{n}+} + \mathrm{ne}^{-}$$
$$\mathrm{Ox} + \mathrm{ne}^{-} \rightarrow \mathrm{Red}$$

- To make the reaction occur spontaneously

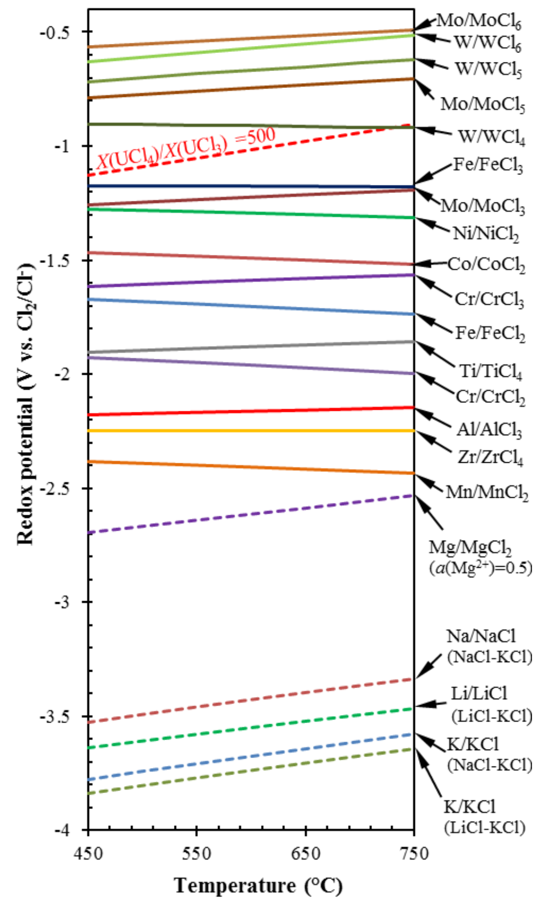
$$E_a < E_C$$

Where $E_{\mathrm{a}} = E_{\mathrm{a}}^{\circ} + \frac{RT}{nF}\ln\frac{a_{\mathrm{M}^{\mathrm{n}+}}}{a_{\mathrm{M}}}$ & $E_{\mathrm{c}} = E_{\mathrm{c}}^{\circ} + \frac{RT}{nF}\ln\frac{a_{\mathrm{Ox}}}{a_{\mathrm{Red}}}$

Redox potentials



Fluoride salts

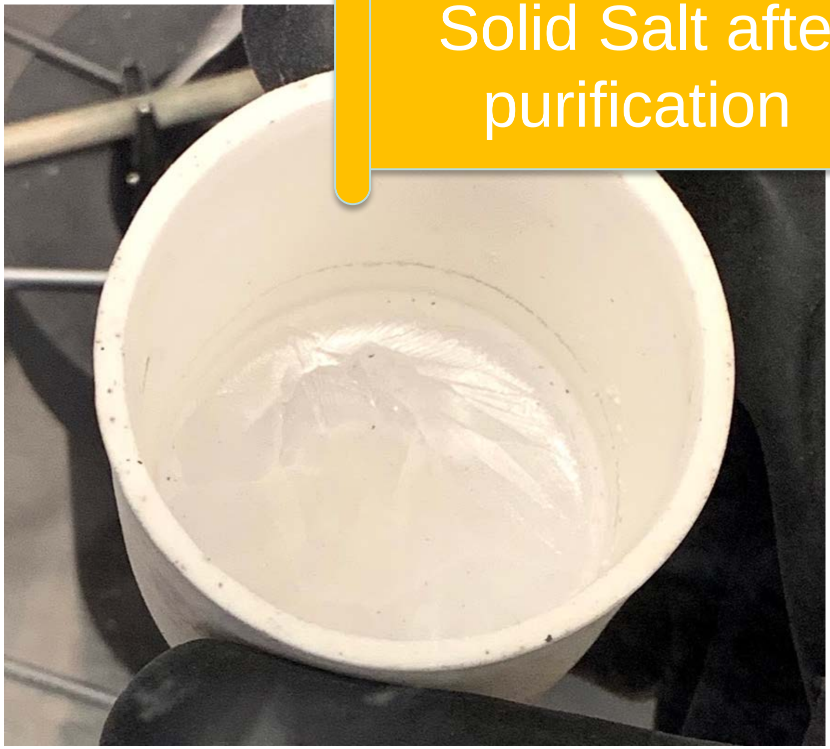


Chloride salts

A Short Summary

- Metal impurity can be a corrosion driver
  - Can be removed by electrochemical Methods
- No metal impurities (O, OH^-, H_2O)
  - can be removed by thermal purification and Chemical purification

After purification [KCl-NaCl-MgCl2 (44.7 mol%)]



Solid Salt after purification

After the Purification:

1) The salt is very clean (no black stuffs).
2) No salt creeping issues
.

Salt Details

- The high purity salt were obtained from the following suppliers,

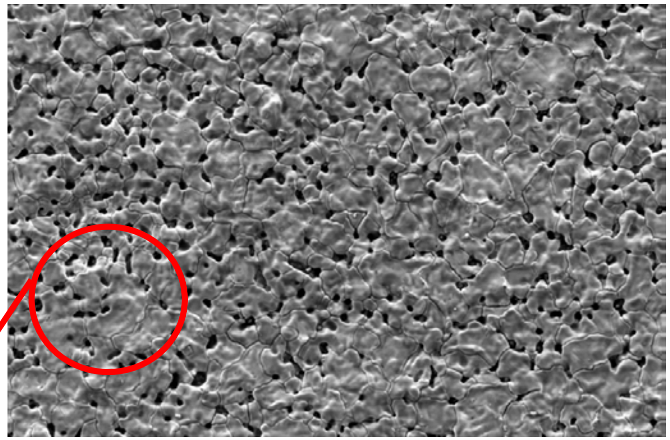
  KCl (>99%) and NaCl (99.999%)

  $MgCl_2$ (>98% with <2% moisture)
- The salt used for the corrosion tests had the following composition

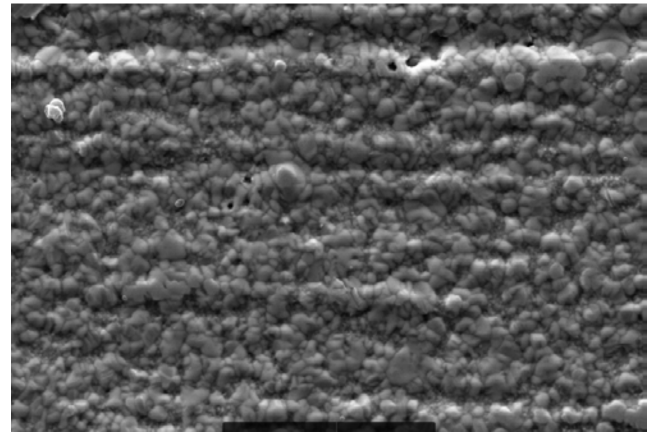
  KCl (45wt%)-$MgCl_2$ (53wt%)-NaCl (2wt%)
- Furthermore, the ternary salt mixture was also obtained from Israel Chemicals Limited (ICL).
- The ICL salt was purified and used for the corrosion test while the HP salt was used to check the effectiveness of the purification process and how it affects the corrosion in alloys.
- Three alloys were tested C276, H230, Alloy 709

SEM Surface Images: H230 Alloy

Pitting

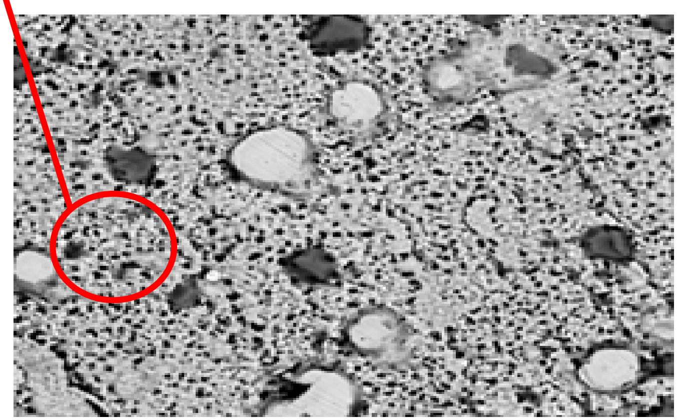


HP salt not treated with Mg

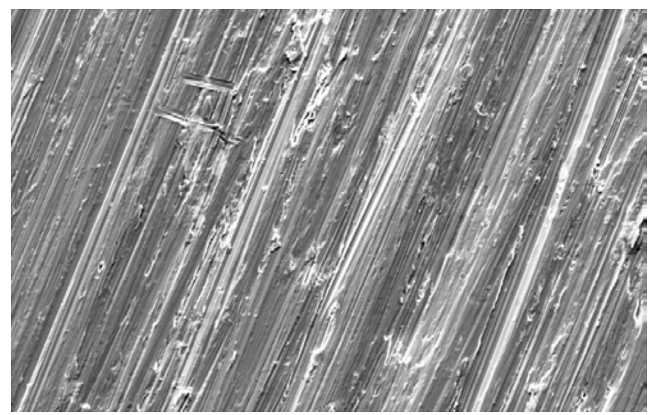


HP salt treated with Mg

The alloy was not corroded

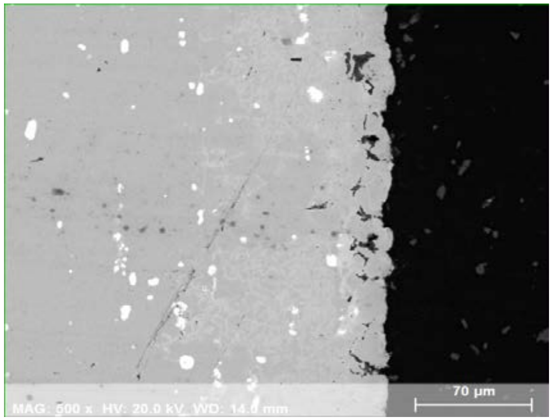


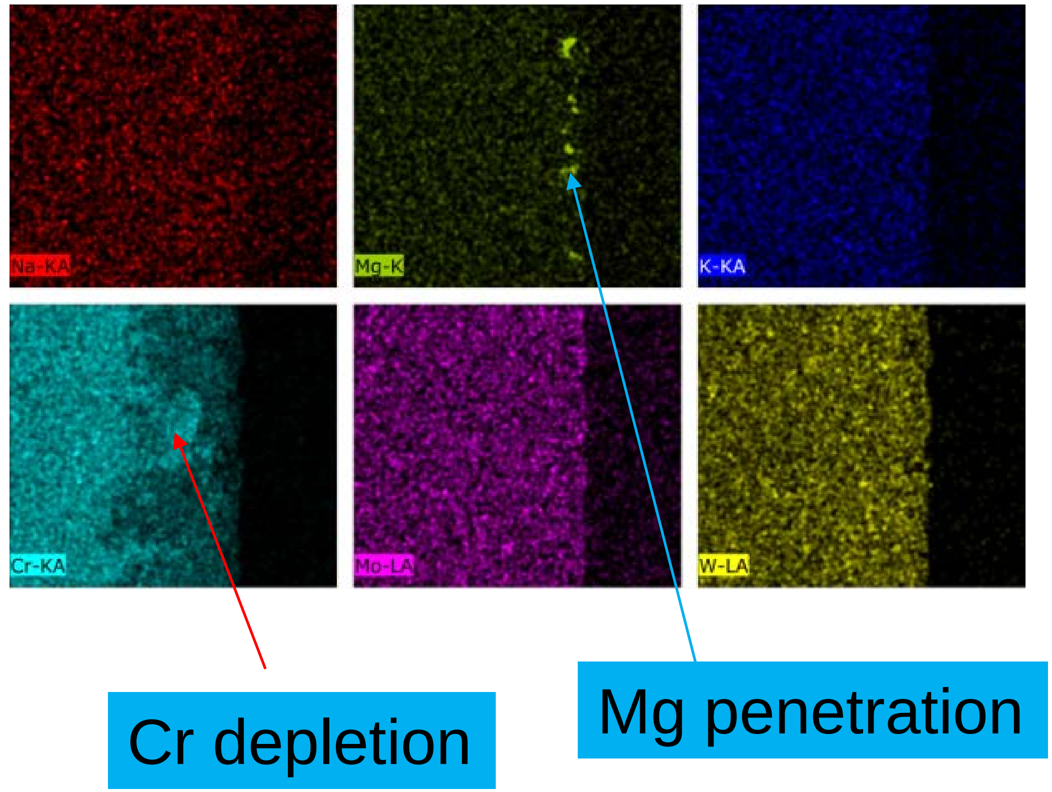
ICL salt



Raw Specimen

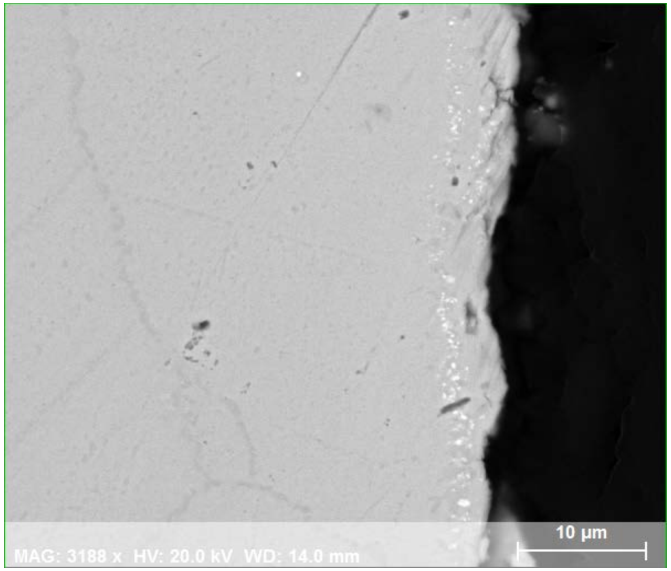
Cross-section of H230(HP salt without Mg Purification)

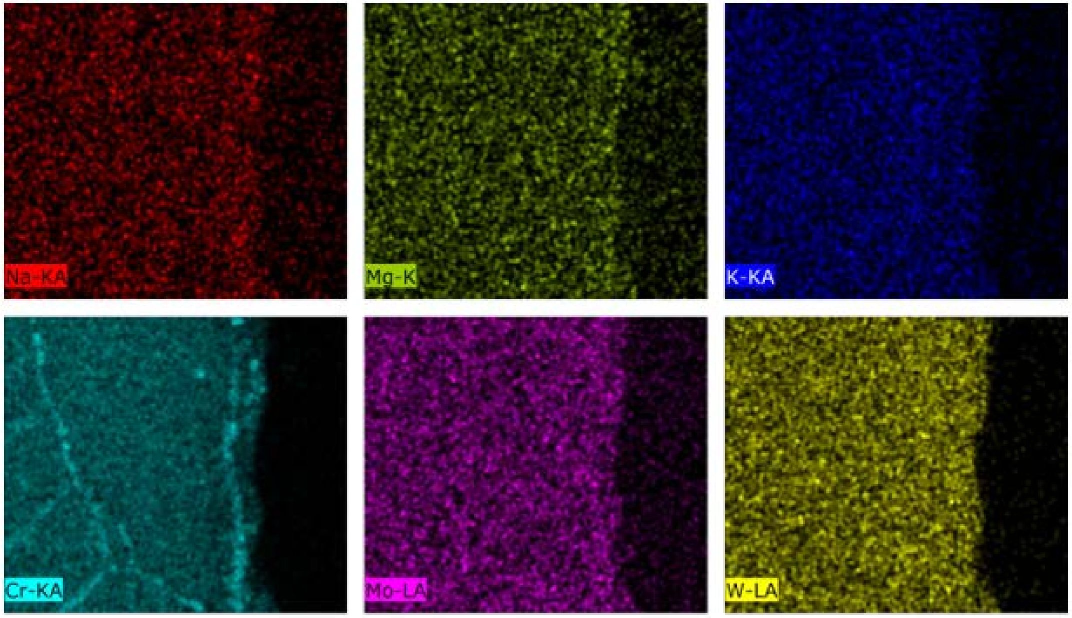




- Maximum depth of attack ~37 um.
- Extensive Cr depletion along the boundary
- Deposition of Mg (Either oxide or chloride) in the pores created due to corrosion.

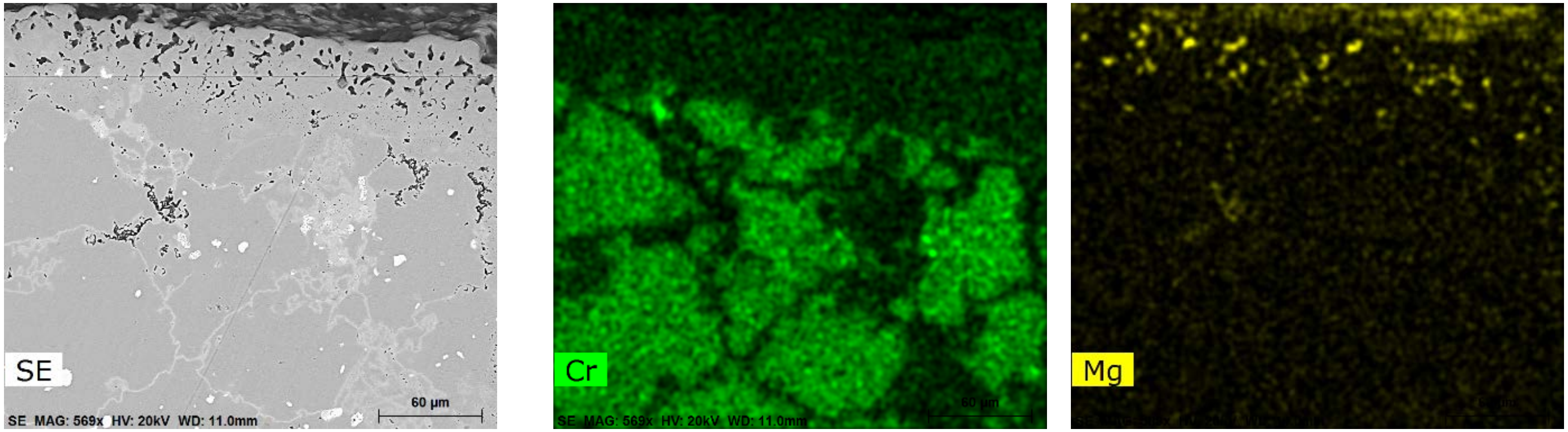
Cross-section of H230(HP salt with Mg Purification)





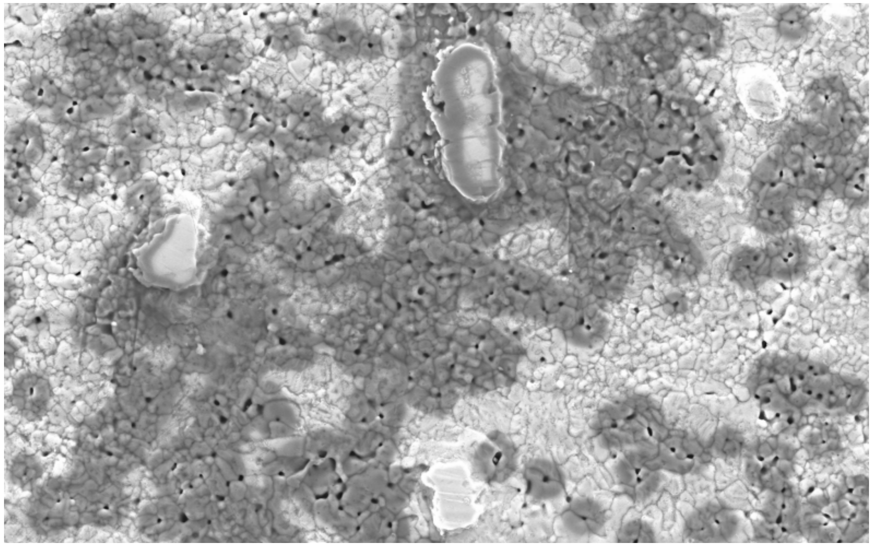
- ❑ Surprisingly, no attack observed.
- ❑ No depletion of Cr or outward diffusion of Ni.

Cross-section of H230(ICL salt without Mg Purification)

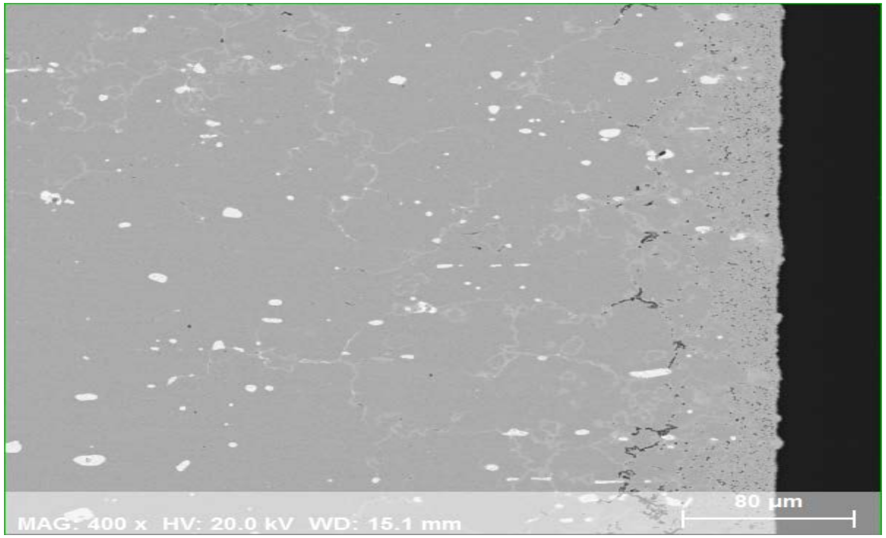


Heavy Cr depletion and Mg penetration

Vapor Corrosion H230



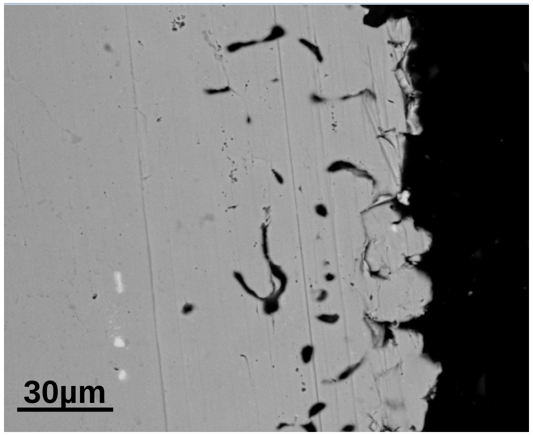
Surface View



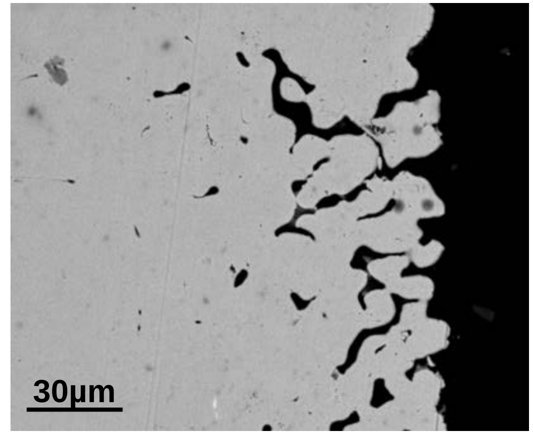
Cross section View

- Similar behavior as observed in immersion test, Pitting on the surface and Cr depletion along the cross section.

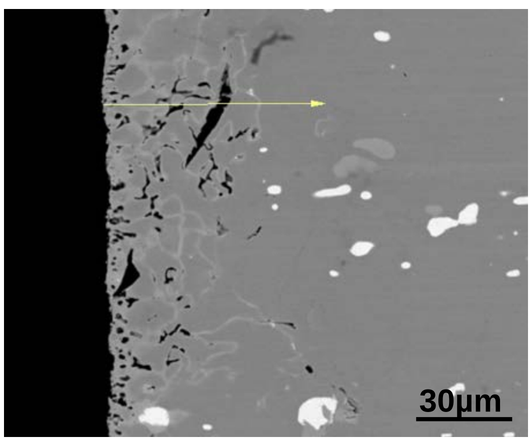
Ni-Base VS Fe-base Alloys



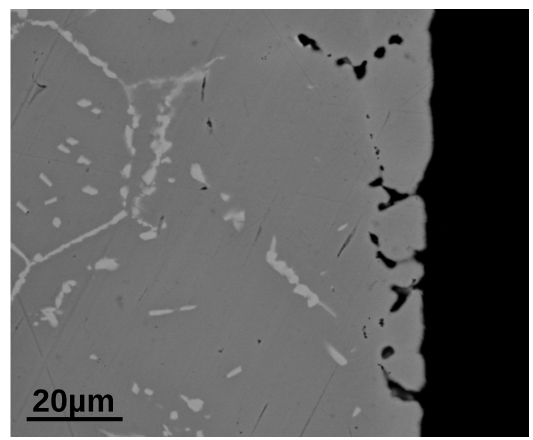
Alloy 709-RBB (Electro slag remelting)



Alloy 709-4B2 (Argon oxygen decarburization)



H230 Alloy

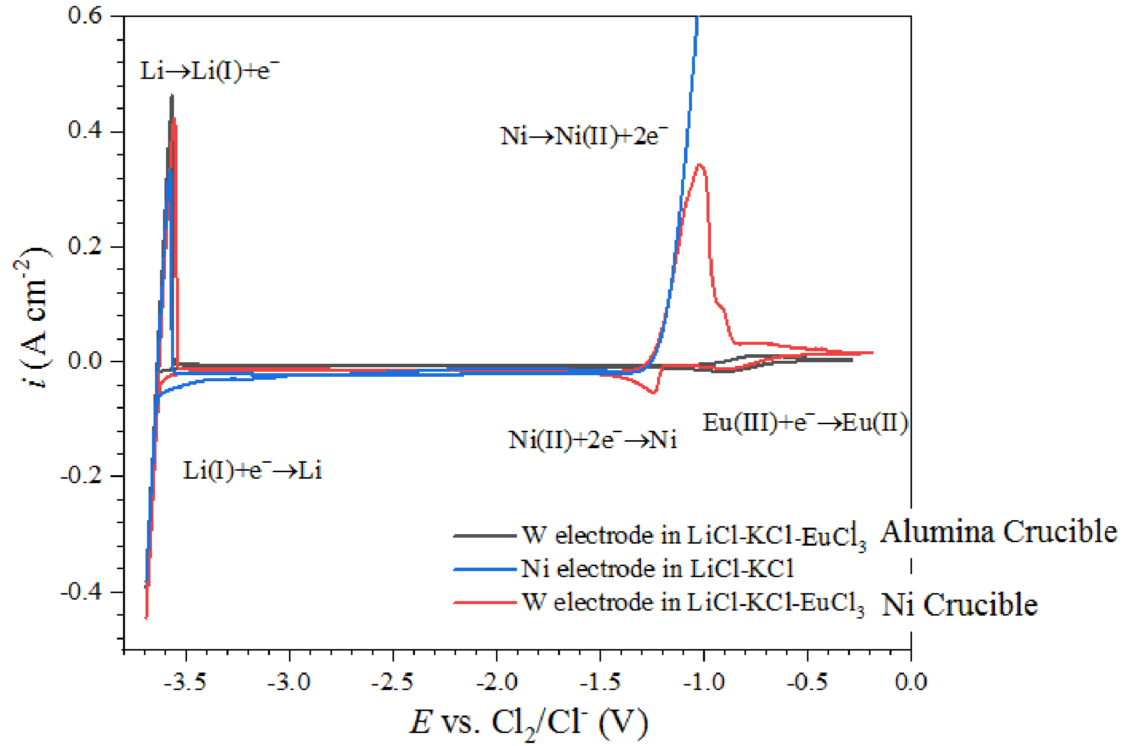


C-276 Alloy

A summary

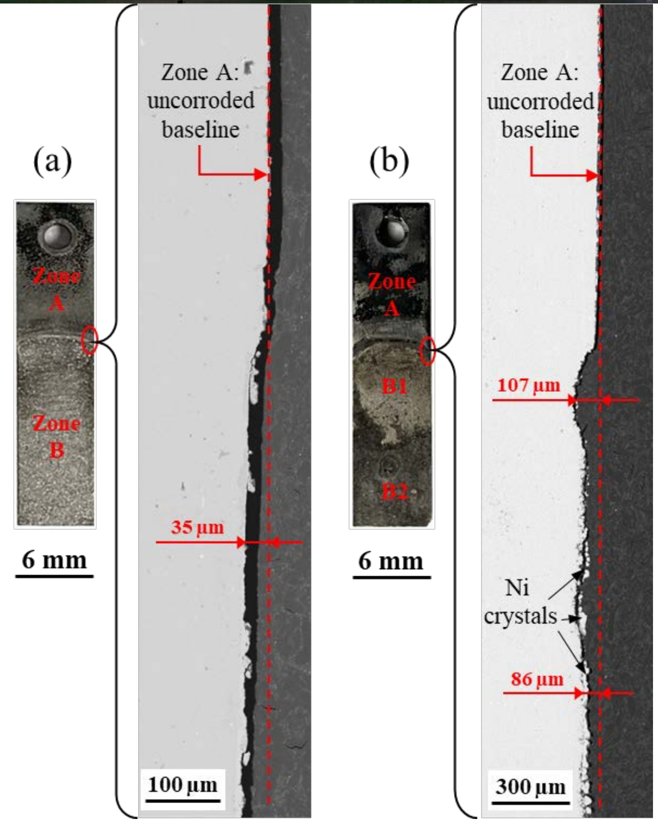
Salt Details	Alloy	Depth of Attack (μm)	Weight Loss (mg)	Corrosion Rate (mg/cm^2)
Immersion Test-ICL Salt	H230	~47	26.8	3.5458
Immersion Test-HP Salt-Without Mg	H230	~37	6	0.7662
Immersion Test-HP Salt-With Mg	H230	No Attack	No Weight Loss	-
Vapor Test- ICL Salt	H230	~64	56.4	7.2096
Immersion Test-ICL Salt	C-276	4-16	6.9	0.7723
Immersion Test-ICL Salt	709-4B2	45-74	162.3	21.0569
Immersion Test-ICL Salt	709-RBB	~40	43.9	5.6528

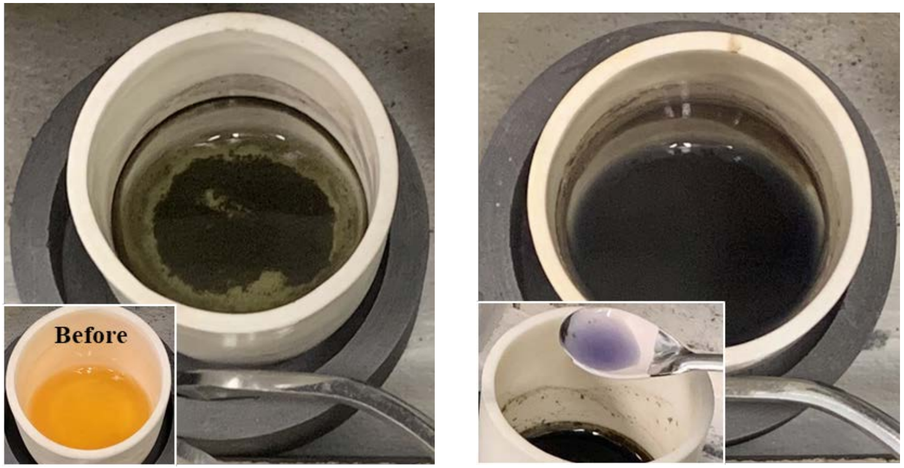
Fission products Eu effects



Cyclic voltammograms obtained in LiCl-KCl (40.5 at%) eutectic and LiCl-KCl-2 wt.%$EuCl_3$ melt contained in pure nickel crucibles. Working electrode is tungsten or nickel rod. Scan rate = 100 mV s^{-1}, and T = 500° C. $EuCl_3$ is selected to accelerate the corrosion

$EuCl_3$ induced corrosion



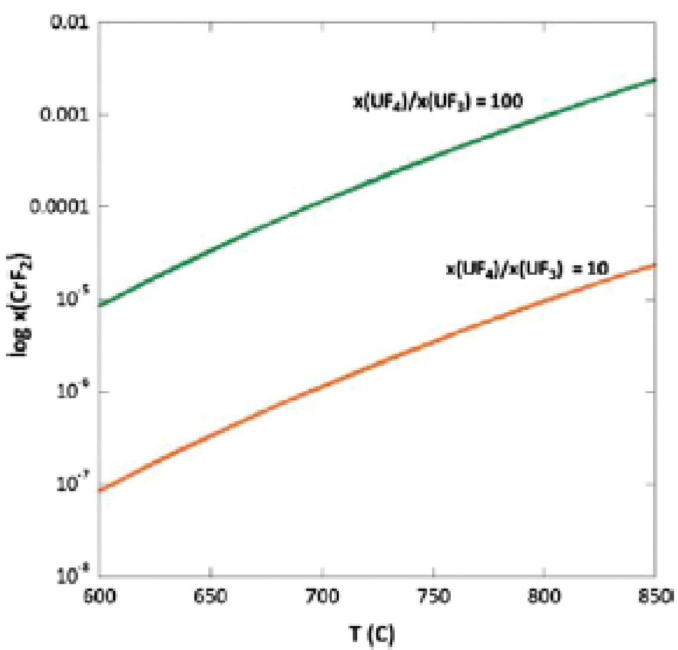


The deposits are oxides

Photos the alloy specimens after ultrasonic clean and the corresponding cross-sectional SEM images after 120 hours (a) Fe-base Alloy 709, and (b) Ni-base Inconel 718.

Molten Salt corrosion-Redox control method

- ❑ Metal/salt control
- ❑ Cover gas control
- ❑ Dissolved-salt control



Corrosion Science 144 (2018) 44–53

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Review

Redox potential control in molten salt systems for corrosion mitigation

Jinsuo Zhang[a,*], Charles W. Forsberg[b], Michael F. Simpson[c], Shaoqiang Guo[a], Stephen T. Lam[b], Raluca O. Scarlat[d], Francesco Carotti[d], Kevin J. Chan[e], Preet M. Singh[e], William Doniger[d], Kumar Sridharan[d], James R. Keiser[f]

[a] Nuclear Science and Engineering, Department of Mechanical Engineering, Virginia Tech, United States
[b] Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, United States
[c] Department of Metallurgical Engineering, University of Utah, United States
[d] Department of Engineering Physics, University of Wisconsin-Madison, United States
[e] School of Material Science and Engineering, Georgia Institute of Technology, United States
[f] Oak Ridge National Laboratory, United States

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ABSTRACT

In a molten salt nuclear reactor system, the redox potential must be controlled for mitigating corrosion of structural materials. The paper presented a critical review on the available knowledge of redox potential control in molten fluoride salt systems. The major phenomena that affect the redox potential and material corrosion are fission, TF production by transmutation, and salt contamination with metal fluorides or other oxidizing impurities. Redox potential control methodologies include gas sparging, contacting the salt with a reducing metal, and adding soluble salt redox buffers to the salt. Redox potential measurement technologies include electrochemical sensors and optical spectroscopy. The paper also analyzed the current technology issues and recommended near future studies.

1. Introduction

The molten salt reactor (MSR) system is one of the Generation-IV reactor concepts [1]. An early MSR concept developed and tested at the Oak Ridge National Laboratory (ORNL) is the Molten Salt Breeder Reactor (MSBR), for which molten LiF-BeF$_2$-AnF$_4$ was selected as fuel and coolant. Recently, there are a number of advanced MSR concepts being considered by many countries that feature molten fluoride salts as the fuel and/or the coolant [2]. Several salt-cooled reactor designs, also known as fluoride salt-cooled high-temperature reactors (FHRs) have been developed in the United States in the past few years. FHR uses clean fluoride salts and solid High-Temperature Gas-Cooled Reactor (HTGR) fuel. There are several alternative coolants being considered but the leading candidate is $^7Li_2BeF_4$. In Europe, Molten Salt Actinide Recycler and Transmuter (MOSART) and Molten Salt Fast Reactor (MSFR) are highlighted for which the fissile materials are dissolved in the fluoride carrier salts (e.g., LiF-(NaF)-BeF$_2$ and LiF-ThF$_4$). A "Thorium Molten Salt Reactor Nuclear Energy System" project was launched by Chinese Academy of Science in 2011, aiming at developing both solid fueled and liquid fueled molten fluoride salt reactors.

From the 1950s through the early 1970s there were large programs to develop MSRs followed by several decades of little activity. There has been a major revival in interest, partly because salt reactors deliver a larger fraction of their heat at higher temperatures than any other class of reactors (Table 1)—a consequence of using a high-temperature liquid salt coolant and the small temperature rise across the reactor core. That implies the ability to deliver higher temperature heat to industry per MWt output and a higher heat-to-electricity efficiency with the ability to efficiently couple to gas turbines and other advanced power cycles. This includes Brayton cycles that operate at base-load with power peaking using natural gas with incremental heat-to-electricity efficiencies near 70% [3].

In general, molten salt systems have many attractive features including low-pressure operation, efficient high-temperature power cycle, and passive heat rejection. The properties of the molten salts contribute to the design simplicity, inherent safety, and economic competitiveness of the various classes of molten salt systems.

More recently there has been a rapidly growing interest in molten chloride fast reactors (MCFRs) based on the recent understanding that a MCFR with ^{37}Cl will have a very high breeding ratio enabling a breed-and-burn reactor. These reactors typically contain chloride salts with sodium, uranium and other components. The high neutron absorption

Conclusion

- Both metal and non-metal impurities can induced corrosion
- Corrosion can be controlled and mitigated through salt purification and salt redox control
- Salt Vapor also leads to materials corrosion
- Some Salt components can penetrate into the alloy
- Fission products.

Virginia Tech
Invent the Future

Materials Compatibility Studies for LFR

Jinsuo Zhang, Virginia Tech
July, 10, 2019

Corrosion Mechanisms-Two Fundamentals

- Liquid metal corrosion
  - Physical dissolution

$$M(solid) \rightleftharpoons M(LM)$$

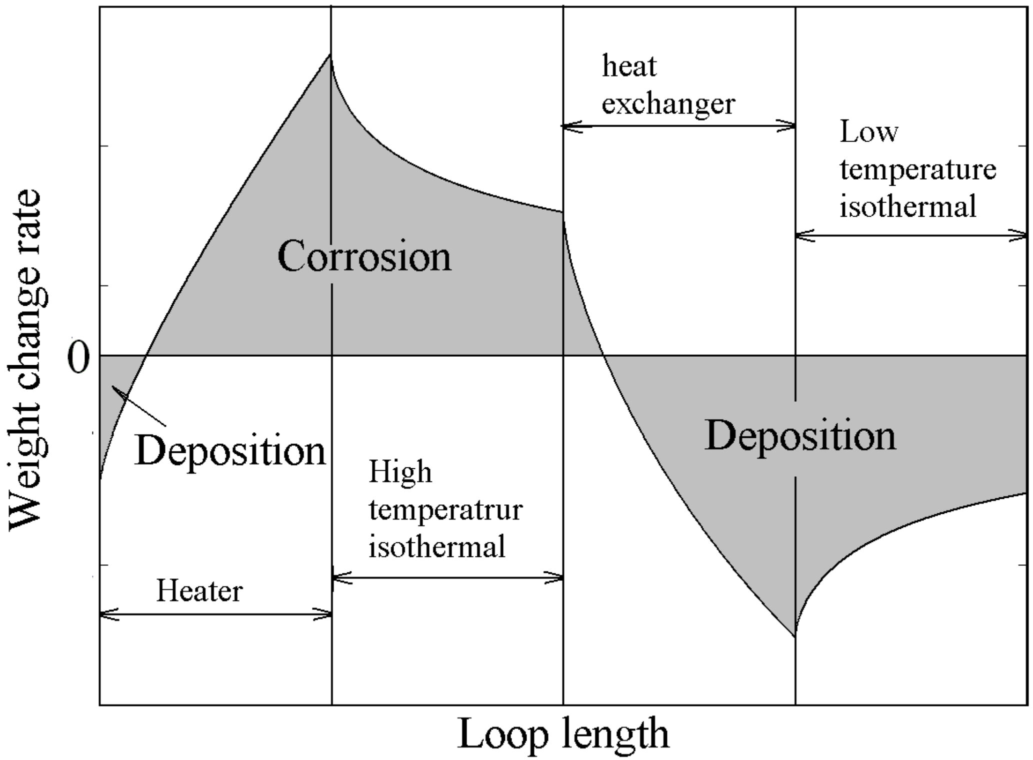
- Molten salt/aqueous corrosion
  - Physical dissolution

$$M - ne \rightleftharpoons M^{n+}$$

$$Ox + ne \rightleftharpoons Red$$

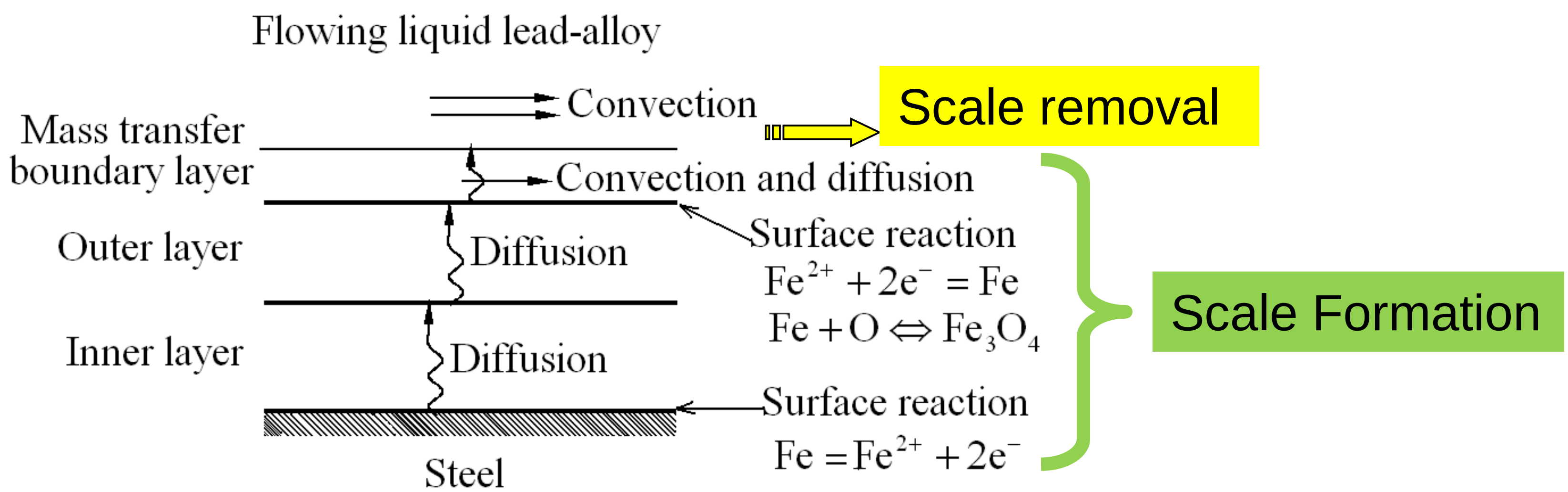
Corrosion Mechanisms-Two Types of Mass transfer

- Temperature-gradient mass transfer
- Dissimilar-metal mass transfer



$$\int_0^L R_w(x)P\,dx = 0$$

Corrosion Mechanisms-Two Processes



How to Mitigate Corrosion –Two Methods

- Corrosion Inhibitor
  - Metallic inhibitors (Zr, Ti, et al)
  - Non-metallic (Oxygen)
- Changing the composition of the alloy or alloy surface (Fe-base, Ni-base, Si-contained alloys)

Metallic Inhibitor

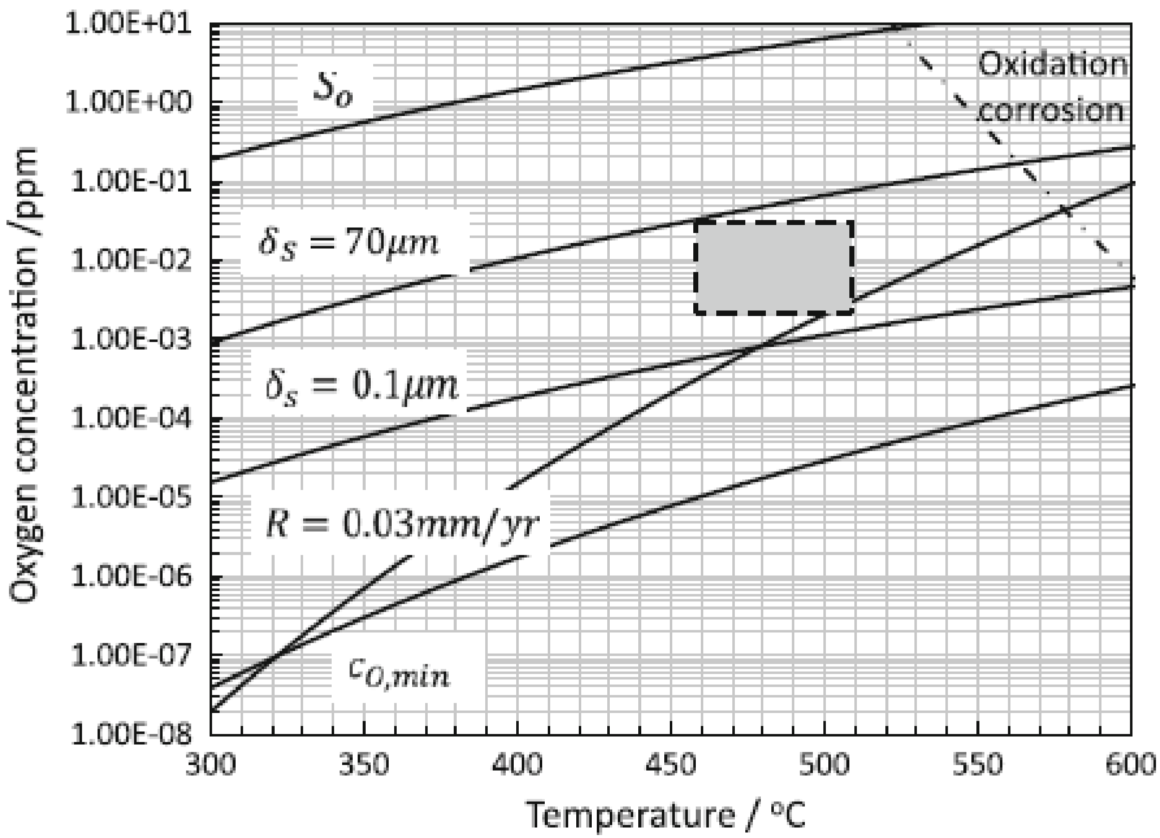
Temperature (K)	Test time (h)	Inhibitor condition	Corrosion results
773-898	<1000	No inhibitor	Severe corrosion
673-823	1000-5000	No inhibitor	Severe corrosion
898	<100	No inhibitor	Severe corrosion
623-923	<5000	Ti added	No corrosion
773-923	<10,000	Zr added	No Corrosion

Corrosion of Croloy 1-1/4 steel in flowing LBE, (Park, et al, Nucl Eng Des. 196, 315, 2000)

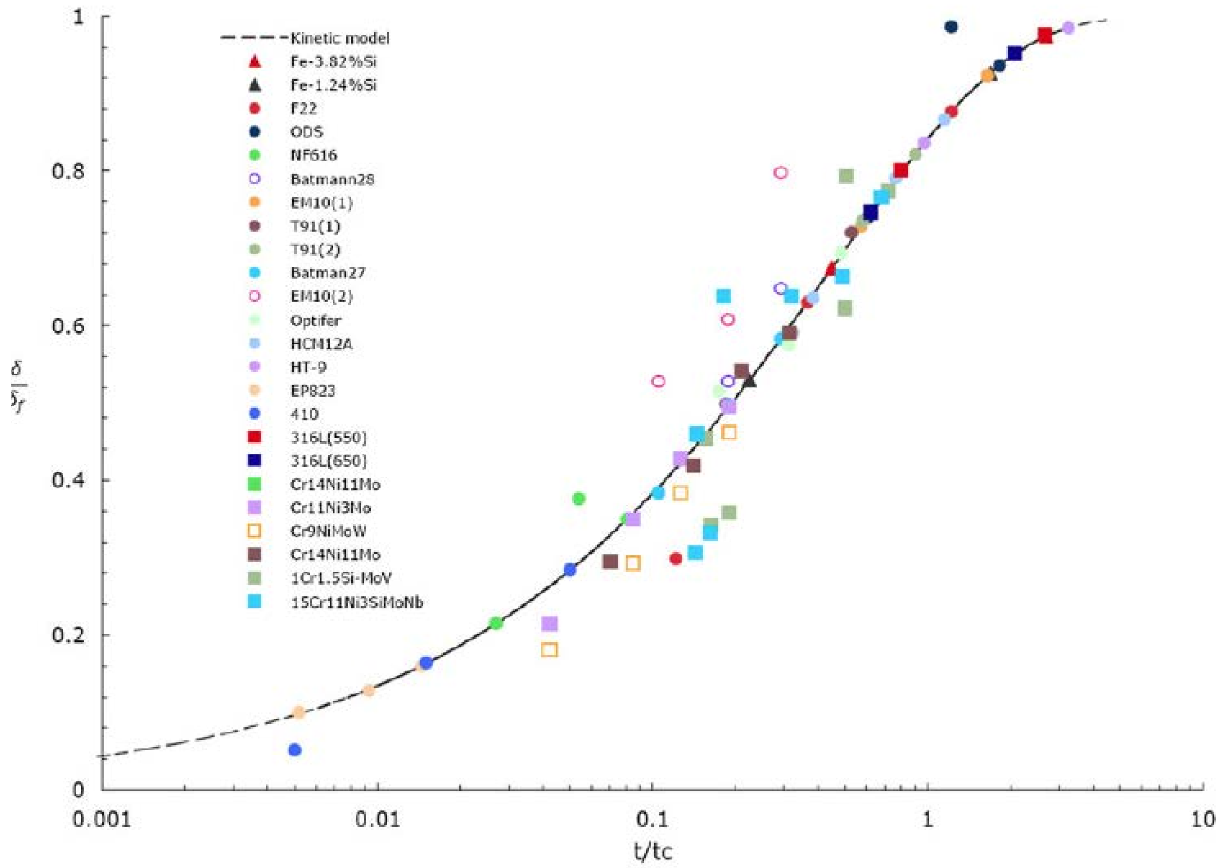
Carbon (%)	Nitrogen (%)	Total C+N (%)	Corrosion, weight loss (mg)
0.0041	0.0288	0.0698	170
0.085	0.0288	0.1138	30
0.234	0.0288	0.2628	<1
0.0041	0.385	0.0426	141
0.0041	0.3100	0.3141	33

Corrosion of Fe with different C and N content in LBE with metallic inhibitor at 1023 K, (Trotrman, J Iron Steel Inst, 194, 319, 1960)

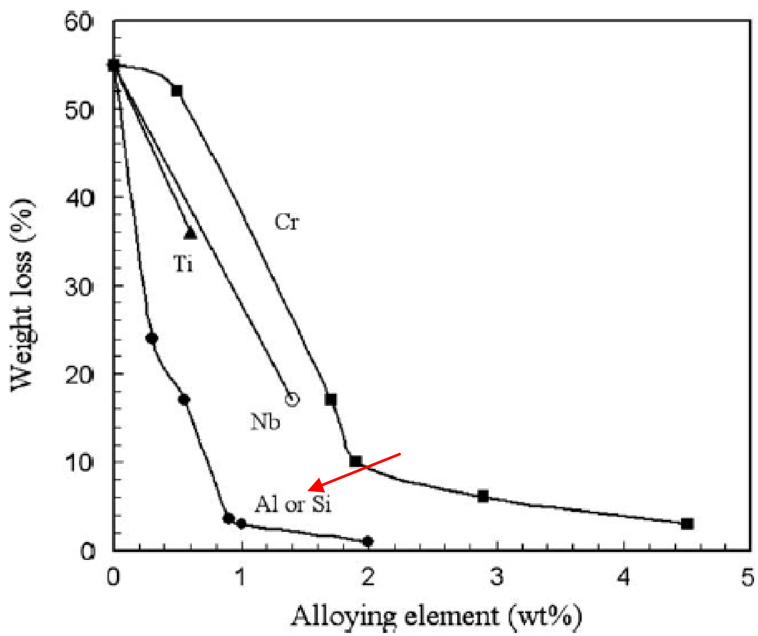
Oxygen Concentration Range



Fe-Base Stainless Steel

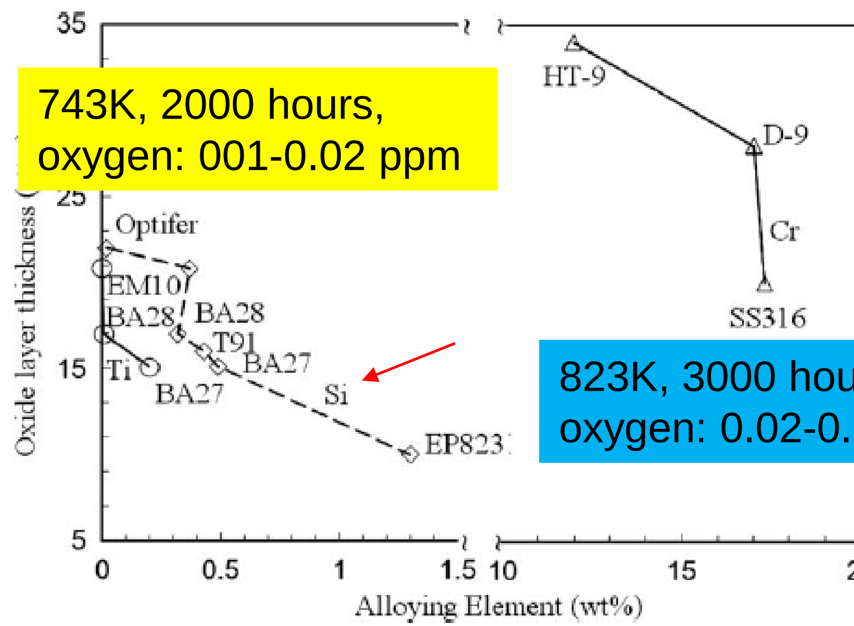


Fe-Base Alloy-Alloy Composition



Low Alloy, Low oxygen, 873K, 6 m/s

*Gorynin, et al, **HLMC-98,** p120*



743K, 2000 hours, oxygen: 001-0.02 ppm

823K, 3000 hours, oxygen: 0.02-0.03 ppm

Stainless steel, high oxygen, 1.9-2 m/s

Fe-Base Stainless Steel

- Direct dissolution of steels in LBE and Pb is too severe for long-term applications without protections
- For unprotected steels, higher solubility of Fe in Bi leads to a factor of 3~10 higher dissolution rate
- **550ºC LBE and 650ºC Pb have about the same steel (Fe) dissolution corrosion rates**

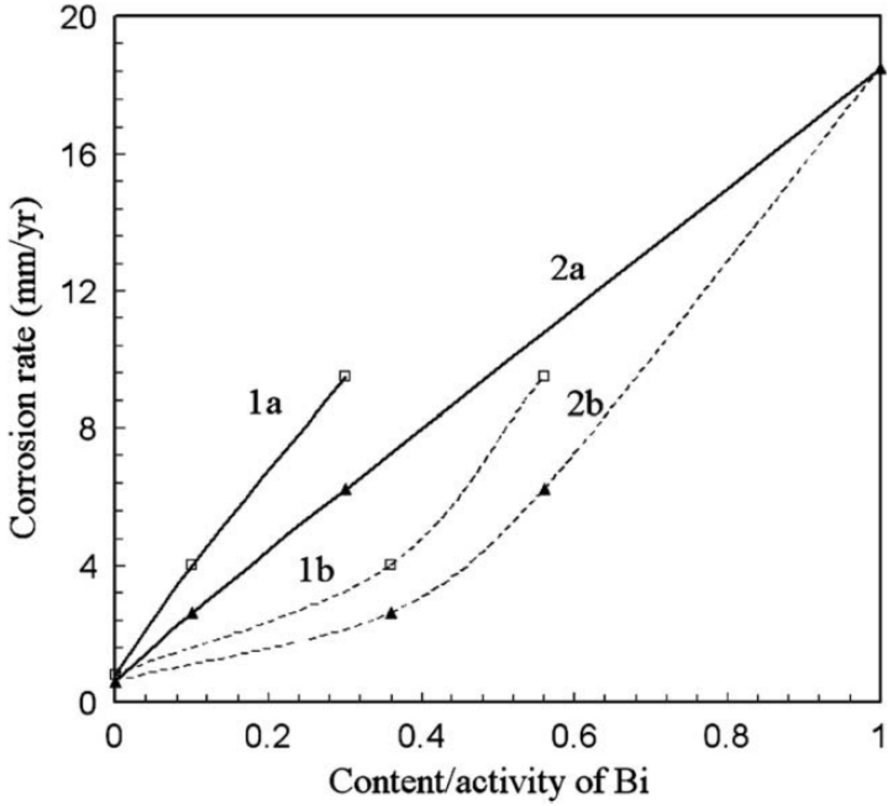
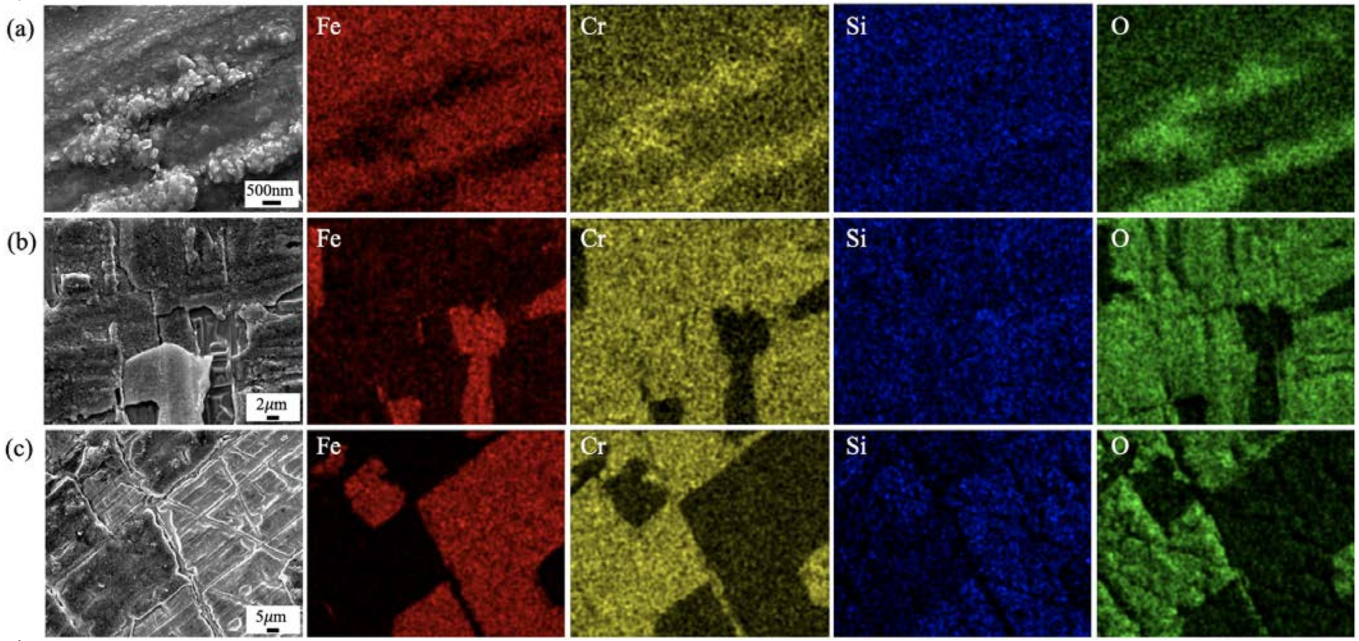


Fig. 2. Corrosion rate as a function of content (1b and 2b)/activity (1a and 2a) of Bi [22]. 1a and 1b: 18Cr–9Ni-type stainless steels; 2a and 2b: low-alloy highest-creep-strength steel 0.5–2.0% CrMoV. The results were from a loop with a maximal temperature of 873 K with a temperature difference 140–150 K and a flow velocity of 1.0–1.5 cm/s. Mg (500 ppm) was added to the liquid to get the oxygen.

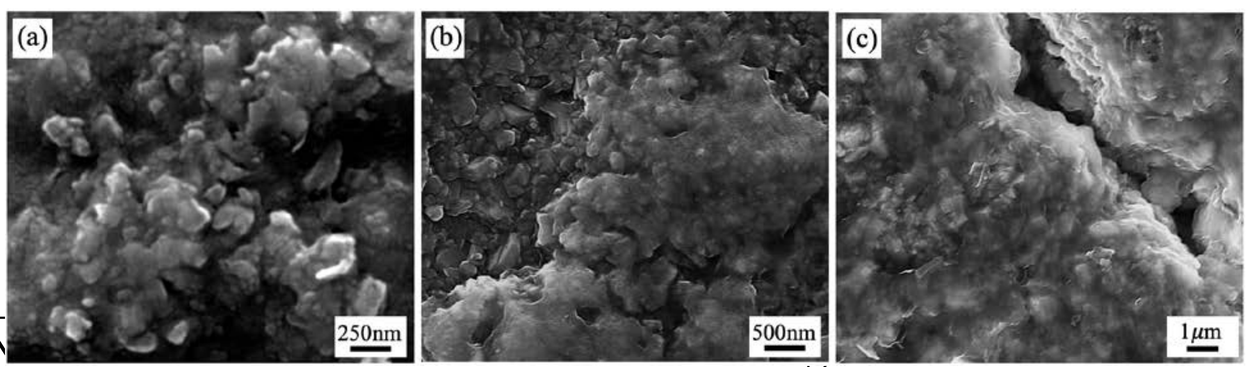
Fe-xCr-2Si Alloy

- MIT's results have shown the alloy has high corrosion-resistance in lead/lead alloy
- The alloy can be used as coating or surface layer of MMLCs
- Our results in high temperature steam has shown the oxide layer is not stable at 1173^{o}C

Fe12Cr2Si-in Steam to oxide layer stability

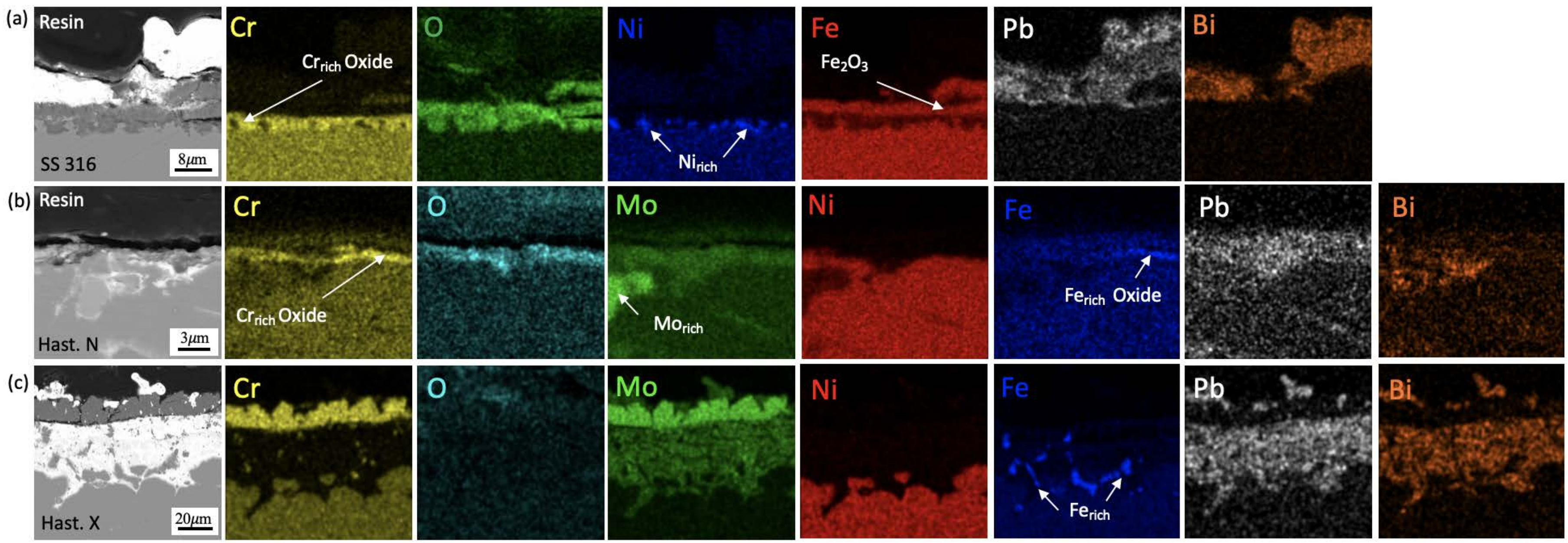


Secondary electron (SE) SEM images with EDS mapping showing the elemental distribution of Fe, Cr, Si, and O on the post-test Fe-12Cr-2Si alloy surfaces at the test temperature of (a) 700°C, (b) 900°C, and (c) 1000°C for 24 hours



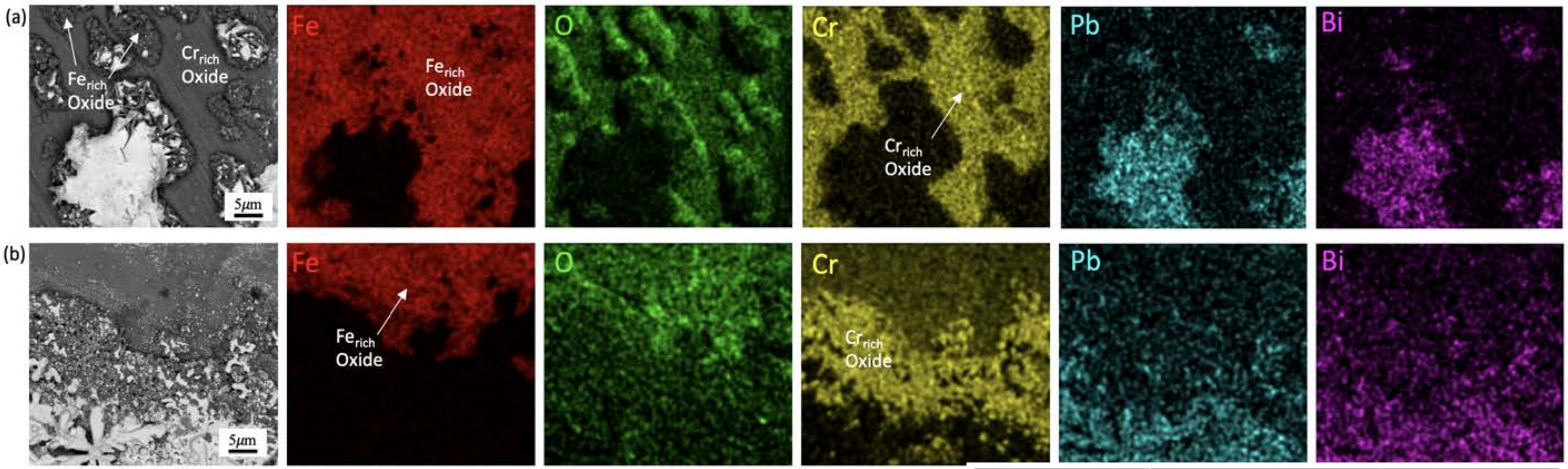
SEM images of Cr_2O_3 formed on the surface of Fe-12Cr-2Si alloy tested at (a) 700°C, (b) 900°C, and (c) 1000°C for 24 hours

Ni-Base Alloy

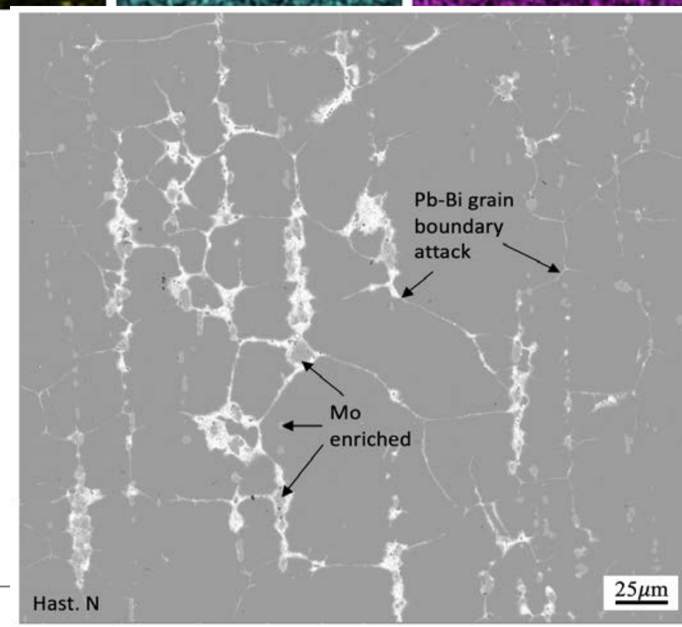


Cross section EDS mapping of (a) SS 316, (b) Hast. N, and (c) Hast. X exposed to Pb-Bi for 24 hours at 600°C

Ni-based Alloy



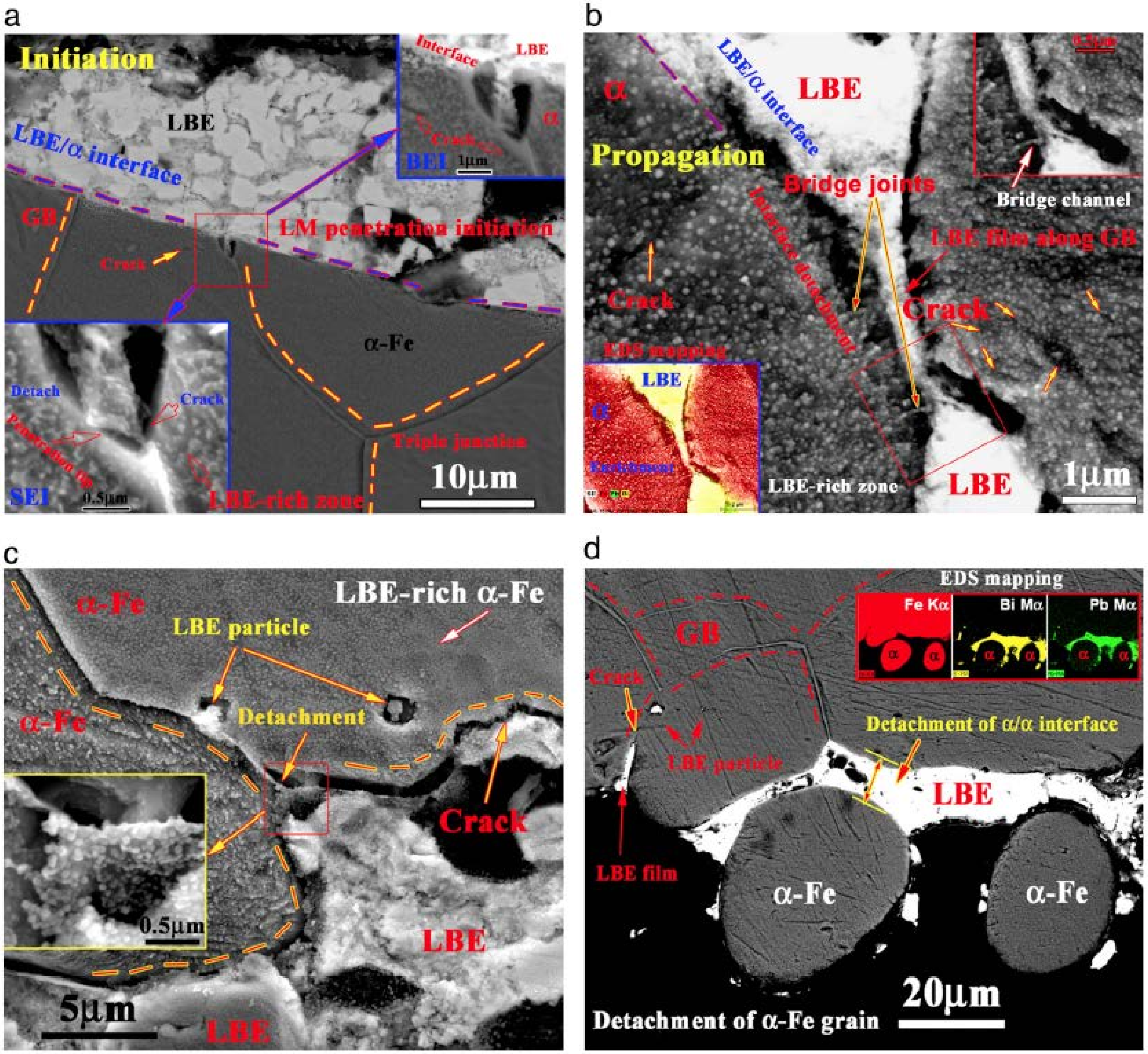
Surface EDS mapping of (a) SS 316, (b) Hast. X exposed to Pb-Bi for 24 hours at 600°C



BSE image of Hastelloy N exposed to Pb-Bi for 24 hours at 600°C, showing Pb-Bi attack along grain boundaries

Embrittlement-LBE

Formation of LBE penetration film and morphology evolution in alpha-Fe at 1173K



Future….

- Embrittlement
- Cracking
- Oxygen control
- Metal purification
- New Alloy Development

FAST REACTOR DATABASES

TANJU SOFU

ARGONNE NATIONAL LABORATORY

FAST REACTOR WORKING GROUP & DOE FAST REACTOR WORKSHOP
July 10, 2019

U.S. SFR TESTING PROGRAM

Background

Past SFR R&D programs focused on development and demonstration by testing of the concepts with high-burnup fuel as well as inherent and passive safety features that lead to no serious consequences even during unprotected accidents

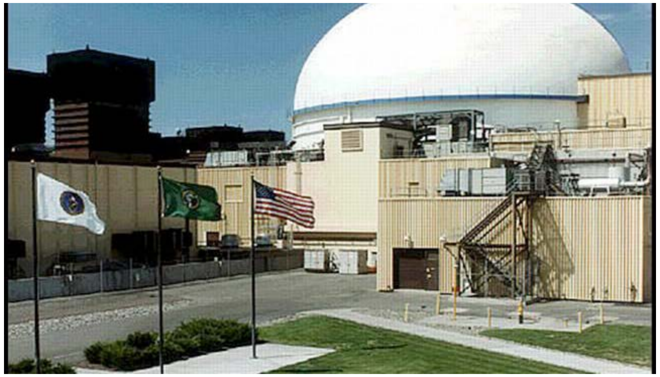
- EBR-II and FFTF metallic fuel irradiation tests
  - Acceptable performance and reliability demonstrated at 10 at.% burnup, with capability established up to 20 at.% burnup
- EBR-II transient testing program
  - Includes landmark EBR-II inherent safety demonstration test
- FFTF passive safety testing program
  - Includes loss-of-flow without scram from half-power, full-flow
- Transient fuel behavior tests:
  - Mild transients on whole fuel assemblies in EBR-II and FFTF
  - Pin disruptive tests on one or a few whole fuel pins in TREAT
  - Lab-tests on segments of fuel pins in the Fuel Behavior Test Apparatus (FBTA) and on whole fuel pins in the Whole-Pin Furnace (WPF) facility

METALLIC FUEL IRRADIATION EXPERIENCE

EBR-II

- Fuel fabrication and design impacts
- Swelling and restructuring vs. burnup
- Influence of high temperatures
- Impact of fuel impurities
- Run beyond cladding breach tests

FFTF



- Fuel column length effects
- Lead metal fuel tests with HT9 cladding
- Commercial metal fuel prototype
- Metal fuel qualification

Reactor	Fuel Type	# of Pins	Clad	Peak burnup
EBR-II	Mark-I/IA (U-5Fs)	~90,000	316SS, D9, HT9	~2.5%
	Mark-II (U-5Fs)	~40,000		~8%
	Mark-IIC/IICS/III/IIIA/IV (U-10Zr)	~16,000		~10%
	U-Pu-Zr	>600		~15-20%
FFTF	U-10Zr	>1050	HT9	~14%
	U-Pu-Zr	37		~9%

EBR-II

Transient testing program

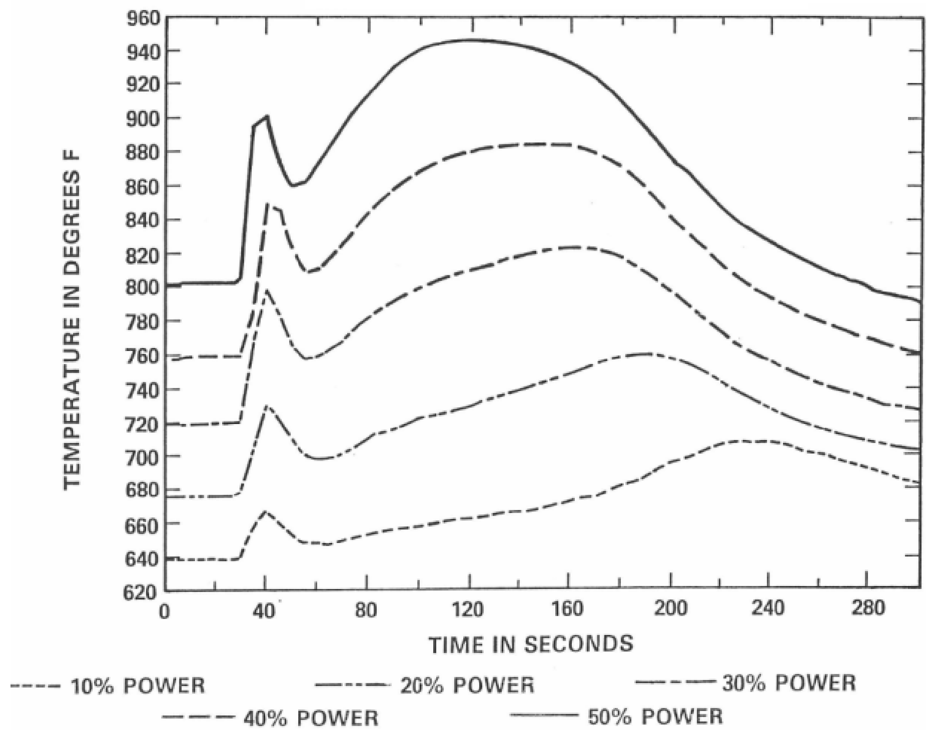
- EBR-II testing program eventually evolved to support assessment of safety performance with emphasis on inherent safety
  - Started with mild natural circulation tests and culminated toward unprotected transients (no scram)
  - I&C system upgraded to measure and collect flow rates and temperatures in the primary, secondary, and steam systems by a data acquisition system
  - Additional control system functions were added to facilitate the conduct of whole-plant dynamic testing

- Over 80 transient tests conducted during 1984-1987 period in several categories:
  - Reactivity feedback characterization tests
  - Loss of flow with scram and transition to natural circulation
  - Loss of flow without scram with different levels of severity
    - Landmark inherent safety demonstration test (station blackout without scram from full power)
  - Dynamic frequency response tests
    - Reactivity perturbation and rod-drop tests
    - Multi-frequency control rod and secondary flow oscillations
  - Loss-of-heat-sink tests (with or without scram)
  - Plant inherent control tests (to demonstrate “load-following” features)

FFTF

Transient testing program

- In late 1980's, a series of passive safety tests were also conducted in FFTF to demonstrate its safety margins
- Of particular interest was a series of Loss of Flow Without Scram tests from power levels up to 50%
  - First series of tests conducted with primary pony motors on so that the minimum flow was ~9% of full flow
  - ULOF tests were then repeated with the same initial conditions, except the primary pony motors were turned off
- Tests also demonstrated effectiveness of Gas Expansion Modules (GEM) as passive reactivity reduction devices to overcome large Doppler feedback and stored heat of oxide fueled core during unprotected loss of flow events



TREAT METALLIC FUEL TESTS

- Transient overpower tests provided data for cladding failure margin, failure modes, location, timing, and insight into accident progression
- Seven tests with three metallic fuel designs
  - Tests M1-M4 tested U-5Fs fuel in 316-SS cladding
  - Tests M5-M7 tested U-Zr and U-Pu-Zr fuels in D9 and HT9 clad
- Tests were designed to be sufficiently severe to cause fuel damage
  - Nominal conditions were for 40 kW/m axial peak, 360^{o}C inlet temperature, and 150^{o}C coolant temperature rise in flowing sodium loop
  - Overpower tests with 8 s period leading to peak power of ~ 4x nominal
- Available experimental information include measurements for flow tube temperatures, cladding failure time and location, fuel axial expansion, fuel-melt fractions and other post-test examinations
- Measurements made with the fast neutron hodoscope demonstrated that:
  - Metallic fuel axially expand before the fuel melting and cladding breach
  - Molten fuel extrudes into pin plenum
  - When cladding fails, molten fuel-clad eutectic mix flows upward and exits the core

OUT-OF-PILE TRANSIENT TESTS

Tests conducted in two computer-controlled radiant furnaces

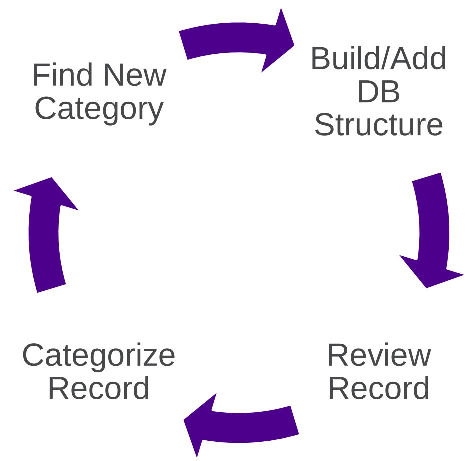
- Fuel Behavior Test Apparatus (FBTA) was capable of heating short (about 1 cm long) segments of irradiated fuel pins
  - >50 fuel-cladding compatibility tests for irradiated pin segments with U-10Zr or U-Pu-Zr fuel in 316SS, D9, and HT9 cladding
  - Segments cut at various axial locations ($0.20<x/L<0.93$) from fuel pins with 3 to 17 at.% peak burnup
  - Tests with 670-850ºC temperature range and 5 minutes to 4 hours duration yielding critical information regarding fuel melting and FCCI
- Whole Pin Furnace (WPF) was capable of accommodating intact whole fuel pins
  - Tests were considered representative of LOF accidents at decay heat levels
  - Six metal fuel tests were performed with U-Zr and U-Pu-Zr pins, all in HT9 cladding in a burnup range of 2.2 to 11.4 at.%.
  - Peak test temperatures varied from 650 to 820°C and test duration ranged from few minutes to 36 hours
  - Tests provided data for comparison with results of fuel behavior models that described modes, mechanisms, and thresholds of cladding failure

DATABASE DEVELOPMENT PROCESS

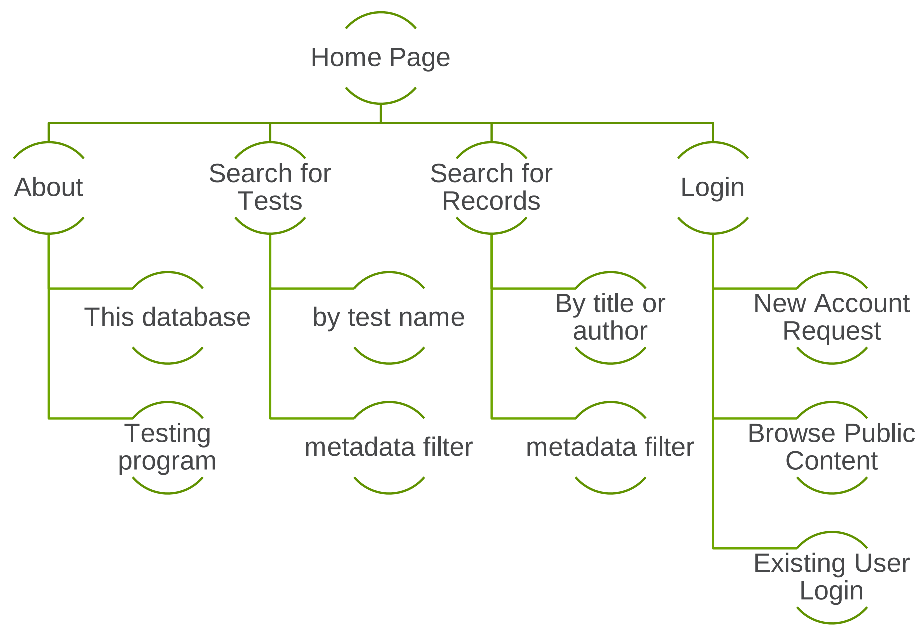
Collect & Preserve Records (Locate, Scan & Save)

Put Records into a Relational Database (Categorize & Index)

Make Accessible via a Web Interface (Distribute)



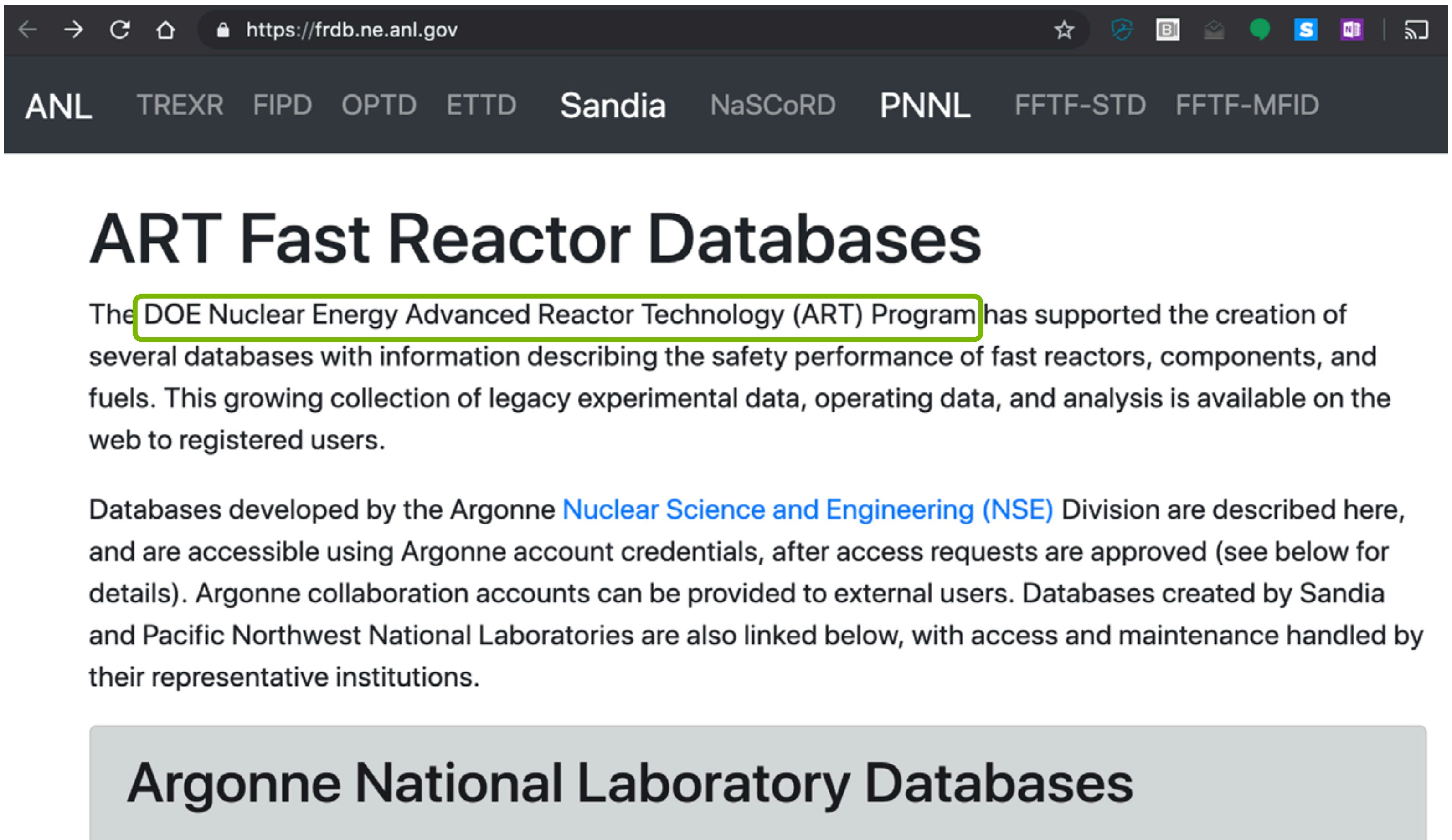
Web interface:

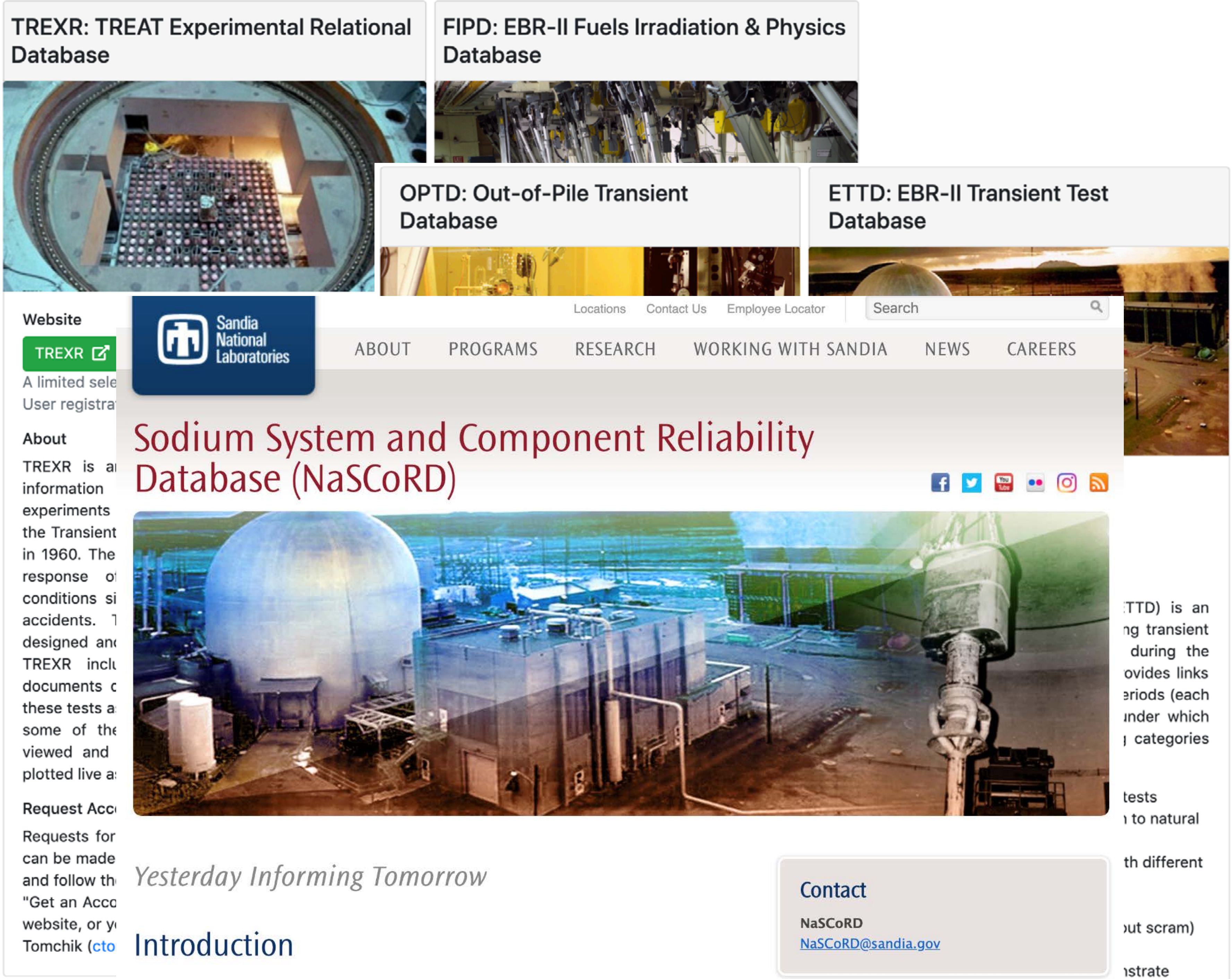


Argonne NATIONAL LABORATORY

ACCESS TO DATABASES

Available at https://frdb.ne.anl.gov/




TREXR: TREAT Experimental Relational Database
FIPD: EBR-II Fuels Irradiation & Physics Database
OPTD: Out-of-Pile Transient Database
ETTD: EBR-II Transient Test Database
Website
TREXR
About
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Sandia National Laboratories
Locations
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PROGRAMS
RESEARCH
WORKING WITH SANDIA
NEWS
CAREERS
Sodium System and Component Reliability Database (NaSCoRD)
Yesterday Informing Tomorrow
Introduction
Contact
NaSCoRD
NaSCoRD@sandia.gov

CONTENT AND USER CLASSIFICATION

- Content classification:
  - Open / Unlimited DOE Laboratory Reports
  - Applied Technology (AT) Reports
  - ECI
  - Other National Laboratory Reports
  - Informal Documents
  - Copyrighted Publications
  - Proprietary Documents
- User classification:
  - Argonne employees
  - Employees from other DOE labs
  - Users associated with US industry or universities
  - Distinction is also made based on the citizenship
- When a report cannot be made readily accessible (e.g., a journal article), bibliographic information is provided so that it can be requested from the publisher

USER ACCOUNTS

- All users given an individual user account
  – Assigned a user group by manual review
- User accounts/passwords stored in ANL Active Directory
  – Centrally managed by ANL IT
  – Conforms to ANL/DOE rules for password strength/security
  – ANL users can use their existing credentials
- External users are given Collaborator Accounts
  – Centrally managed by ANL IT
  – Exist specifically to give external users username/password credentials to access ANL computational resources
- Multifactor Authentication (MFA)
  – ANL Cyber rules require multifactor authentication (MFA) when accessing “sensitive” content (includes OUO and ECI)
  – ANL Cyber approved Duo MFA service
    - Users required to approve password logins using smartphone application

SEQUENTIAL ROLLOUT

- Rollout of external access to ANL databases is performed in stages
- Allowed for sequentially testing the application, server, settings, and firewall settings, in increasingly “open” network environments
- Also allowed for targeted testing of the application’s design and usability by a subset of testers who were able to provide feedback to the developers
- The rollout typically proceed in the following stages:
  - Stage 1: Argonne Nuclear Science & Engineering Division
  - Stage 2: DOE lab networks (e.g., INL, ORNL, Sandia)
  - Stage 3: Specific US company or university end users
  - Stage 4: Open Internet
- TREXR, ETTD, FIPD and NaSCoRD are available for external access
  - User accounts can be requested
- OPTD and FFTF databases are currently under development
  - FFTF database rollout is to be accelerated via GAIN funds

MODELING AND VALIDATION PROJECT PIPELINE

Database Development	Validation	Leveraged Industry and Intl. Collaborations
EBR-II Safety Testing Database	SHRT-17, 43R 45, 45R, BOP-301 and 302	IAEA-CRP, NRC, CNWG, KAERI SPP, CIAE CRADA
FFTF Passive Safety Test Database	LOFWOS #4-13	IAEA-CRP, NRC, TP, CNWG
TREAT Test Database	M5, M6, M7	KAERI SPP, CNWG,NRC TP, GEH

Access to Phenix, Monju, CEFR and CABRI tests and T/H tests (WAJECO, PLAJEST, PLANDTL, SUPERCAVNA, TALL3D)

BENCHMARK SPECIFICATIONS

ANL-ARC-226 Rev.1

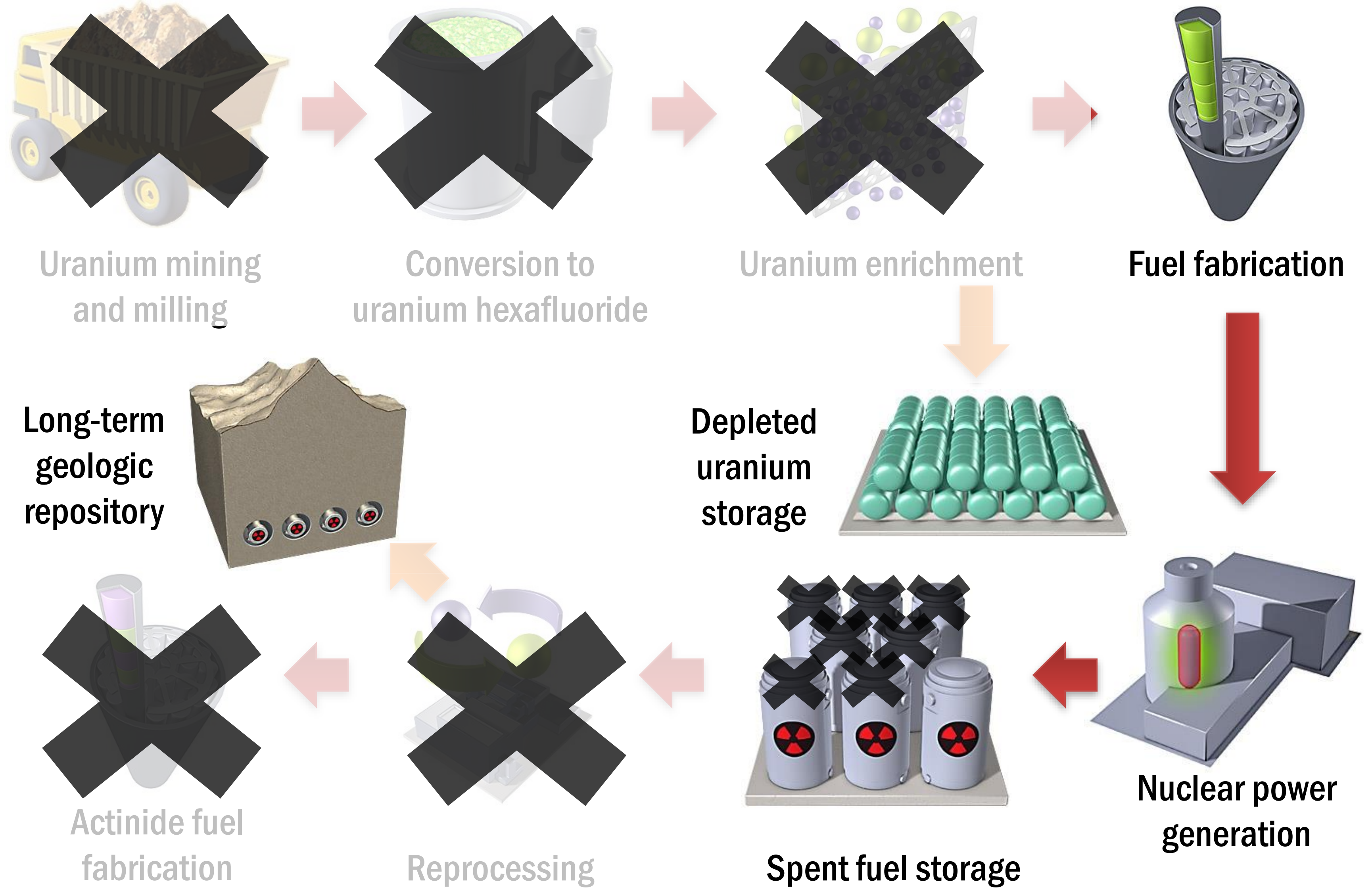
Benchmark Specifications and Data Requirements for EBR-II Shutdown Heat Removal Tests SHRT-17 and SHRT-45R

Nuclear Engineering Division

TerraPower Fuel Cycle Strategy

Pavel Hejzlar and Phil Schloss

TWR Once-Through Fuel Cycle versus LWR Fuel Cycle

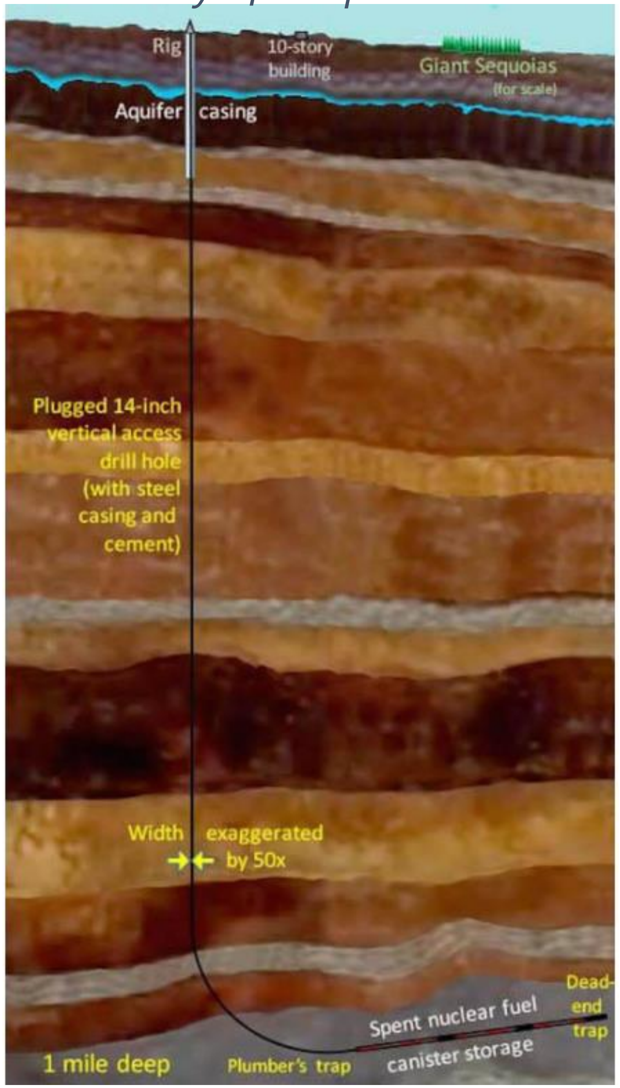


- First core uses enriched fuel, reloads DU or NU
- Subsequent plant can use fuel from first plant as is
- Deep breed and burn in situ with high burnup
- Good proliferation resistance – no reprocessing, reduced requirements for enrichment
- High Pu240/P239 content – unattractive Pu vector
- Up to 30x higher U utilization than LWR
- Simpler cycle, lowers overall cost of overall nuclear energy process

TerraPower

Deep Borehole Permanent Waste Disposal for TWR fuel

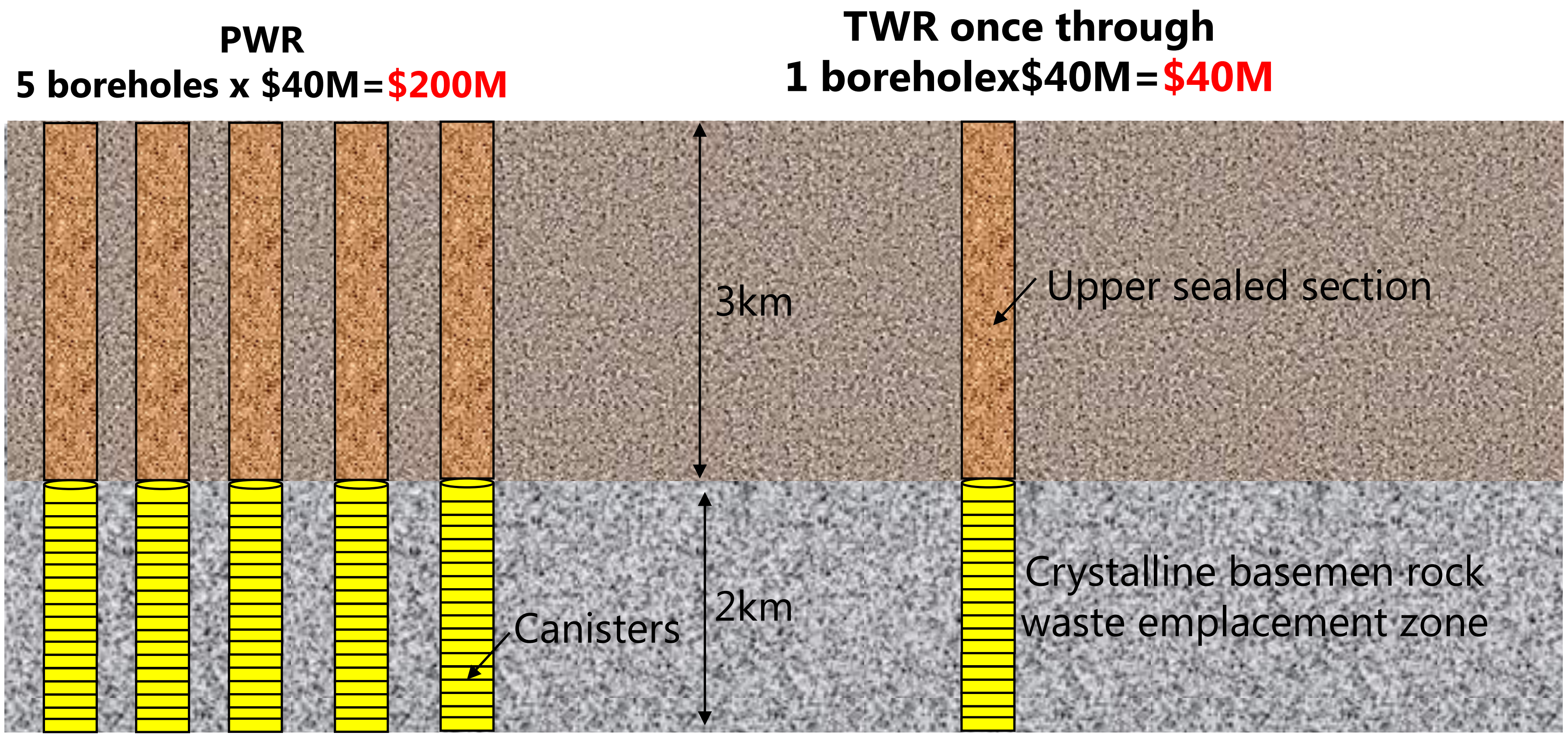
Deep Isolation Techology
Courtesy of Deep Isolation



- **Attractive and more economic option for TWR spent fuel**
  - LWR spent fuel has about <u>95% energy unused</u>
  - TWRs could use ~20% energy, hence irretrievability argument is easier to make for TWR fuel
  - Moreover, irretrievability becomes benefit once energy is mostly used since fuel cannot be used as a Pu mine in future
  - Boreholes are not heat limited and are simpler to analyze and understand
  - SNL study (Arnold et al, 2011) concluded that deep boreholes are operationally feasible with low total costs of $150/kgHM.
    - Cost ~$40M/borehole including canister loading and hole sealing
    - Compares to US fees of $400/kgHM based on 1mill/kWhr
  - Recently startup company Deep Isolation announced partnership with Bechtel to develop its patented even more economic deep borehole technology
    - https://www.forbes.com/sites/jamesconca/2019/06/24/deep-borehole-nuclear-waste-disposal-just-got-a-whole-lot-more-likely/#524aa9e767c8

TWR spent fuel borehole repository has 5 times lower cost than LWR

- Borehole with 2km of storage height can hold 400 canisters, each 5m tall
- Number of boreholes to emplace spent fuel produced over 60 year life



Versatile Test Reactor Update

Jordi Roglans

Argonne National Laboratory

VTR Deputy Program Manager

FRWG

July 9-10, 2019

All information shown is preliminary; DOE has not yet made a decision regarding technology or location choice. This work is to provide information for the DOE decision process.

Argonne NATIONAL LABORATORY | INL Idaho National Laboratory | Los Alamos NATIONAL LABORATORY | OAK RIDGE National Laboratory | Pacific Northwest NATIONAL LABORATORY | SRNL SAVANNAH RIVER NATIONAL LABORATORY

BACKGROUND

Nuclear Energy Innovation Capabilities Act of 2017

S.97 - Nuclear Energy Innovation Capabilities Act of 2017 (approved 9/18) requires:

- Determine the mission need for a versatile reactor-based fast neutron source.
  - *With a high neutron flux, irradiation flexibility and volume for many concurrent users, multiple loops, considering lifetime operating costs and lifecycle costs,*
- DOE to construct a Versatile Reactor-Based Fast Neutron Source;
- To the maximum extent practicable, approve start operations no later than December 31, 2025.

Executing the S.97 direction requires:

- Selection of a high TRL proven technology with significant operating experience, a **sodium fast reactor** (use of more mature technology, previously used/tested fuels….)
- **Leverage existing designs** to reduce design time,
- **Immediate initiation** of project activities. (Extremely challenging schedule for a nuclear build)

VERSATILE TEST REACTOR

- Testing to advance reactor fuels and materials for multiple technologies
  - Bridge capability gaps (fast neutrons, high dpa, large volumes)
  - Provide capability for accelerated testing of advanced fuels and materials
  - Irradiation capabilities for a range of coolants (sodium, lead, salt, gas…)

- Strategy established to minimize risk:
  - Use of existing, mature technology
  - Leverage existing reactor design and modify for test reactor use
    - GE Hitachi PRISM design selected as basis for adaptation to test reactor mission
  - Extensive team formed with Laboratories, Industry, Universities
  - Experiment development team with Laboratories, Universities, Industry
    - Involvement ensures VTR meets industry needs

- DOE safety and regulatory work initiated, Safety Design Strategy under DOE review, and a DOE and the NRC collaboration framework is under development

- With CD-0 approval in February, 2018, project is progressing through a conceptual design and assessment and selection of options

CD-0 Cost & Schedule – CD Dates

Follows DOE O 413.3B

CD-0 Cost and Schedule Range

- Cost Estimate: $3.0 to $6.0 Billion
- Completion Estimate: 2026 to 2030

Milestone	Fiscal Year
CD-0	FY 2019
CD-1	FY 2021 (1st Qtr)
CD-2/3	FY 2022
CD-4	FY 2026

Critical Decisions ("CDs")	CD-0 Approve Mission Need	CD-1 Approve Alternative Selection and Cost Range	CD-2 Approve Performance Baseline (PB)	CD-3 Approve Start of Construction	CD-4 Approve Start of Operations

Critical Decision-1 (CD-1)

CD-1, *Approve Alternative Selection and Cost Range*

- Conduct an analysis of alternatives
- Develop a conceptual design with schedule and cost range.

Strategy

- Focus on usable engineering products to support cost range generation
- Use engineering products to position program for expedited preliminary/final design completion
- Establish clear understanding of PMRC expectation for cost range approach and accuracy
- Continue to align closely with PM
- Progress to date demonstrates the path to an expedited design build is achievable.

Important elements for success

- Analysis of alternatives and NEPA must be closely focused on practical/achievable outcomes
- Continuity of funding for FY 2020/2021 is necessary for productivity and efficiency of program team
- Determination of the fuel source material must be made.

Preliminary requirements/assumptions

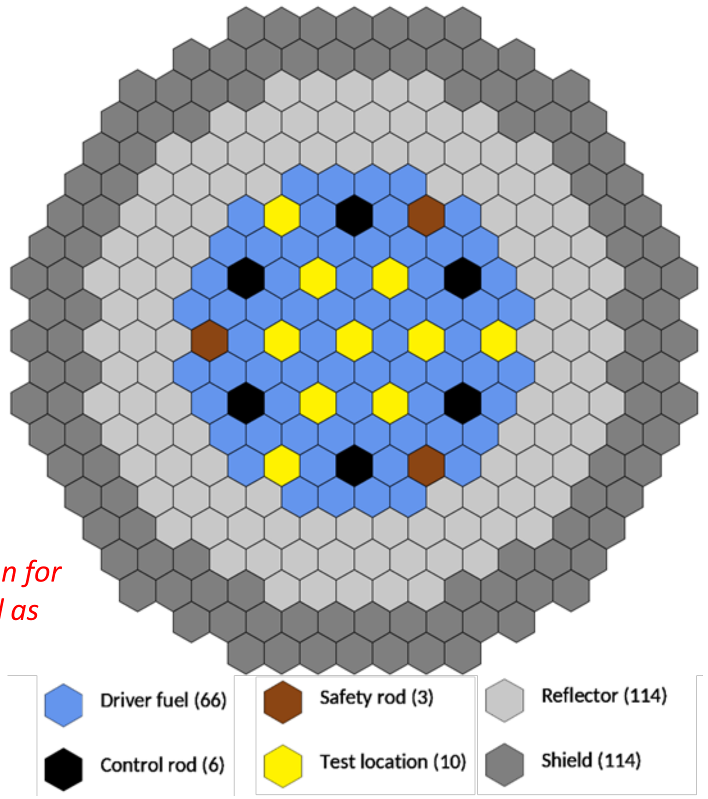
Parameter	Target
High neutron flux	≥ 4 x 10^{15} n/cm^2-s
High fluence	≥ 30 dpa/yr
High test volume in the core	≥ 7 L (multiple locations)
Representative testing height	0.6 ≤ L ≤ 1 m
Flexible test environment	Rabbit & Loops (Na, Pb, LBE, He, Salt)
Advance instrumentation & sensors	In-situ, real time data
Extensive capability	Ability to accommodate multiple experiments simultaneously
Experiment life cycle	Experiment support infrastructure

ASSUMPTIONS – pending AOA and NEPA:

- Mature Technology: Sodium-cooled pool type reactor, inherent and passive safety
- Metallic alloy fuel (HALEU, LEU+Pu, DU-Pu)
- Pool-type design: for versatility and experimental flexibility
- Novel testing capabilities

Core – Initial Tradeoff Studies and Current Status

- Comparison of fuel compositions and assessment of flux levels achievable with each of them, as well as the required core size and power level required to achieve these fluxes
- Some of the design parameters:
  - Preferred use of ternary metallic fuel – experimental database
  - Fuel design parameters supported by experimental database
    - Sodium-bonded, solid fuel slugs
    - HT9 or 316SS cladding, wire-wrap, and duct
    - Fuel length: 80 cm
  - Core Power: 300 MWth
  - No electricity production – heat rejection to atmosphere
  - 6 control rods and 3 safety rods fixed locations
  - Sodium inlet/outlet temperature: 350°C/500°C
  - Peak cladding temperature: ≤650°C
  - Nominal fuel bundle pressure drop: ≤0.5 Mpa
  - Coolant velocity: ≤12 m/s



Note: Specific test location for cartridge loops selected as design progresses

Nuclear Build Risks Addressed in Plan, Design, and Budget

- Design completion
  - **Complete design as required to support nuclear design and mitigate potential cascading risks**
  - Complete analysis and calculations supporting key design aspects
  - Utilize modern construction management tools: virtual design and construction/building information management
  - Identify long-term R&D to be conducted in parallel with acquisition and operations.
- Authorization documentation
  - Complete preliminary safety basis documents, supporting calculations, and analysis; receive review and concurrence by the regulator
  - Complete NEPA development strategy and associated preliminary NEPA documentation.
- Supply chain and construction planning
  - Verify supply chain with viable acquisition path for all key components
  - Perform required construction planning for cost estimating and early site work.
- Cost estimate and schedule
  - Use qualitative and quantitative risk-based cost estimate scope control processes
  - Address entire design, construction, and operational testing scope
  - Requires maturity in design, safety, and supply chain as noted above.
- Quality and independent review processes
  - Embrace NQA-1 quality approach: progressive quality increase from concept through mature product
  - Collect peer and independent reviews to ensure early external engagement and risk reduction.

Three Major Elements of Scope

- **Integration, Core, Fuel, Safety Analysis, Safety Basis, PRA, Support Facilities:**
  - DOE Laboratories
- **Reactor Concept Design, Cost Estimate:**
  - Industry
- **Experiment Concept Development:**
  - DOE Laboratories
  - Industry
  - Universities

VTR Team

Universities (includes co-PIs)

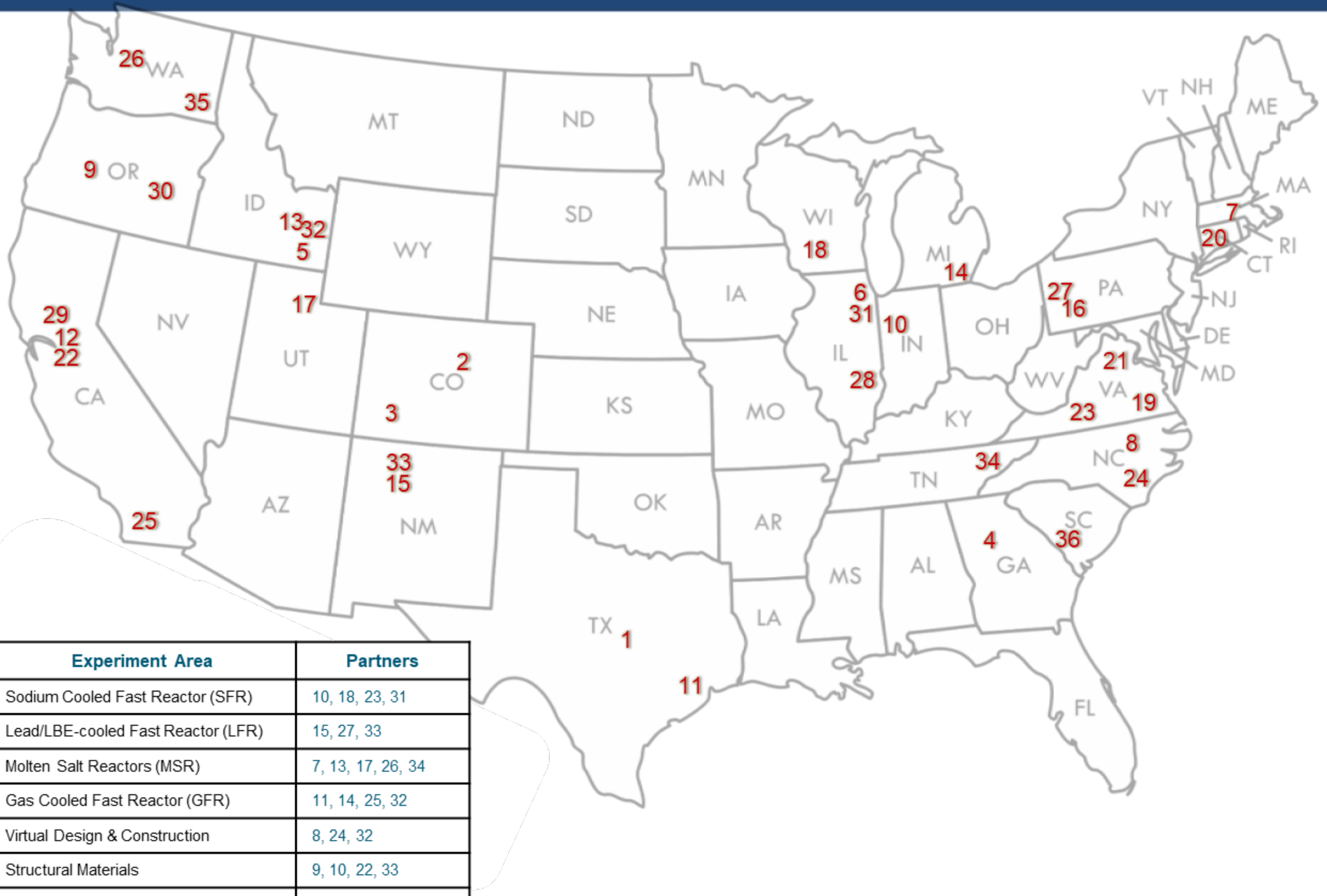
1 Abilene Christian University
2 Colorado School Of Mines
3 Fort Lewis College
4 Georgia Institute Of Technology
5 Idaho State University
6 Illinois Institute Of Technology
7 Massachusetts Institute Of Technology
8 North Carolina State University
9 Oregon State University
10 Purdue University
11 Texas A&M University
12 University of California, Berkeley
13 University of Idaho, Idaho Falls
14 University of Michigan
15 University of New Mexico
16 University of Pittsburgh
17 University of Utah
18 University of Wisconsin - Madison
19 Virginia Commonwealth University
20 Yale University

Industry

21 Bechtel Corporation
22 EPRI Electric Power Research Institute
23 framatome
24 General Electric
25 General Atomics
26 TerraPower
27 Westinghouse
28 The HDF Group
29 The Cameron Group, Inc.
30 Columbia Basin Consulting Group

Laboratories

31 Argonne National Laboratory
32 Idaho National Laboratory
33 Los Alamos National Laboratory
34 Oak Ridge National Laboratory
35 Pacific Northwest National Laboratory
36 Savannah River National Laboratory



Experiment Area	Partners
Sodium Cooled Fast Reactor (SFR)	10, 18, 23, 31
Lead/LBE-cooled Fast Reactor (LFR)	15, 27, 33
Molten Salt Reactors (MSR)	7, 13, 17, 26, 34
Gas Cooled Fast Reactor (GFR)	11, 14, 25, 32
Virtual Design & Construction	8, 24, 32
Structural Materials	9, 10, 22, 33
Data Analytics Combined with M&S	1, 2, 4, 7, 19, 28, 32
Rabbit Systems	11, 35
Strategic Initiatives	9, 16, 32

Preliminary Information

Argonne National Laboratory · Idaho National Laboratory · Los Alamos National Laboratory · Oak Ridge National Laboratory · Pacific Northwest National Laboratory · SRNL Savannah River National Laboratory

Reactor Design Support Contract – GEH/BNI

DOE

BEA

GEH

Bechtel

GE-HITACHI & BECHTEL

- **Personnel**
  - Gerald Goldner, Project Manager, GEH
  - Eric Loewen, Chief Engineer, GEH
  - Steve Routh, Project Manager, Bechtel.
- **Deliverables after CD-0**
  - ✓ Adapted PRISM concept for VTR mission; delete/add/modify SSCs
  - ✓ Advanced conceptual/preliminary design
  - ✓ High-confidence cost estimate
  - ✓ High-confidence schedule estimate.

Safety and Design Approach

- General approach to safety
  - Utilize inherent and passive safety possibilities of SFRs
  - Will be licensed under the DOE framework
    - Utilize a risk-informed authorization approach

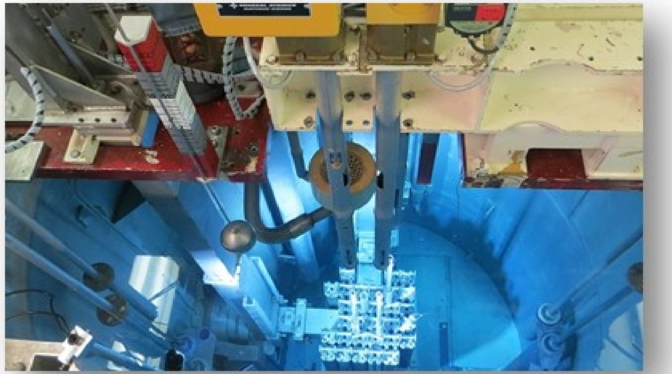
- Safety analysis
  - To support preparation of the Safety Analysis Report
  - Initially focuses on postulated protected transient scenarios
  - Used to inform some of the design decisions

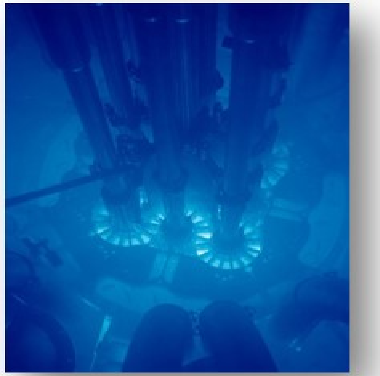
- Probabilistic Risk Assessment
  - Being developed as part of the VTR project
  - Based on DOE and ASME standards

- Design evolution within project
  - Overall plant design based on PRISM
    - Adaptation to test mission
    - Inclusion of core design provided by laboratory team to meet experimental design requirements
  - Multiple tradeoff studies to select preferred design options
  - Iterative process between project participants
  - Digital Engineering – Requirements management

Regulatory Approval Pathway

- Regulatory/authorization strategy
  - Leverage DOE experience in authorizing operations of a wide variety of reactor and non-reactor facilities (four operating reactors at INL site)
  - Accelerate schedule through early development of Safety Design Strategy
    - Establishes regulatory certainty and common understanding of expectations.
- NRC and industry engagement
  - Interacting with industry/NRC Licensing Modernization Project (LMP) (NEI 18-04)
  - Making VTR process consistent with LMP process, tailored to meet DOE requirements
  - Likely first application of the process for a large reactor.
- NRC/DOE MOU
  - Allows NRC to inform its licensing regulatory development by observing a DOE process
  - Provides opportunity for outside feedback to DOE approval authority.

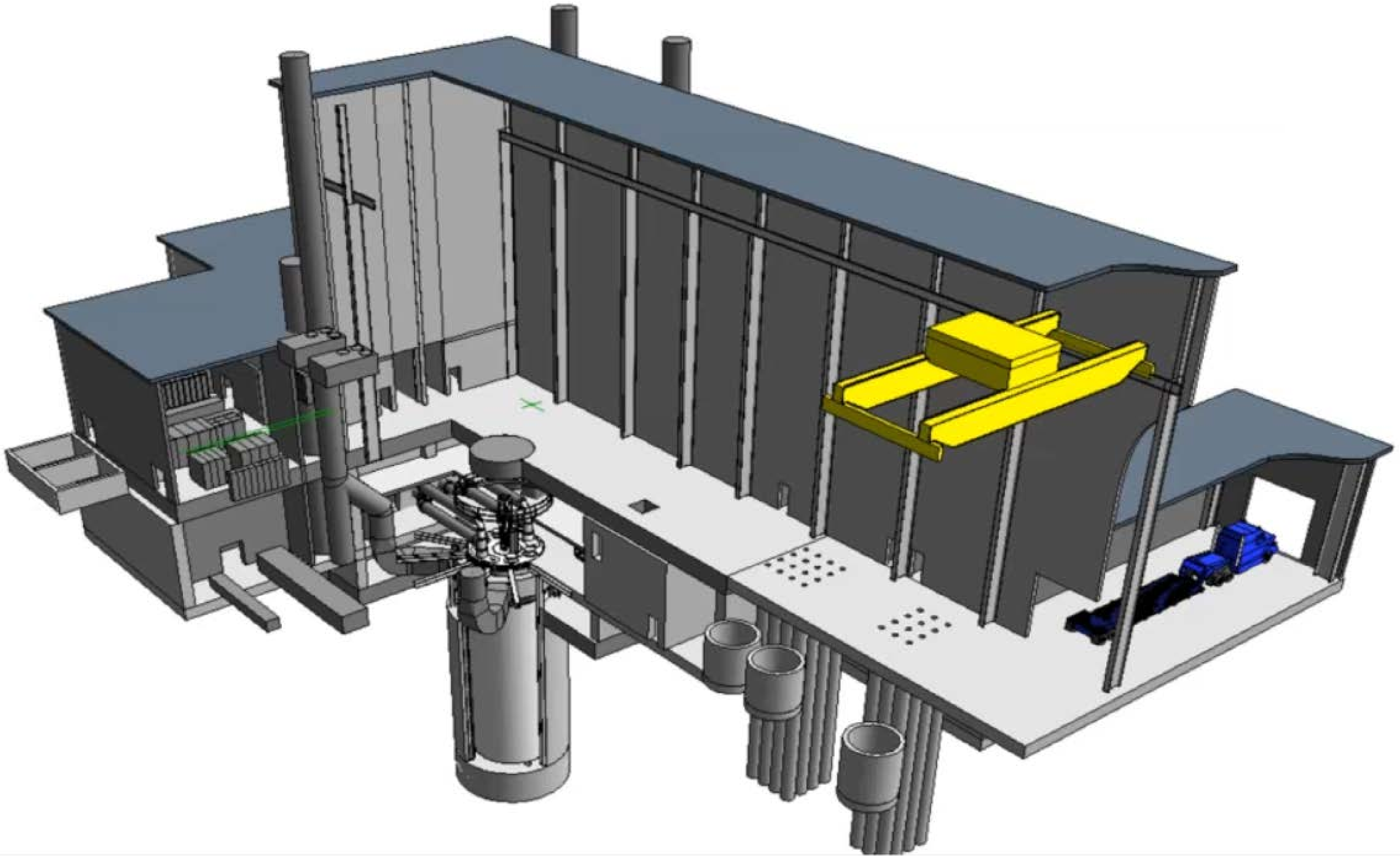




Example Design Decisions for Tailoring PRISM

- Reduce power from ~500 MWt to ~330 MWt by reducing the primary flow while keeping dP the same, optimize EM pumps for best efficiency at reduced flow
- Keep the primary vessel, guard vessel, and major elements of head, rotating plug, and upper internal structure the same
- Change subassembly flow control to accommodate test assemblies and core area storage
- Replace Steam Generators with Sodium to Air Heat Exchangers
- Modified RVACS design: stacks are steel piping vs concrete stacks
- Addition of an Experiment Hall in place of Refueling Building, 125 ton crane, in ground exp. storage
- Steel building above ground
- Experiment rooms adjacent to the head access area, and at grade elevation
- Spent fuel casks will be for a single subassembly, cask transport is by building crane and by truck
- Cleanup and preheat of fuel and experiments before insertion is by hot Argon, proven effective at FFTF
- Reactor protection system will be analog, Diverse protection system will be PLC or FPGA, and control computer will be a modern industrial control such as Triconix

VTR – General Arrangement

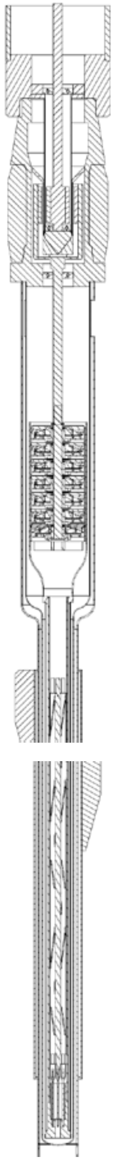


Preliminary Information

Experiment Capability Considerations

VERSATILITY

- VTR offers extensive testing capabilities:
  - At least four instrumented test locations
  - A rabbit system for quick irradiations insertion and retrieval at power
  - Cartridge loop for alternate/independent coolants
  - Any driver can be replaced with test assembly
  - Additional experiments in the reflector region do not impact the core performance
- As well as very attractive set of irradiation conditions:
  - Peak fast flux in central test location: ~4.2×10^{15} n/cm^2-s
  - Peak total flux in central test location: ~6.0×10^{15} n/cm^2-s
  - Possible testing length up to 250 cm



Overview of Progress

- A multi-laboratory, university, and subcontractor team has been established.
- Strategy established to:
  - Leverage existing fast reactor design and modify for test reactor use; *GEH/Bechtel team was selected to modify the PRISM Mod A design,*
  - Utilize industry to ensure experiment capability answers the industry need, and utilize universities to assist in experiment development; *Participants were selected and are under contract*
- Users' information gathered on the desired experimental capabilities for the reactor
- VTR task force under NEI advanced reactor working group has been established
- DOE safety and regulatory work initiated, the Safety Design Strategy is under DOE final review, and a DOE and the NRC collaboration framework is under development
- DOE approved the Mission Need (CD-0) on February 28th, 2019
- CD-1 conceptual design, conceptual safety design, other documentation well under way

Thank you!