Overview of Metal Research in the LWRS Program Materials Research Pathway

LWRS LIGHT WATER REACTOR SUSTAINABILITY

Xiang (Frank) Chen, Thomas M. Rosseel, Keith J. Leonard, & MRP Metals Research Team

U.S. NRC International Workshop on Age-Related Degradation of Reactor Vessels and Internals
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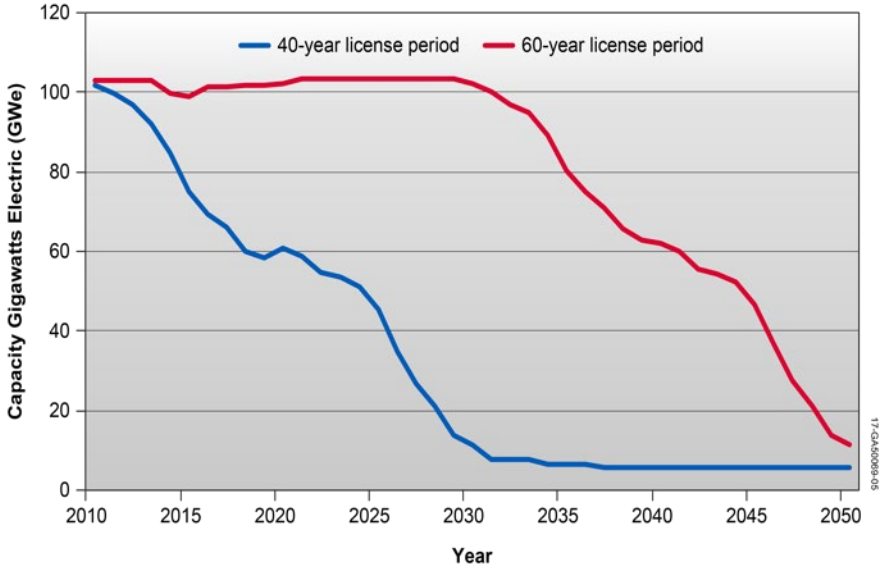
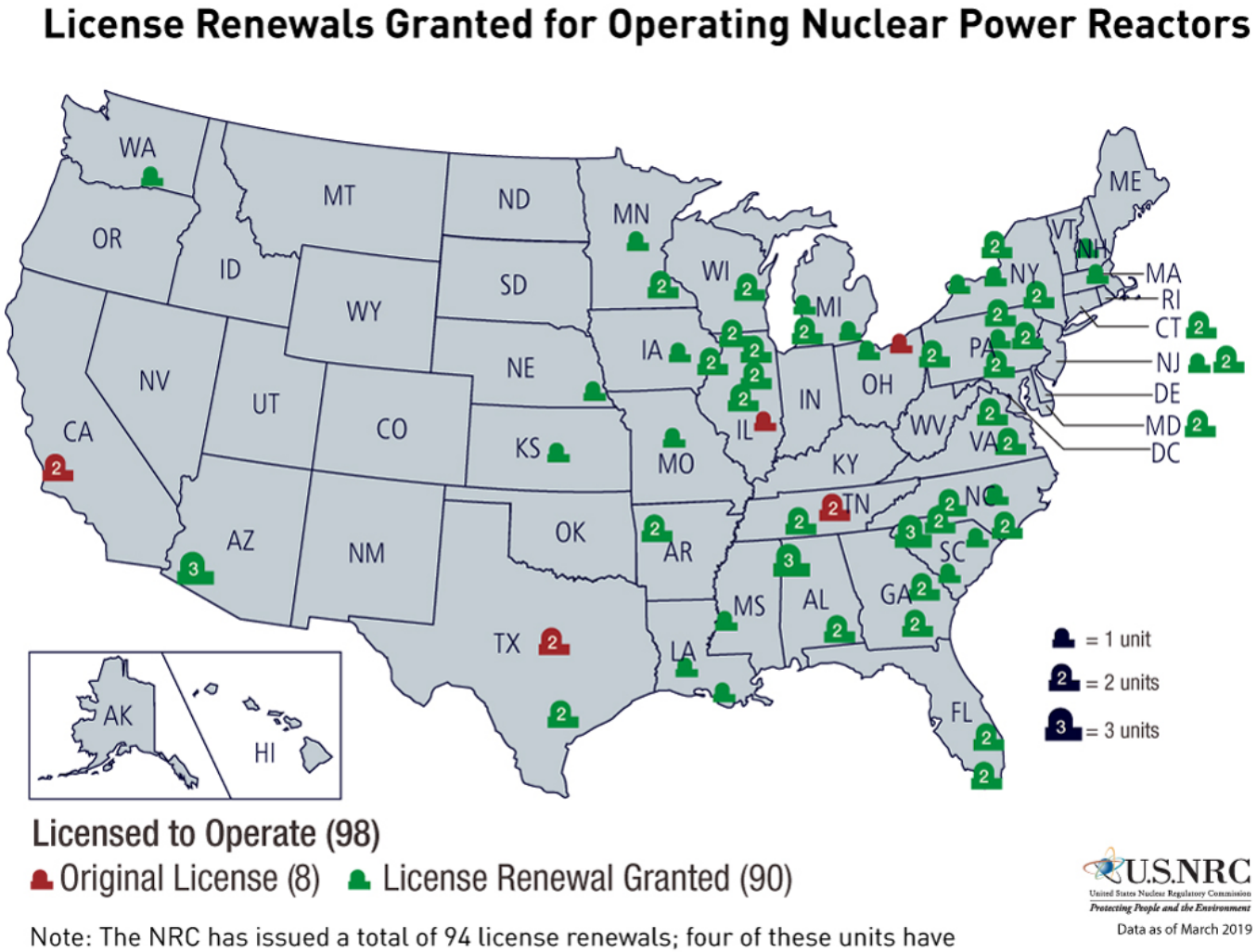
Rockville, MD

Extended operations of the existing commercial power generation reactor fleet is in the U.S. national interest

- Our LWRs are a national asset
  - ~100 GWe of low-carbon generation
  - Low-cost, reliable generation
  - Energy diversity
  - Nearly $1T replacement cost

- Even with first 20-year extension, nearly all plants will reach the end of their 60-year license between 2030 and 2055





Note: The NRC has issued a total of 94 license renewals; four of these units have permanently shut down. Data are as of March 2019.

LWRS Program Goals and Objectives

The mission of the Light Water Reactor Sustainability Program (LWRS) is to develop the scientific basis, and science-based methodologies and tools, for the safe economical long-term operation of the nation's high-performing fleet of commercial nuclear energy facilities

- **Objectives**
  - Provide science and technology-based solutions to industry to overcome the current labor-intensive business model and associated practices
  - Manage the aging of systems, structures, and components so nuclear power plants can continue to operate safely and cost effectively

- **Pathways**
  - Materials Research
  - Plant Moderation
  - Risk-informed Systems Analysis

Nine Mile Point (courtesy Exelon)

The LWRS program is the primary U.S. DOE program for light water reactor research, development and demonstration

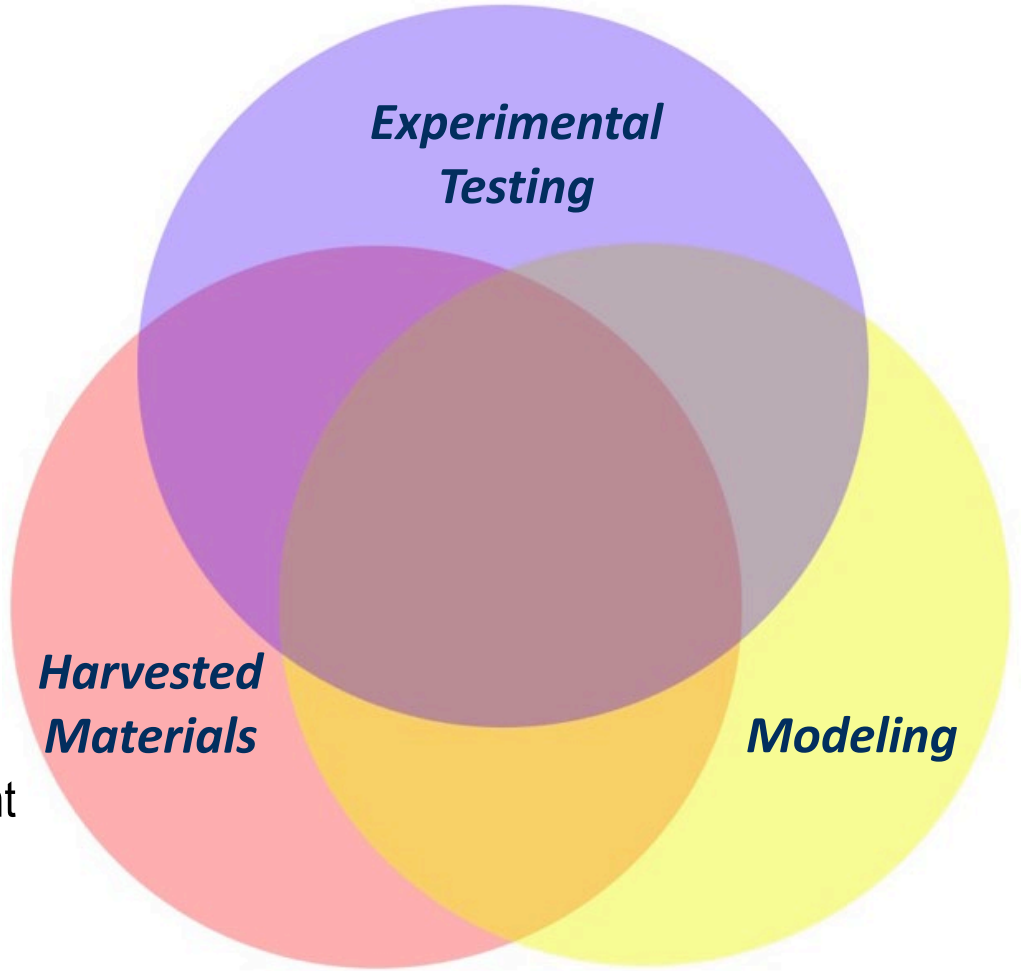
Materials Research Pathway (MRP)

- **Objectives**
  - Develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants

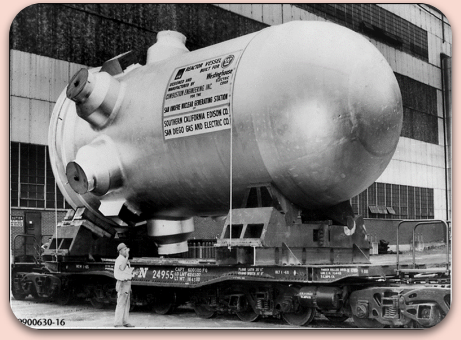
- **Approach: guided by the "5M"**
  - Measurements of degradation
  - Mechanisms of degradation
  - Modeling and simulation
  - Monitoring
  - Mitigation strategies

- **Research benefits**
  - Understanding which components are susceptible to certain forms of degradation, and their predictive behavior, will permit more focused component inspections, component replacements, and more detailed regulatory guideline
  - The R&D products will be used by utilities, industry groups, and regulators

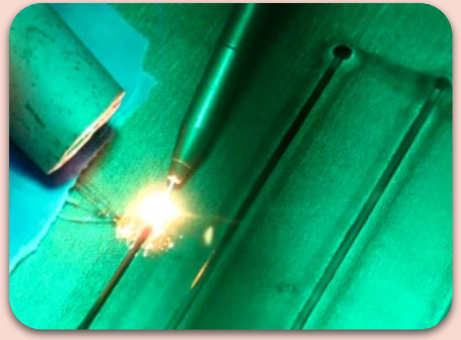
- **Partners**
  - EPRI, Westinghouse, PWROG, CRIEPI, Rolls Royce, Exelon, U.S. NRC



Materials Research Pathway Metal-Related Project Portfolio

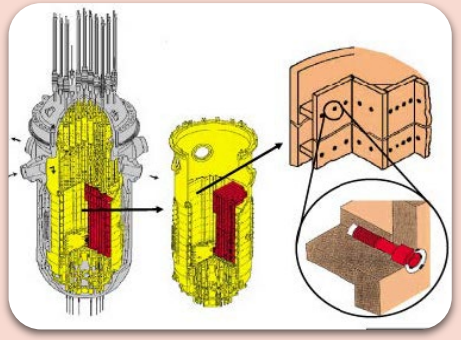


Reactor Pressure Vessel (RPV)



Mitigation Technologies

Harvesting and Integrated Research



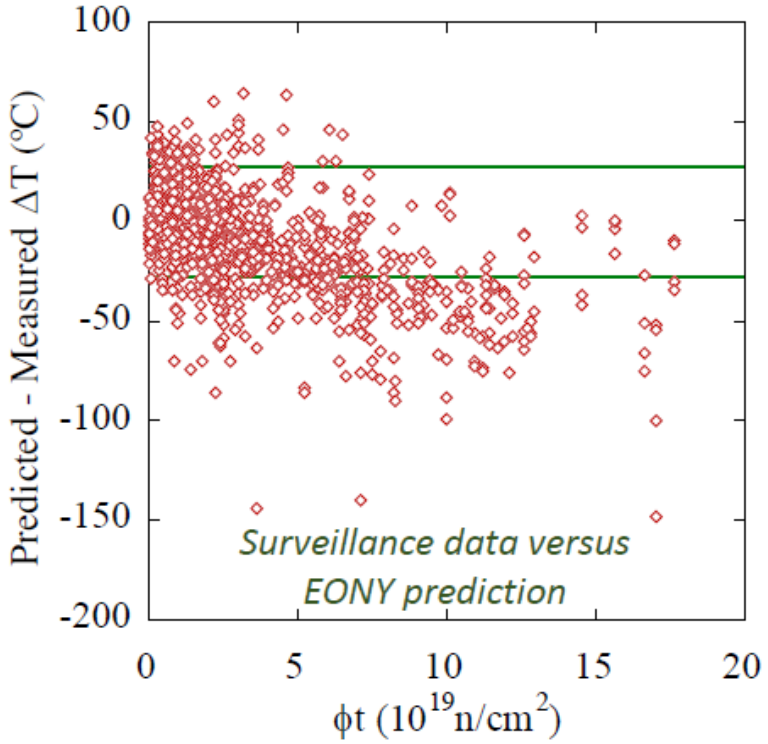
Core Internals and Piping

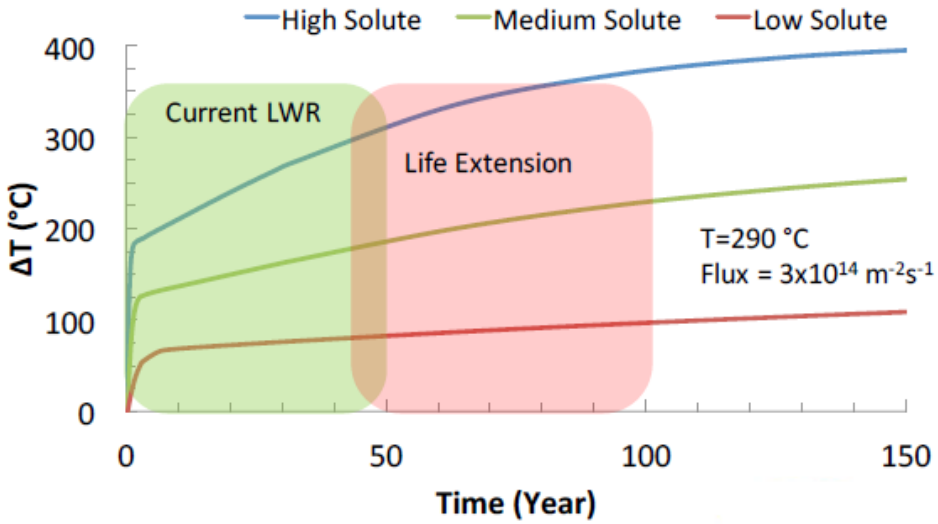
Other Activities

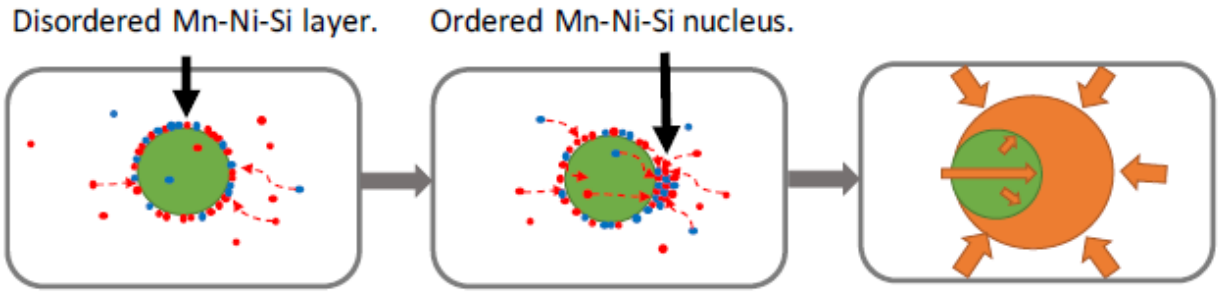
Reactor Pressure Vessel (RPV)

- **ATR-2: UCSB, Odette**
  - Expand database (microstructure / mechanical properties) on effects of fluence, flux, temperature over a large range of alloy compositions

- **Modeling of Cu and MnNiSi precipitates: U. Wisconsin, Morgan**
  - Predict the Cu and MnNiSi formation for a wide range of compositions, neutron fluxes, and temperatures
  - Provide information on the formation mechanism of Cu-core-MnNiSi-appendage microstructure

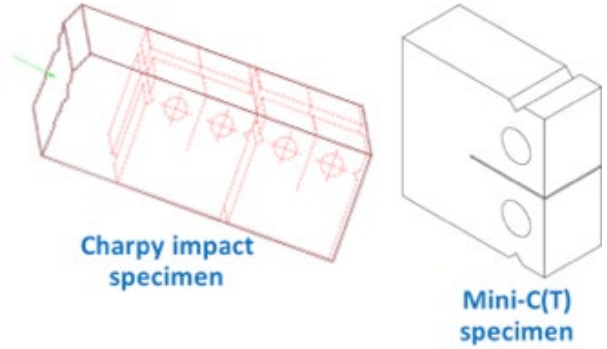


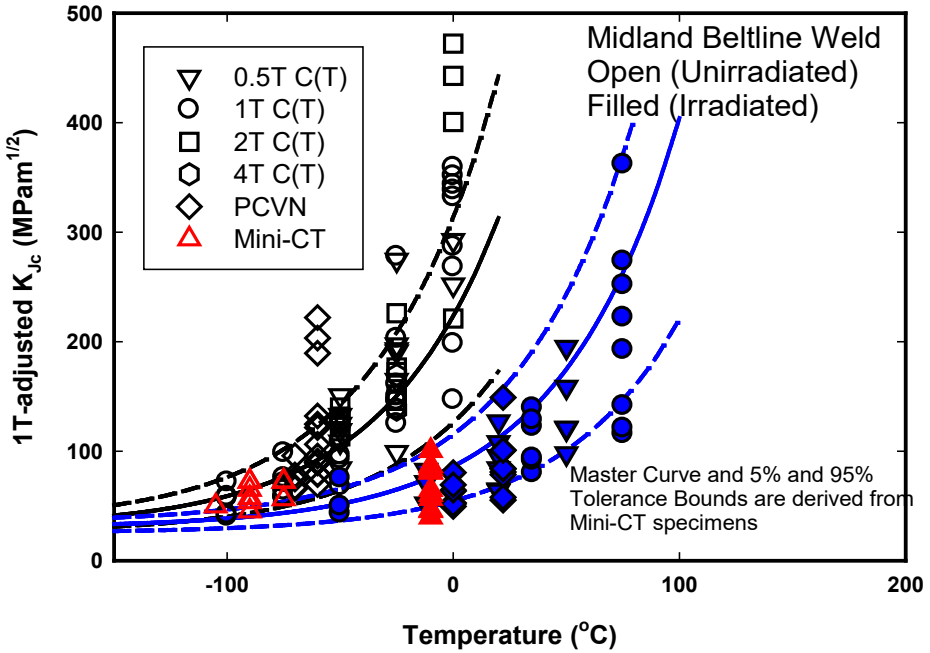




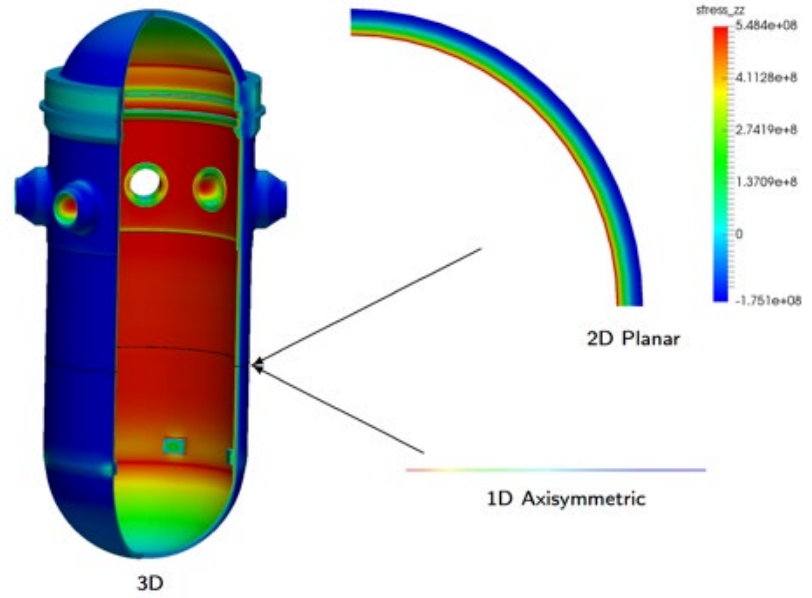
Reactor Pressure Vessel (RPV)

- **Mini-compact tension specimen: ORNL, Sokolov**
  - Develop the testing technology for determining Master Curve fracture toughness for RPV steels





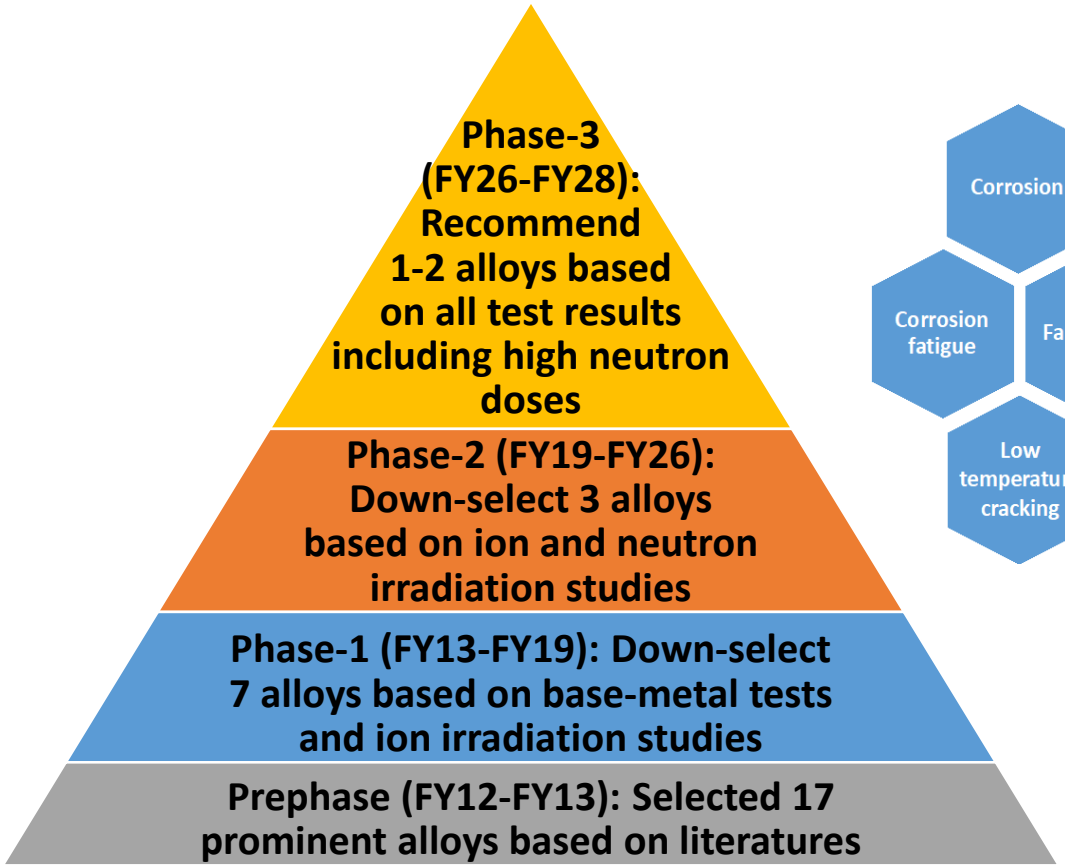
- **Grizzly multi-physics simulation: INL, Spencer**
  - Develop a simulation tool to predict the progression of aging mechanisms and their effects on integrity of multiple critical nuclear power plant components (RPVs, concrete structures, etc.)

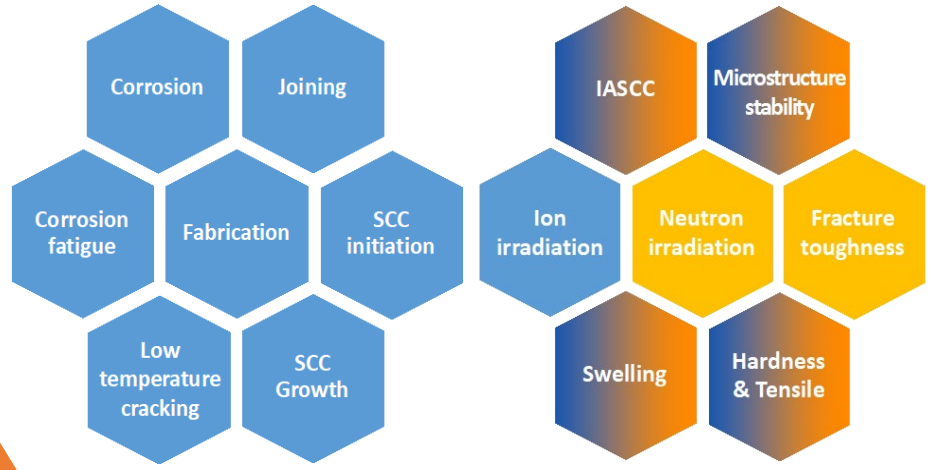


Results of 1D axisymmetric, 2D planar, and 3D Grizzly models of the global response of an RPV at a point in time during a PTS event

Mitigation Technologies

- **Advanced replacement alloys: ORNL/Tan, U. Michigan/Was, EPRI ARRM program, KAPL, GE, PNNL, INL**
  - Down-select and develop advanced radiation resistant materials (ARRM) for core internal support components and fasteners as replacement materials in current LWRs and new materials in future reactors



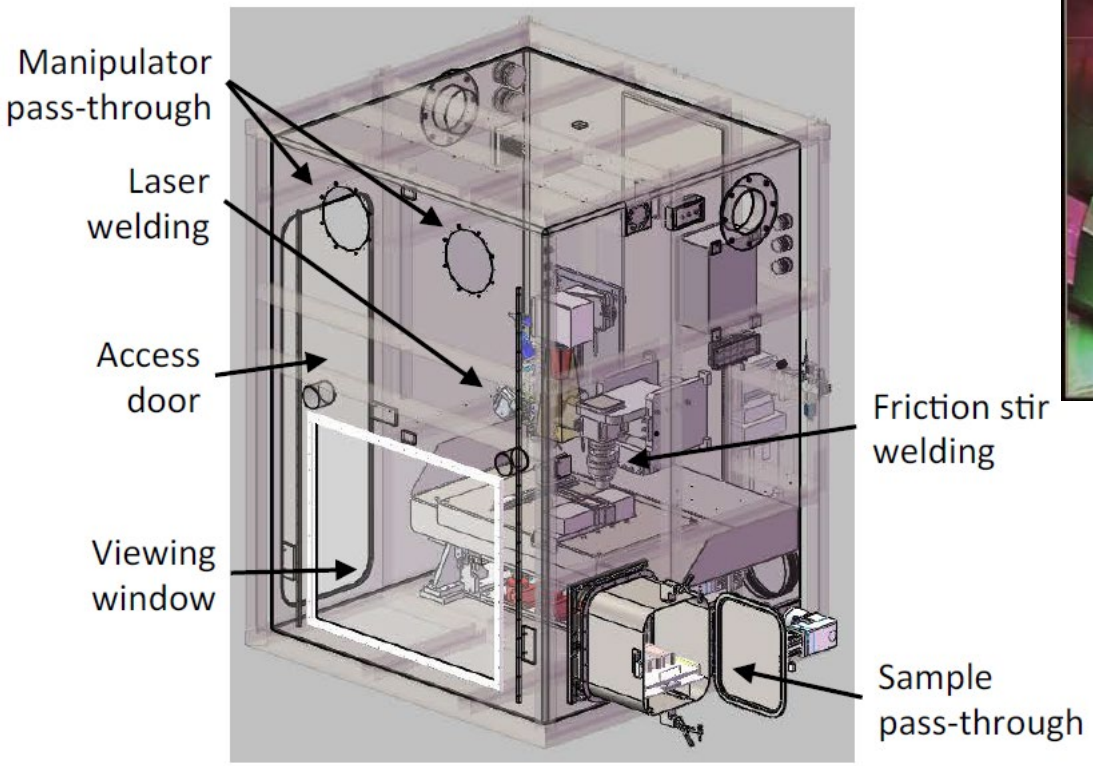


Non-rad & Rad tests

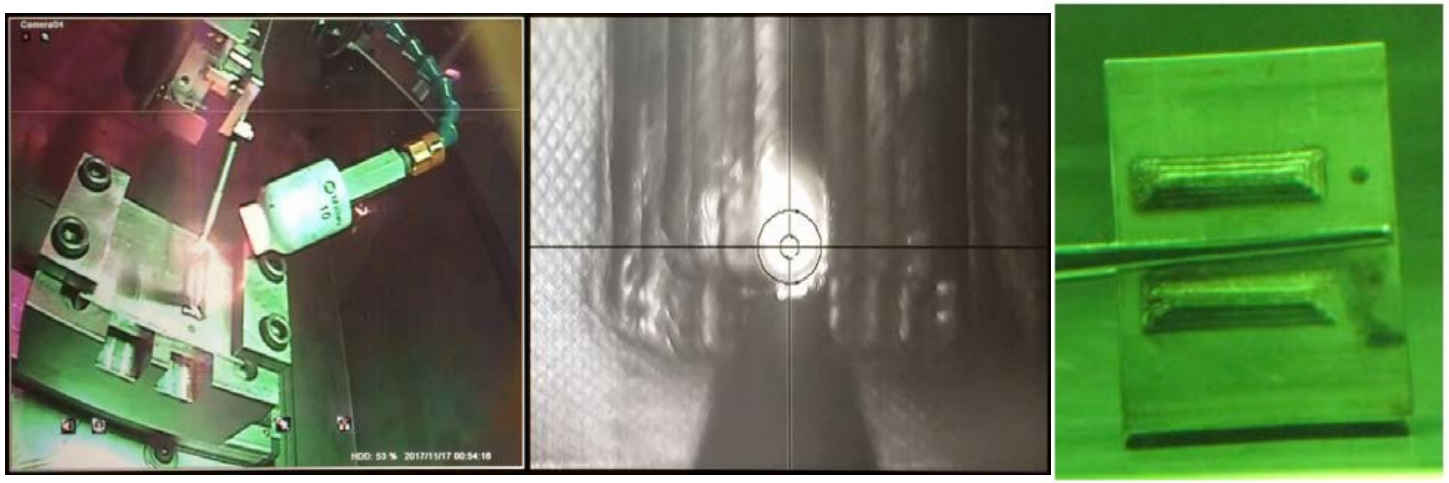
Candidate Alloys	
Low Strength	
316L (standard)	
310	
800	*IASCC*
Grade 92 (NF616)	
HT9	*Fracture toughness*
C22	*Radiation-induced hardening*
625	*IASCC*
690	
Zr-2.5Nb	*Accident tolerance concerns*
Ti alloys	*Fracture toughness*
High Strength	
X-750 (standard)	
439	*Fracture toughness / processing*
625 plus	*IASCC*
625 DA	*IASCC and stability*
725	
718 (heat A)	
14WYT	*Fracture toughness / fabr.*
Hi-Cr, Al ODS	*Examined under FCRD*

Mitigation Technologies

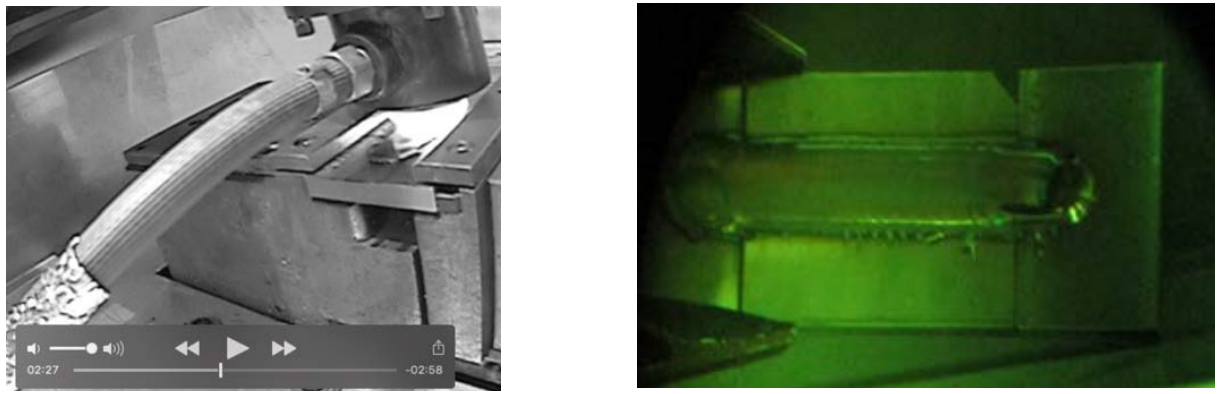
- **Welding repair: ORNL (Feng et al.) & EPRI (Frederick et al.)**
  - Develop advanced welding technologies to weld highly irradiated materials while avoiding helium induced cracking



Irradiated materials welding facility at ORNL



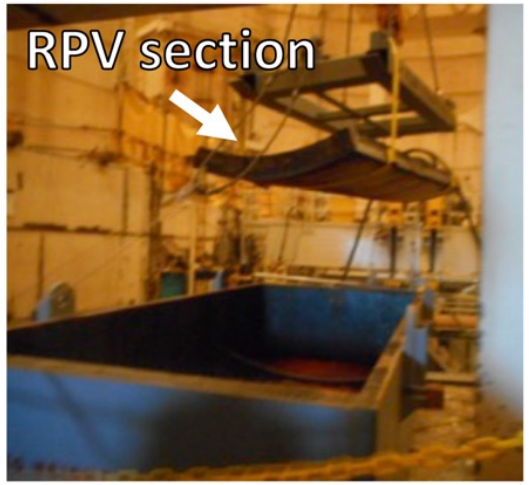
Laser welding of irradiated 304L

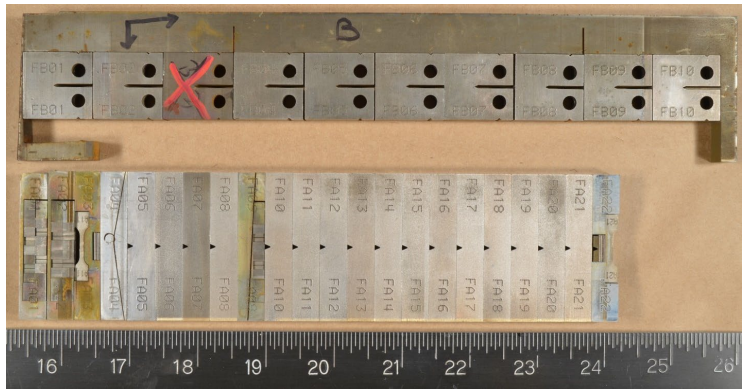


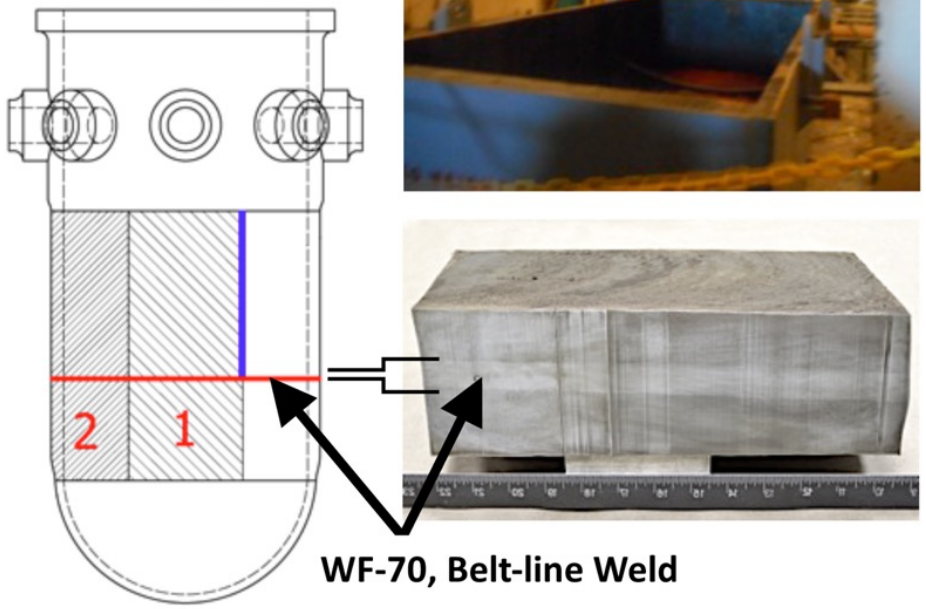
Friction stir welding of irradiated 304L

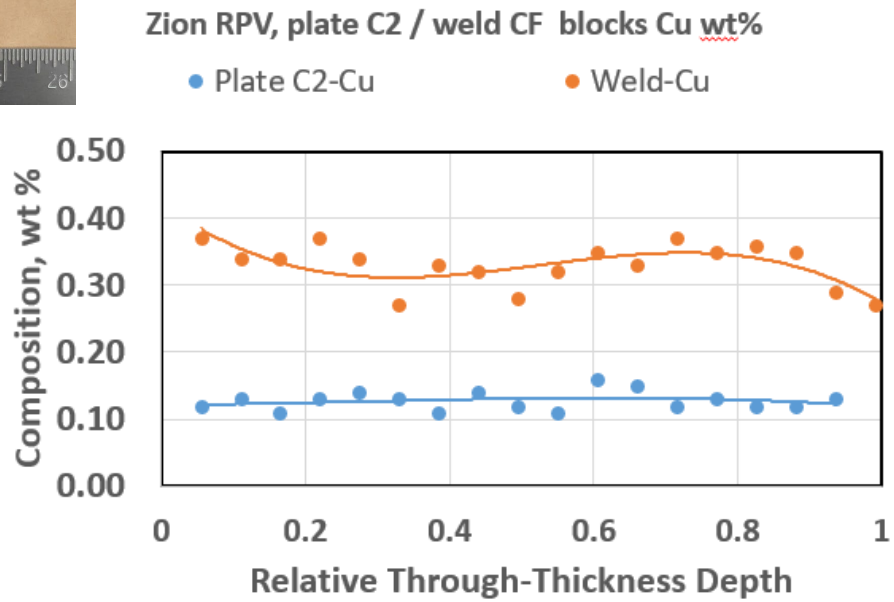
Harvesting and Integrated Research

- **Zion RPV: ORNL, Rosseel & Sokolov**
  - Evaluate radiation damage models and compare results to surveillance and test reactor experiments
  - Evaluate attenuation and through wall variations in properties and composition of the base metal and the belt-line







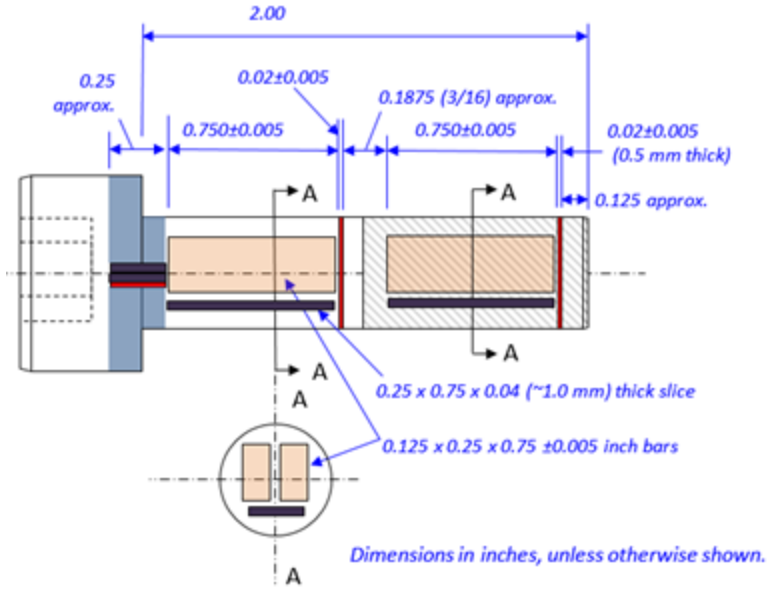


Harvesting → Machining → Testing

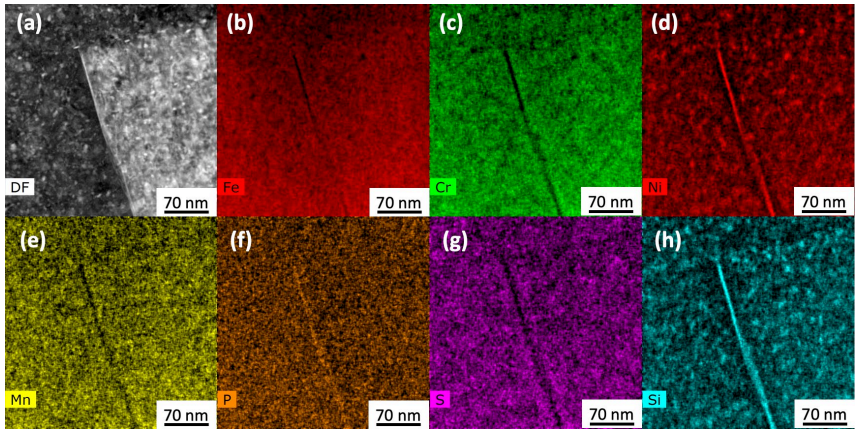
Harvesting and Integrated Research

- **Baffle former bolts : ORNL, Chen**
  - Provide critical information for evaluating end of life microstructure and properties as a benchmark of international models developed for predicting radiation-induced swelling, segregation, precipitation

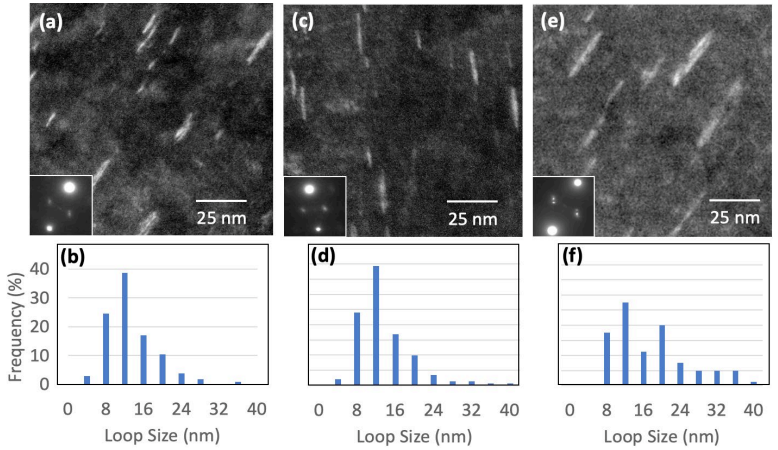
Bolt #	Fluence (10^{22} n/cm^2, E>1 MeV)/Estimated dpa		
	Head	Mid-shank	Mid-thread
4412	2.78/41	2.27/34	1.46/22
4416*	1.91/29	1.56/23	1.00/15



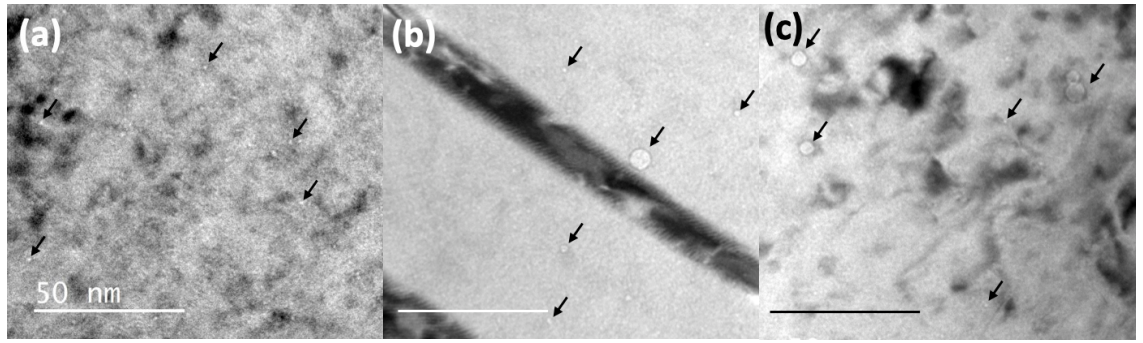
Baffle former bolt machining plan



STEM/EDS mapping showing RIS



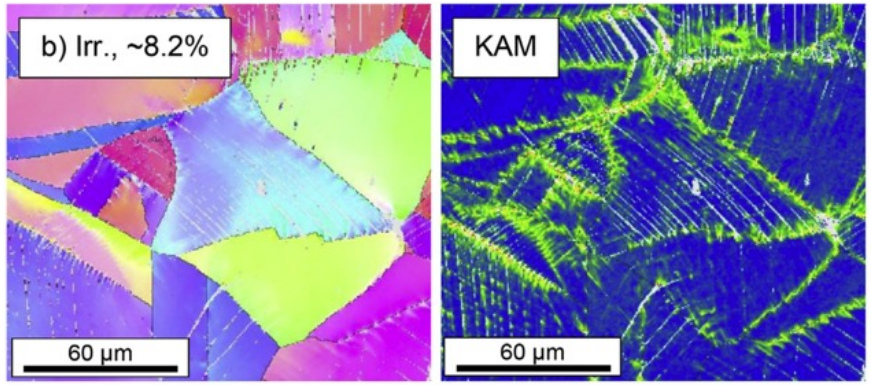
Dislocation loop characterization



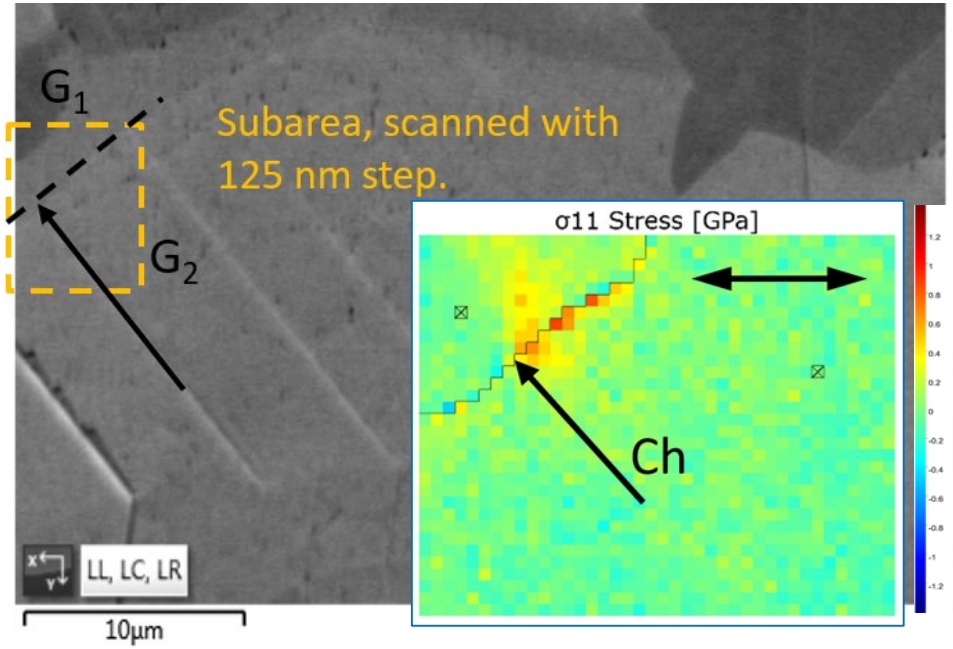
Cavity characterization

Core Internals and Piping

- **IASCC: ORNL, Gussev**
  - Investigate strain localization mechanisms and internal stress evolution in irradiated austenitic steels via in-situ testing to understand their contribution to IASCC

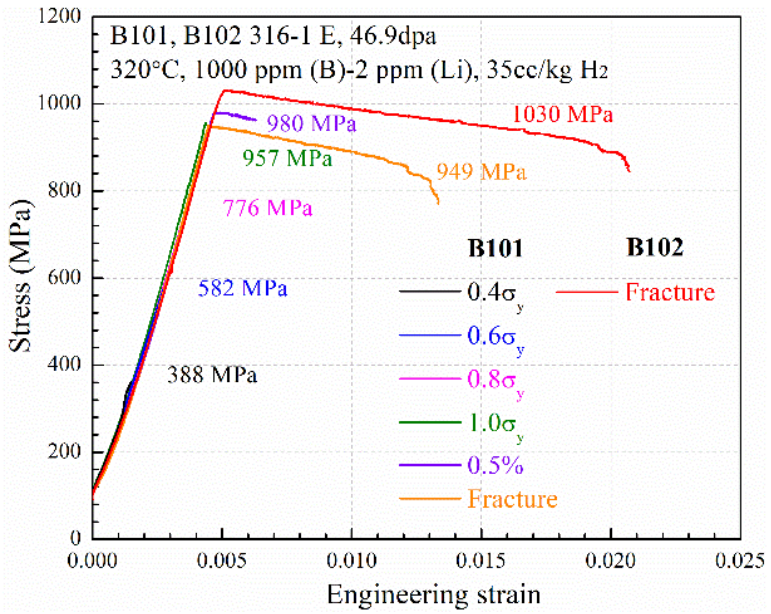


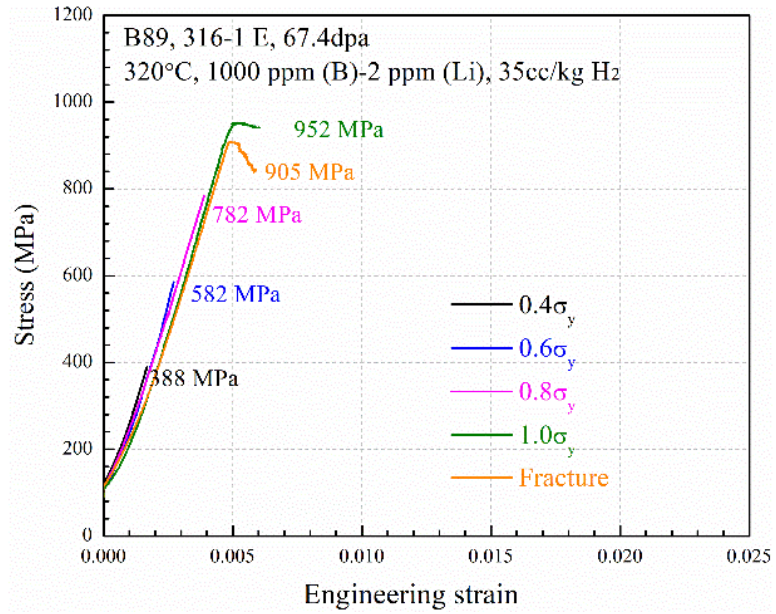
EBSD show strong strain localization near grain boundaries [JNM, 517, 2019]



HR-EBSD for measuring stress/strain localization

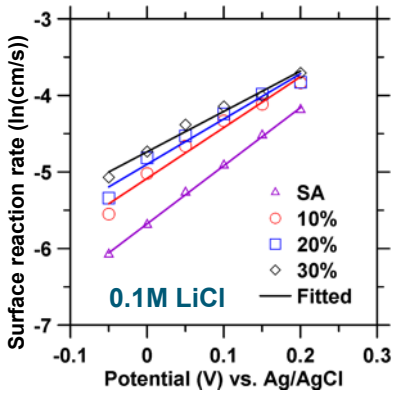
- **IASCC: U. Michigan, Was**
  - Evaluate the IASCC susceptibility of stainless steels irradiated to high dose

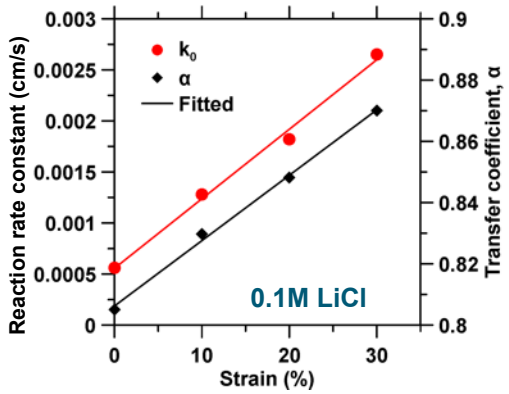


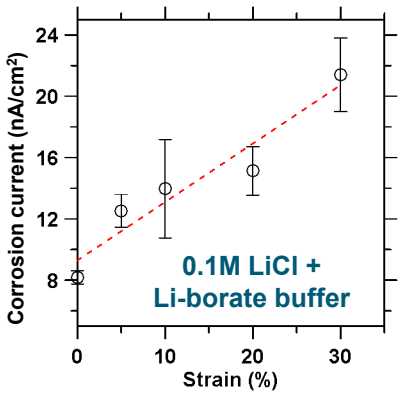


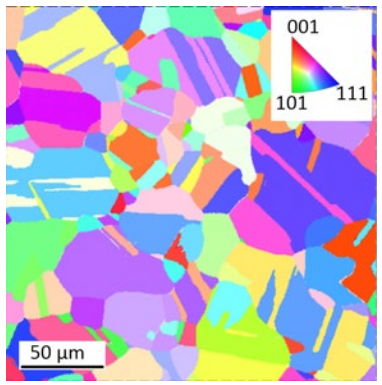
Core Internals and Piping

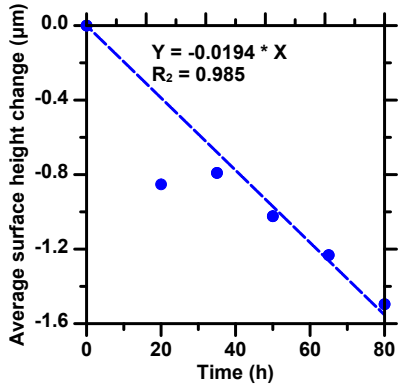
- **SCC in stainless steel: UCLA, Sant**
  - Assess and quantify the effects of stain and grain orientation on the corrosion rates of stainless steel

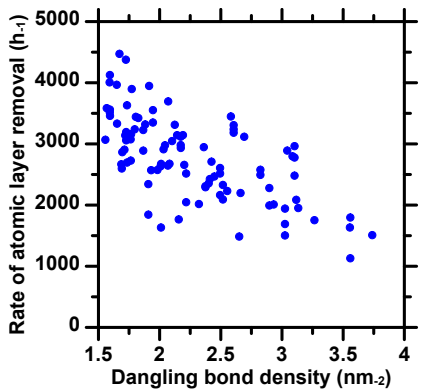




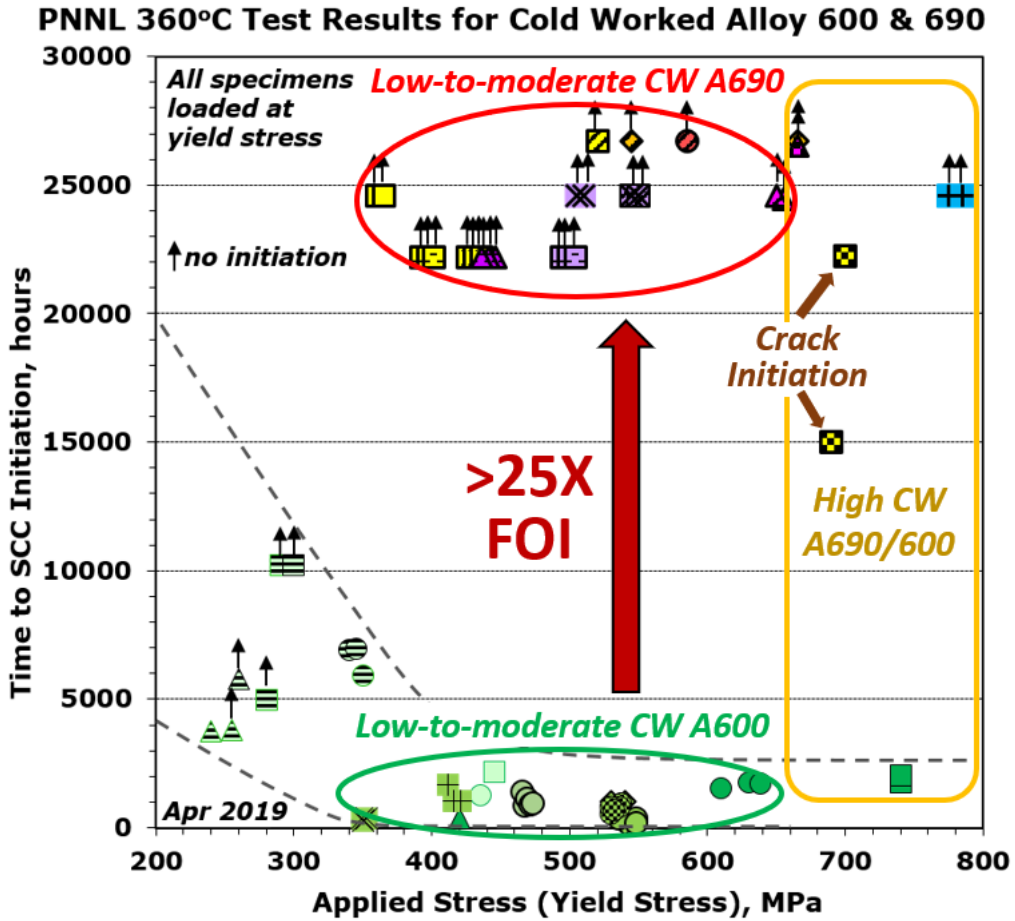






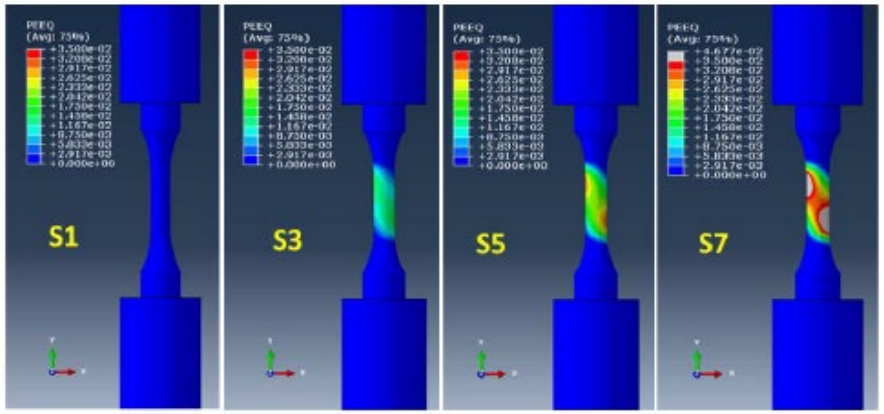


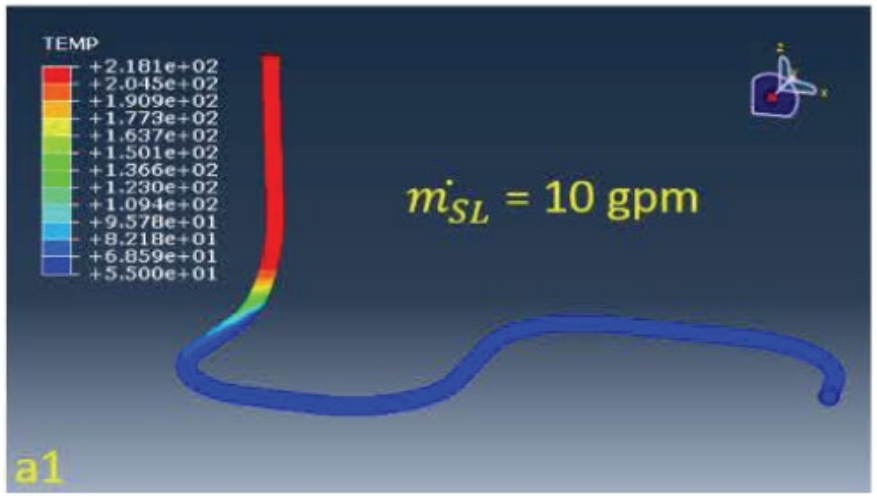
- **SCC in Ni alloys: PNNL, Bruemmer & Zhai**
  - Study SCC initiation mechanisms of Ni-based alloys in LWR systems



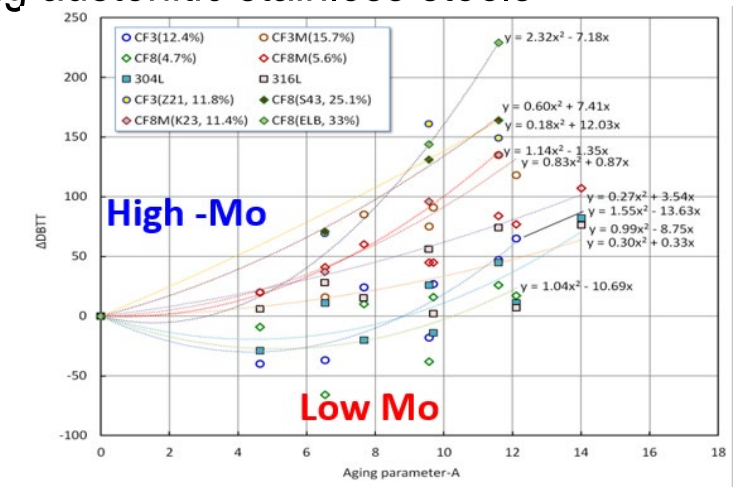
Core Internals and Piping

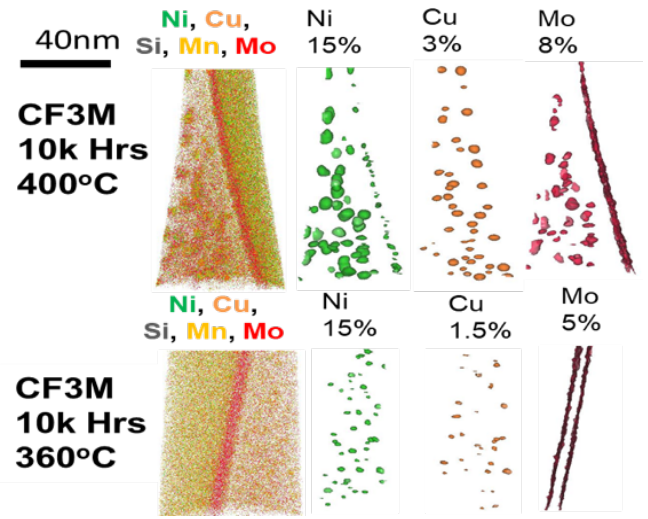
- **Environmentally assisted fatigue: ANL, Natesan & Mohanty**
  - Combine testing and modeling for LTO life estimation of primary-pressure-boundary reactor components with dissimilar weld nozzles



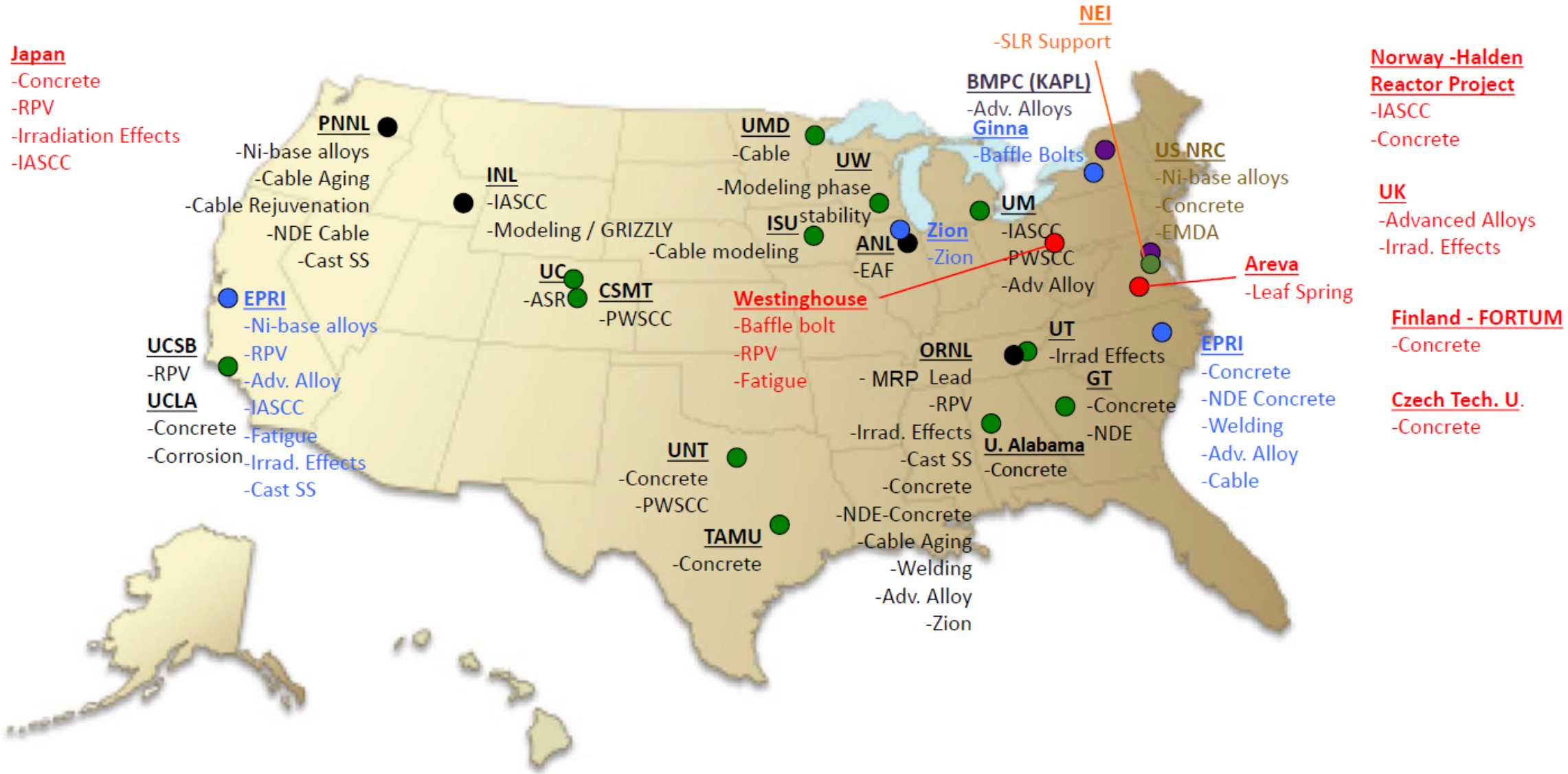


- **Cast stainless steel aging: PNNL, Byun**
  - Study thermal degradation of mechanical properties of casting austenitic stainless steels





Other Activities



Sustaining National Nuclear Assets

http://lwrs.inl.gov

Materials Research Pathway Contact:

Thomas Rosseel, Pathway Lead, 865-574-5380, rosseeltm@ornl.gov

Xiang (Frank) Chen, Pathway Deputy Lead, 865-574-5058, chenx2@ornl.gov