LWRS Overview and Select Research Highlights

J. Busby and K. Leonard

Oak Ridge National Laboratory

with a host of contributors

Overview Presentation
for the Nuclear Energy Enabling Technologies
September 15, 2015 Video Conference

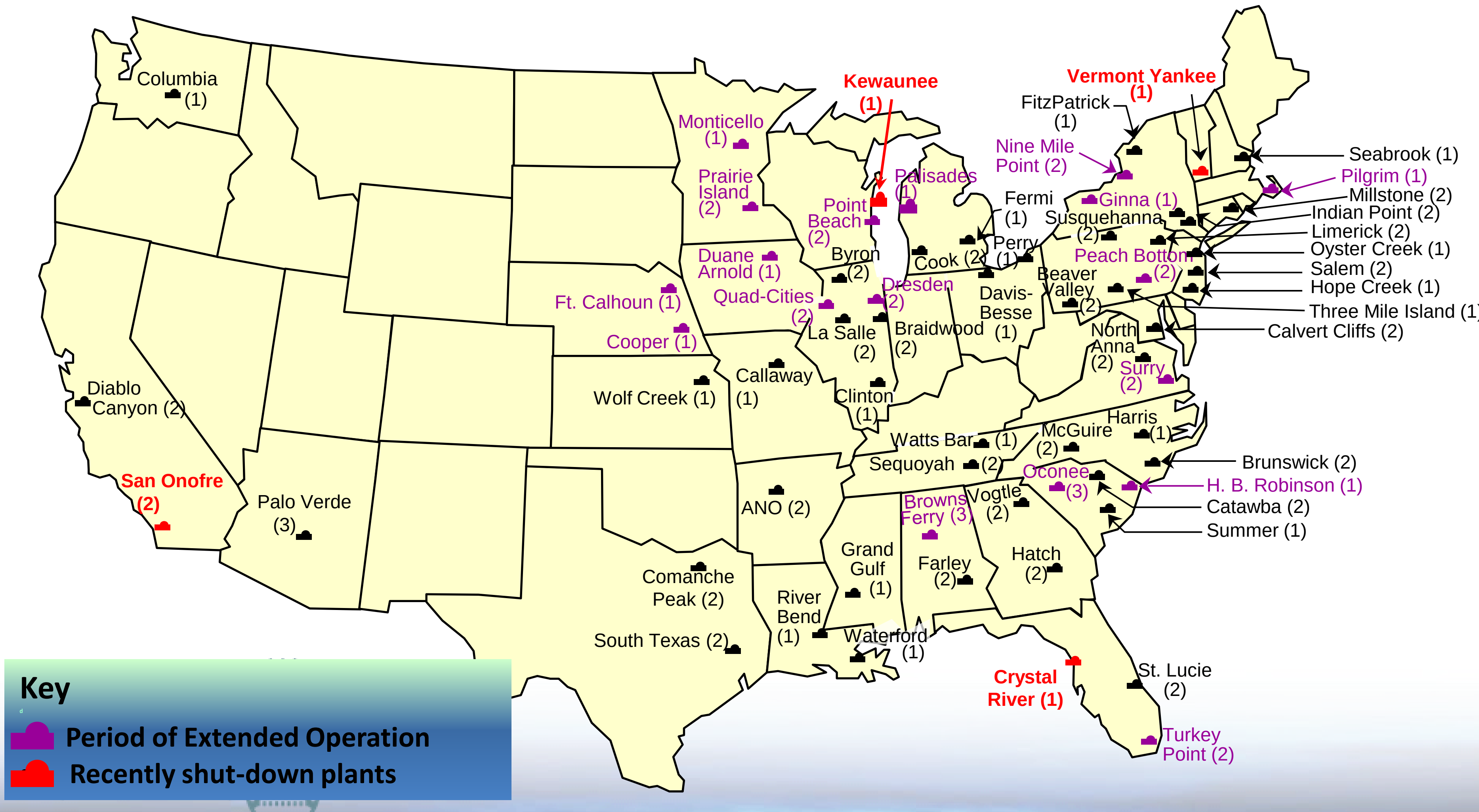
Light Water Reactor Sustainability R&D Program

FY16 - Consolidated Innovative Nuclear Research

- Light water reactor sustainability (LWRS) is one of the four technical areas in the Reactor Concepts Research, Development and Demonstration Program sections in the FOA.
- Activities in LWRS also overlap with NEET programs that support and enable development of new and advanced reactor designs, particularly in the area of materials.
- NEET-NSUF-1.1a Targeted Irradiations of LWR Core Internal Materials
- NEET-NSUF-1.1b Gamma Irradiation of LWR Cables.
- NEET-NSUF-1.1c Irradiation of LWR Weld Material.
- Also, in the Program Supporting: Nuclear Reactor Technologies section - RC4 Materials Aging and Degradation.

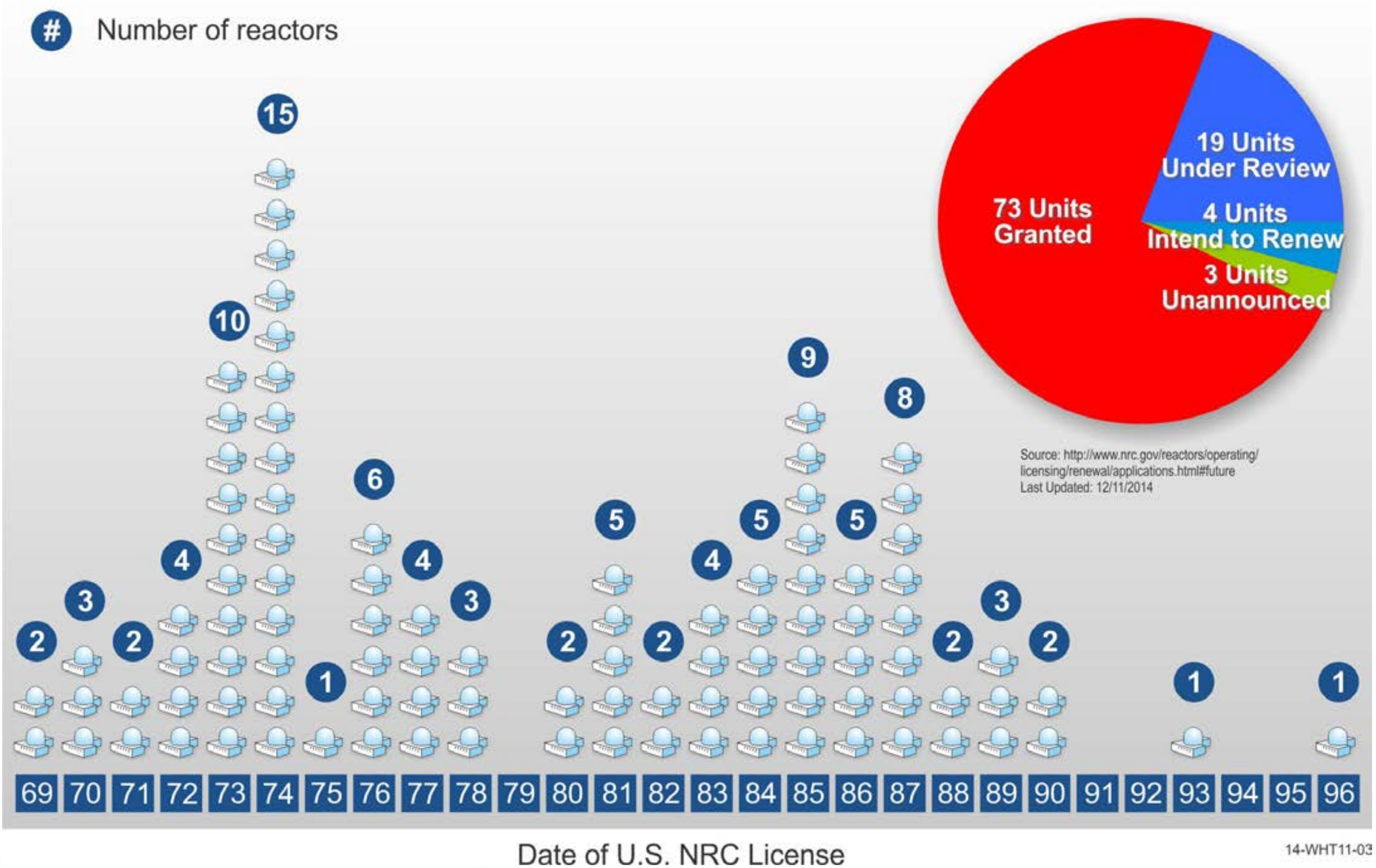
LWRS
Light Water Reactor Sustainability

There are 99 U.S. Nuclear Power Plants in operation today



Extended Operations of the Existing Reactor Fleet is in the National Interest

- Our LWRs are a national asset; without them, we lose:
  - ~100 Gwe of low-carbon generation over about 20 years
  - Low-cost generation
- It is unlikely that new plants can be built quickly enough to both replace LWR retirements and meet demand for new clean electricity



The DOE-NE Light Water Reactor Sustainability Program Supports Decisions on Long-Term Operation

Vision

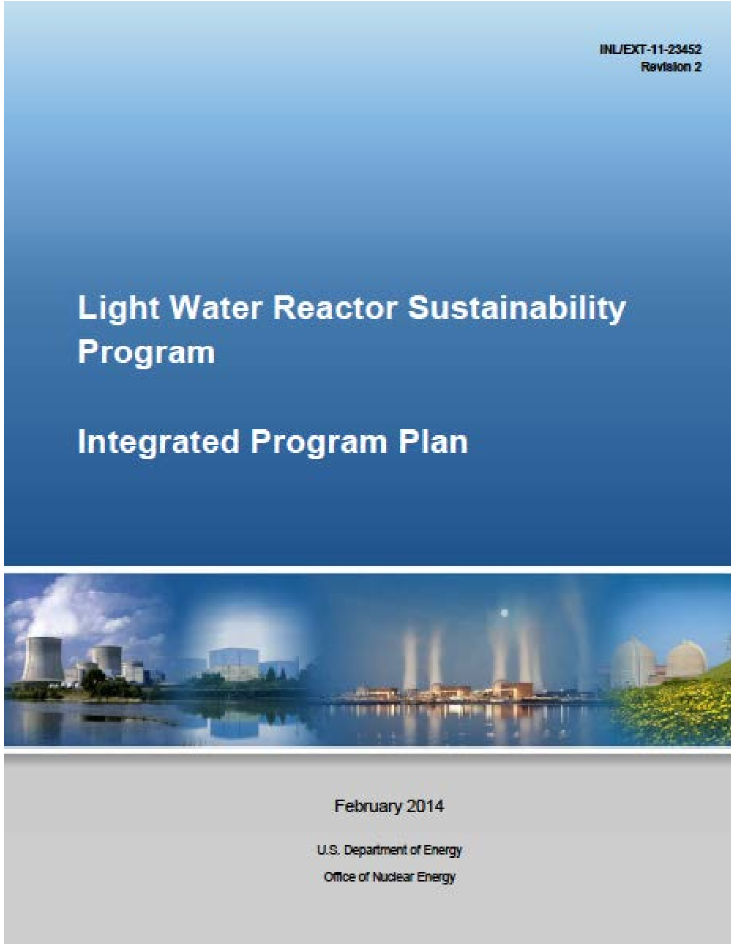
- *Enable existing nuclear power plants to safely provide clean and affordable electricity beyond current license periods (beyond 60 years)*

Program Goals

- Develop fundamental scientific basis to understand, predict, and measure changes in materials as they age in reactor environments
- Apply this knowledge to develop methods and technologies that support safe and economical long-term operation of existing plants
- Research new technologies that enhance plant performance, economics, and safety

Scope

- Materials Aging and Degradation
- Advanced Instrumentation and Controls
- Risk-Informed Safety Margin Characterization
- Reactor Safety Technology
- *Advanced Nuclear Fuels*



INL/EXT-11-23452
Revision 2

Light Water Reactor Sustainability Program

Integrated Program Plan

February 2014

U.S. Department of Energy
Office of Nuclear Energy

LWRS Integrated Program Plan (INL/EXT-11-23452, Rev. 2) Available on www.inl.gov/lwrs

Materials issues are a key concern for the existing nuclear reactor fleet

- Materials research is already a key need for the existing nuclear reactor fleet
- Materials degradation can lead to increased maintenance, increased downtime, and increased risk.
- Materials issues must be resolved for:
  - Reactor Pressure Vessels and Primary Piping
  - Core Internals
  - Secondary System
  - Weldments
  - Concrete
  - Cabling
  - Buried Piping

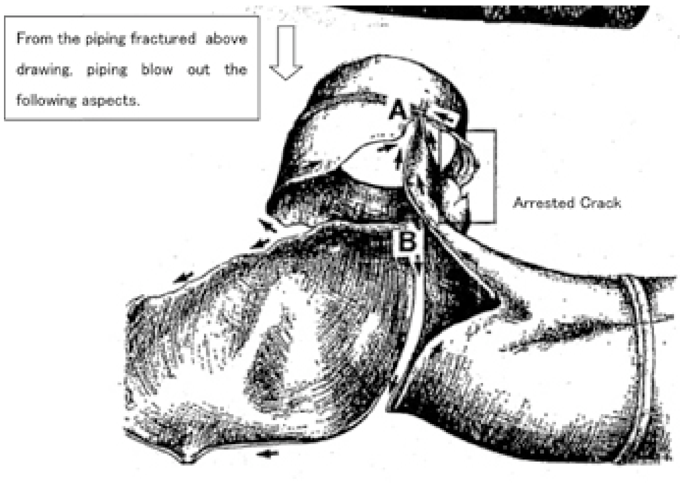
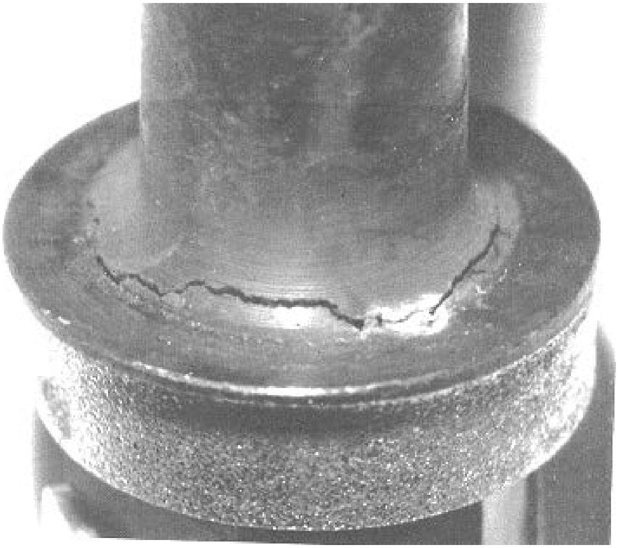


Fig.3 Detail Drawing of the Breaking Portion

Light Water Reactor Sustainability

Extension of service life may cause new challenges for materials service

- Increased lifetime leads to increased exposures
  - Time at temperature
  - Stress
  - Coolant
  - Neutrons
- Extending reactor life to 60, 80 years or beyond will likely increase susceptibility and severity of known forms of degradation
- New mechanisms of materials degradation are possible



- The motivation of the Materials Aging and Degradation Pathway is to deliver fundamental understanding to enable and support nuclear power plant (NPP) life extension decisions in a timely manner

Light Water Reactor Sustainability

There are many materials in a modern power reactor

Steam separator and dryer:
- components, 304
- welds 308L

Steam outlet nozzle:
- carbon steel

Closure studs:
- alloy steel

Steam piping:
- carbon steel

Turbine:
- rotor, low alloy steel
- blades, alloy steel
- attachments, low alloy steel

Vessel plates:
- low alloy steel;
- clad: 309 SS

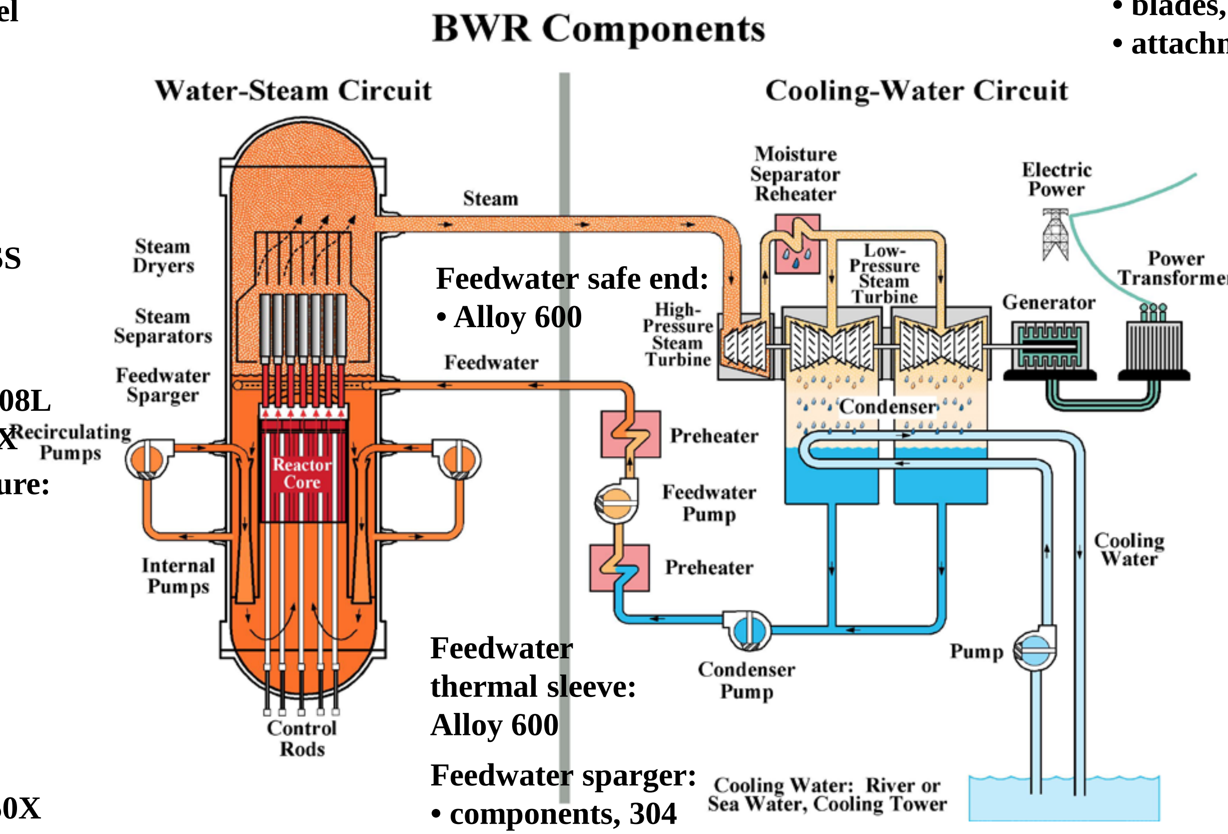
Fuel supports:
- wrought or cast SS

Core shroud:
- plates, 304 SS
- plate welds, 308/308L
- springs, alloy 750X

Core support structure:
- 304 SS

Fuel:
- Cladding, Zr-2
- Fuel, UO_2

Jet pumps:
- piping, 304 SS
- welds, 308L
- hold down bar: 750X



Feedwater safe end:
- Alloy 600

Condenser structure:
- carbon steel

Condenser tubing:
- Ti or SS

Preheater tubing:
- SS

Feedwater thermal sleeve:
Alloy 600

Feedwater sparger:
- components, 304
- welds, 308L

Control rods:
- blades, 304/316 SS
- poison, B_4C

CRD stub tubes:
- Alloy 600
- upper weld, 182
- Inconel to Inconel: 182

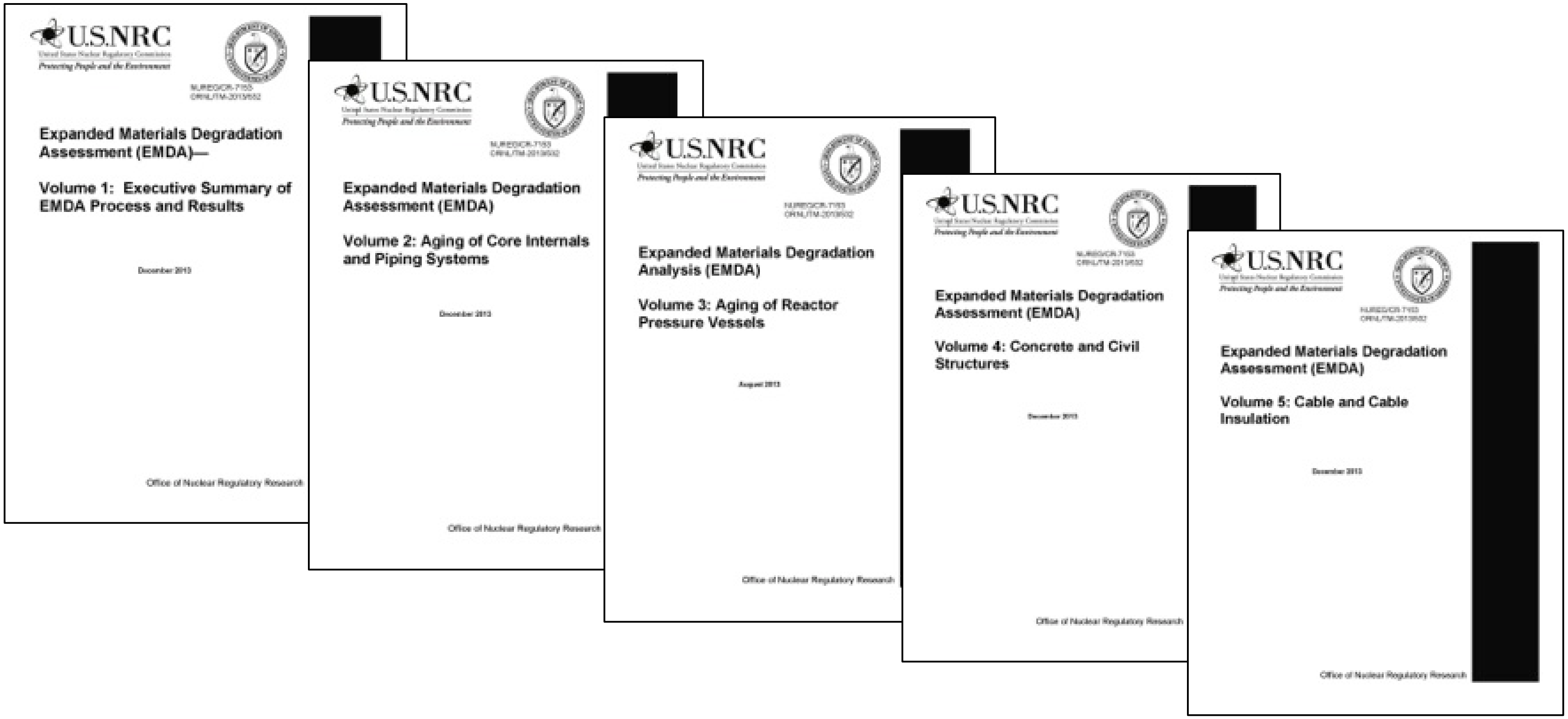
CRD safe ends:
- 316 SS

Feedwater piping:
carbon steel

Dissimilar metal welds
- Safe ends: 182/82

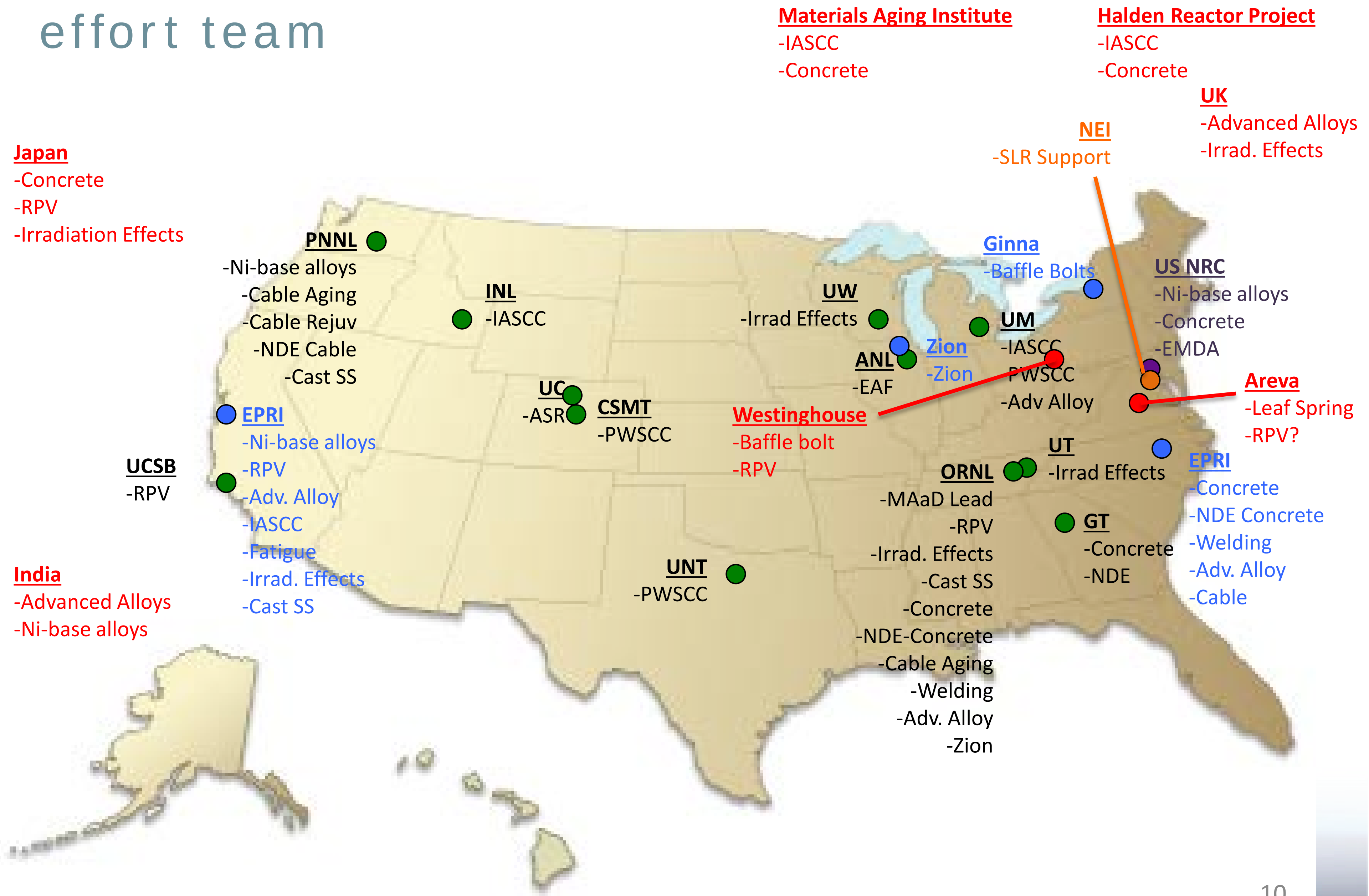
Source: R. Staehle

NRC and DOE have investigated issues of reactor aging beyond 60 years to identify possible knowledge gaps



Final PUBLISHED version of NUREG CR7153 is now available on-line!

MAaD includes a diverse materials research effort team



Materials Aging and Degradation tasks provide results in several ways

- **Measurements of degradation**: High quality data will provide key information for mechanistic studies, but has value to regulators and industry on its own.
- **Mechanisms of degradation:** Basic research to understand the underlying mechanisms of selected degradation modes will lead to better prediction and mitigation.
- **Modeling and simulation:** Improved modeling and simulation efforts have great potential in reducing the experimental burden for life extension studies. These methods can help interpolate and extrapolate data trends for extended life.
- **Monitoring:** While understanding and predicting failures are extremely valuable tools for the management of reactor components, non-destructive monitoring must also be utilized.
- **Mitigation strategies:** While some forms of degradation have been well-researched, there are few options in mitigating their effects. New technologies may overcome limits of degradation in key components and systems.

LWRS Areas of Research
Include both Experimental and Theoretical Efforts

- RPV Steels
- Mechanisms of IASCC, Crack Initiation in Ni-Base Alloys
- High Fluence Effects on Core Internal Materials
  - Swelling, phase transformations, IASCC
- Environmentally Assisted Fatigue
- Advanced Replacement Alloys
- Concrete
  - Irradiation effects, Alkali Silica Reactions, NDE Techniques
- Cable Aging and Degradation
- Advanced Weld Repair
- Analysis of Reactor Harvested Materials

LWRS
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Materials Aging and Degradation Pathway Major Deliverables

by 2016 – Lead Plant for SLR Selected

by 2018 – First SLR License Application Submitted

by 2020 – First SLR License Approved by NRC

- **Expanded Materials Degradation Assessment**
- **Model for transition temperature shifts in RPV steels**

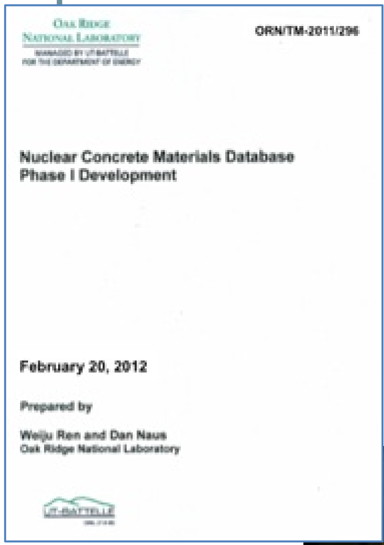
- **Model for environmentally-assisted fatigue in LWR components**
- **Model for cable degradation**
- **Model for cast austenitic**

- **Predictive capability for end of useful life for cable insulation**
- **New or improved NDE technologies for concrete and cable**

- **Complete characterization of demonstration of RPV sections following annealing and reirradiation**
- **Complete development and testing of new advanced alloy with superior degradation resistance with ARRM partners**

Concrete and Civil Structures Research

Concrete structures are of interest for long-term operation


ORNL/TM-2011/296

Nuclear Concrete Materials Database
Phase I Development

February 20, 2012

Prepared by

Weiju Ren and Dan Naus
Oak Ridge National Laboratory

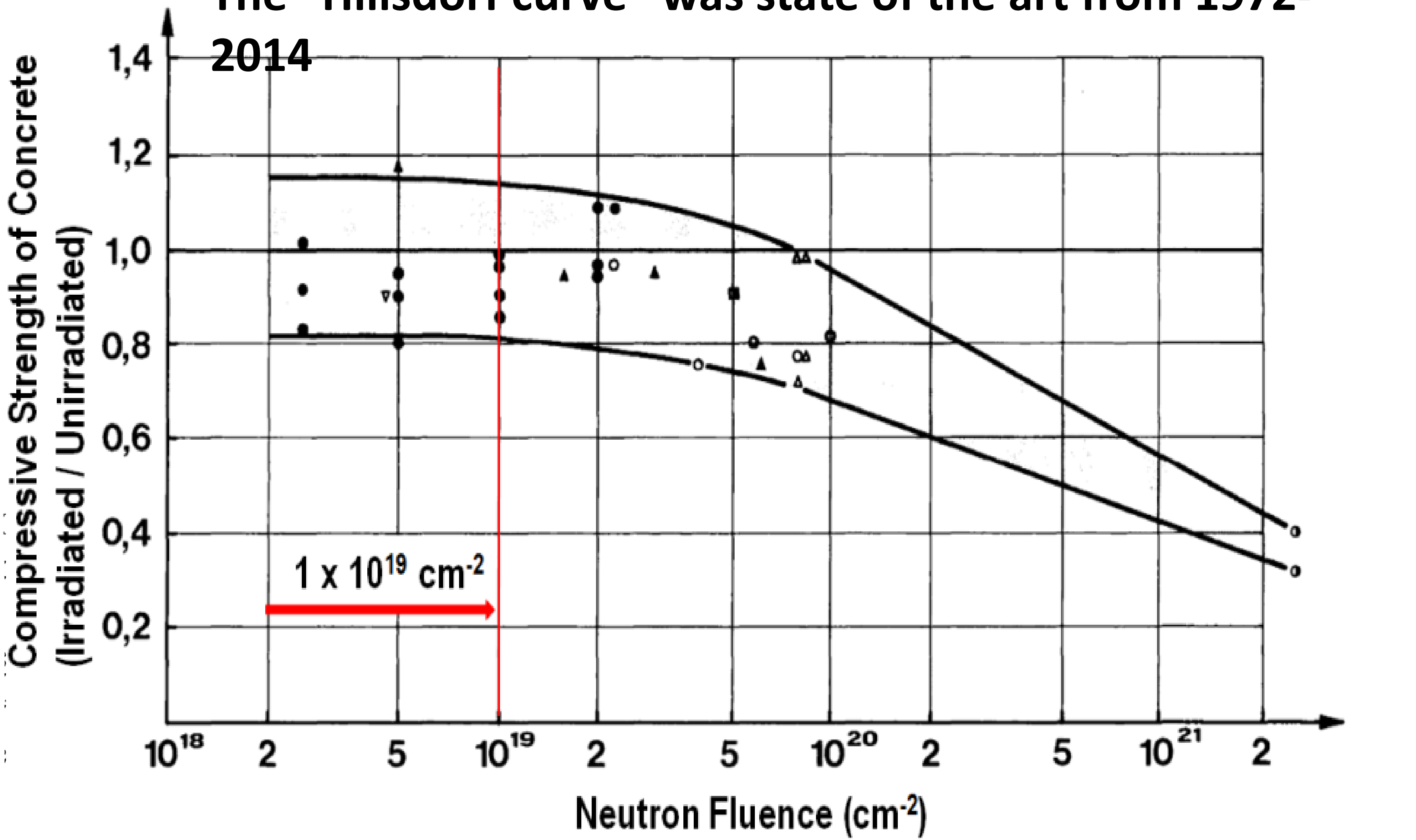
ORNL/TM-2011/296

Concrete coring to obtain samples for evaluating effects of aging and environmental stressors

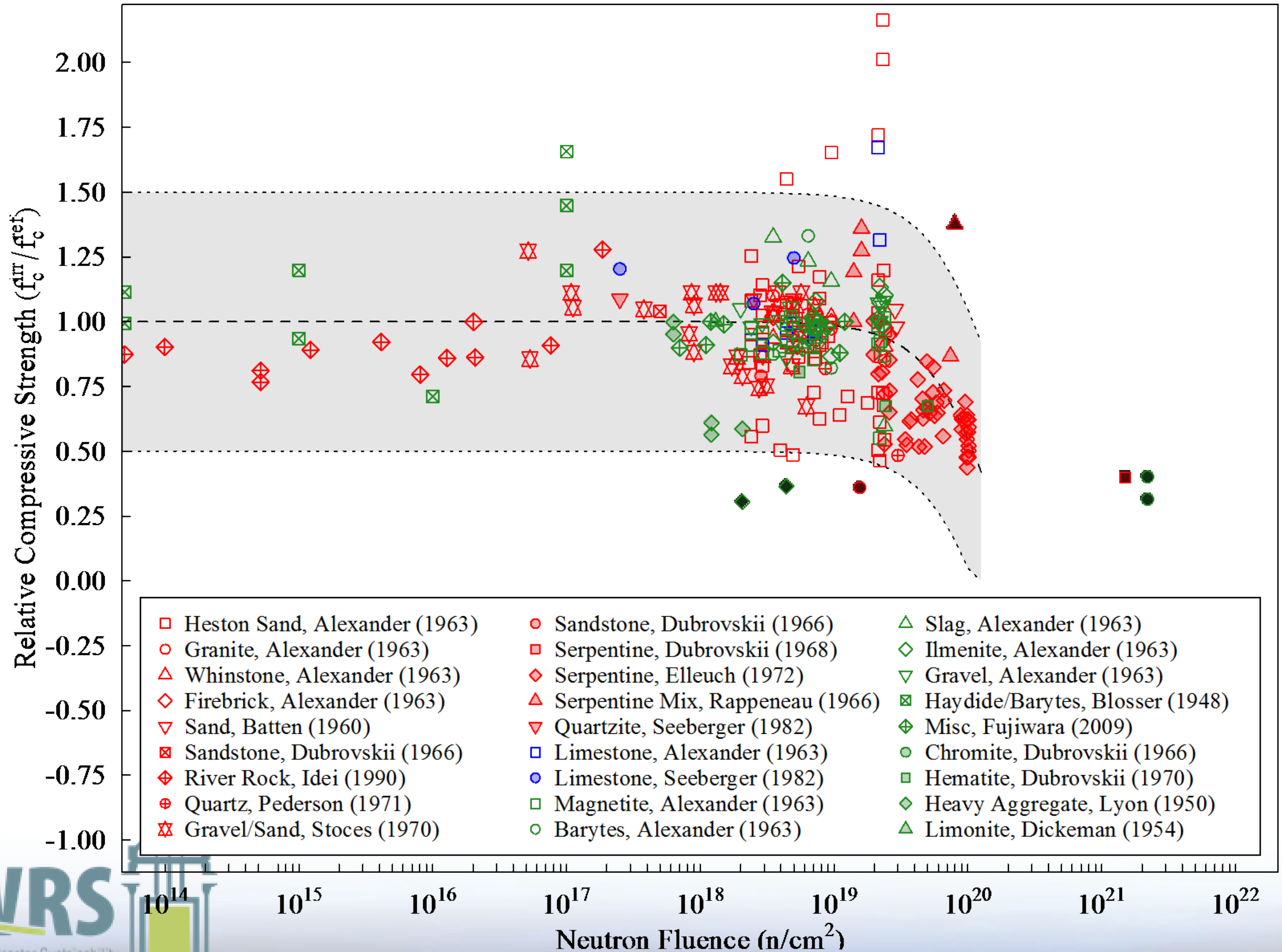
- A number of potential environmental effects are important for long-term aging
  - Aging
  - Elevated temperature
  - Irradiation
  - Migration of hostile species (e.g., Cl^-, SO_4, CO_2)
- Phase I of NCMDB completed, additional information within EMDA.
- Concrete irradiation damage working group formed
  - Development of protocols related to removal and testing of irradiated concrete cores
  - Identification of potential sources of irradiated concrete cores

A major effort has yielded a new database and better understanding of concrete performance

The “Hillsdorf curve” was state of the art from 1972-2014



This improved database also allows for examination of complicated variables



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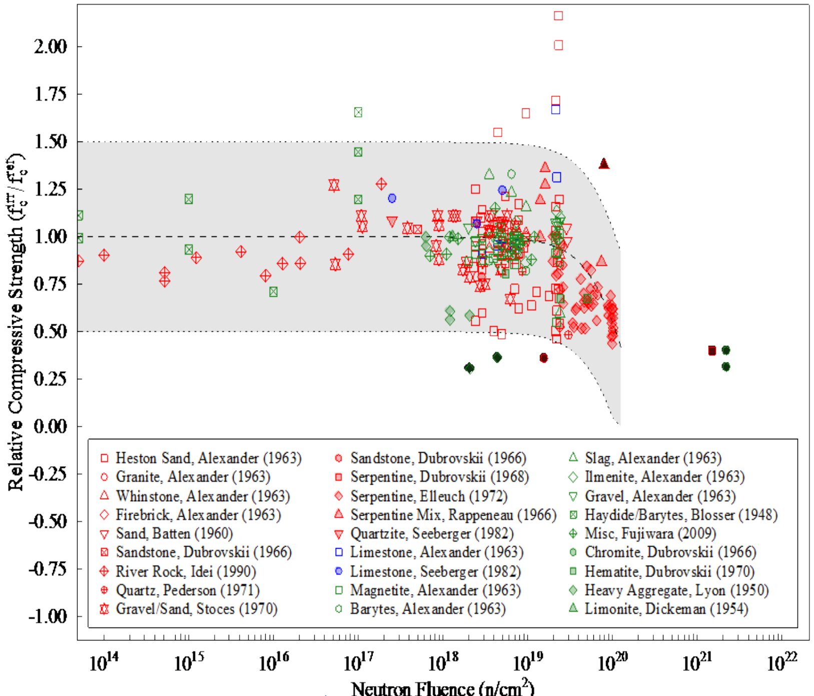
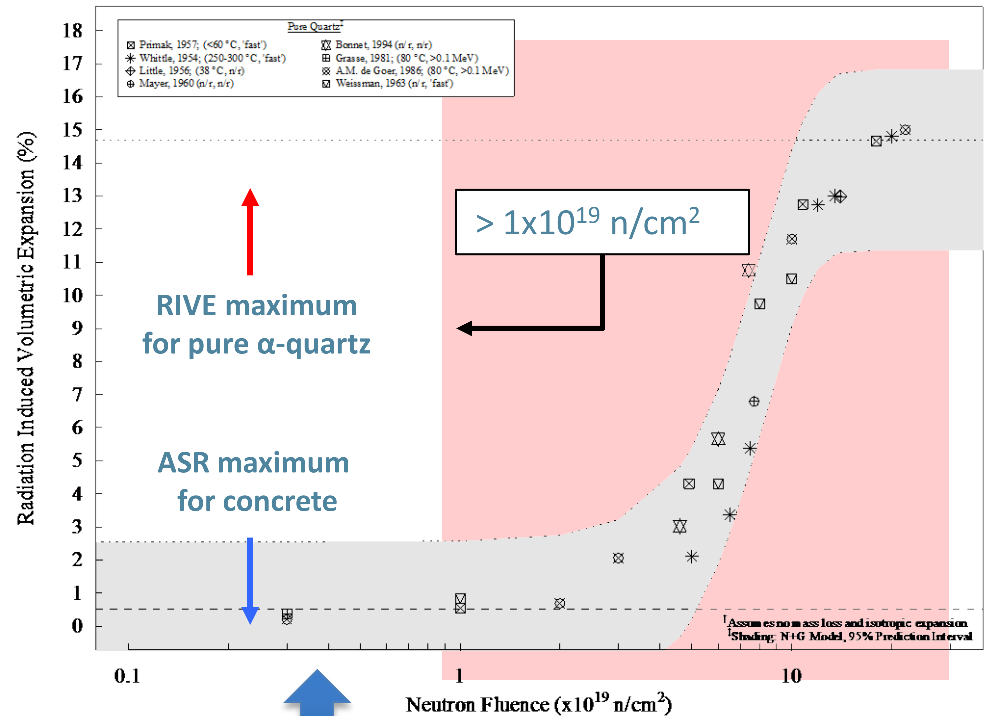
Major Effort Yielded an Expanded Database that Suggested Controlled Irradiation Experiments, & Foundation for Improved Models





300 data points culled from reports, theses, and previously hard to find data

Quartz expands as it undergoes amorphization or Radiation Induced Volumetric Expansion (RIVE)

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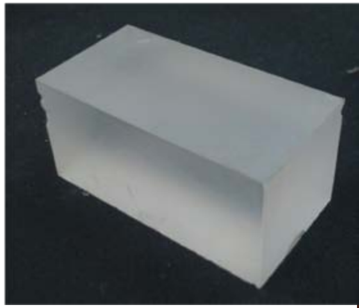
Field, LePape, ORNL

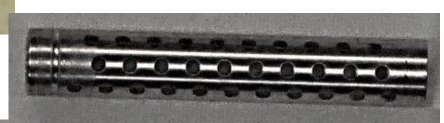
Preliminary mineral and hardened cement paste HFIR irradiations have now been completed



Quartz

Calcite

Dolomite

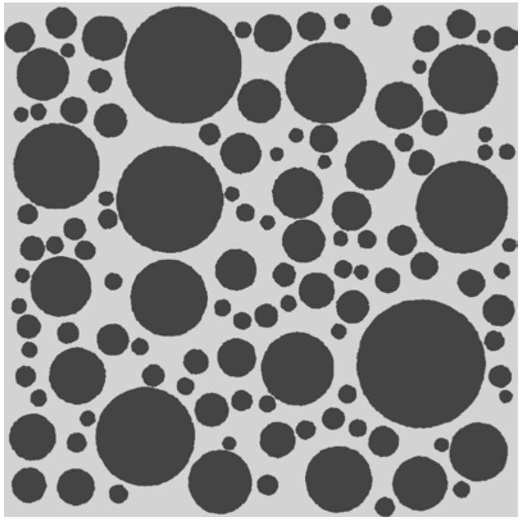


Perforated capsule

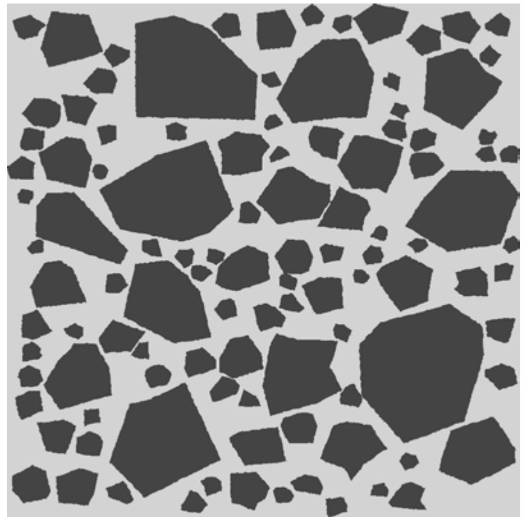
Sealed capsule

- Irradiation of hardened cement paste and single crystals representative of the most common aggregate mineral components
  - Quartz *SiO_2*
  - Calcite *$CaCO_3$*
  - Dolomite $Ca\mathit{Mg(CO_3)_2}$
- Two different irradiation capsule designs to test irradiation temperature control
  - Sealed
  - Perforated to allow best heat transfer with the core coolant
- Fast neutron fluences
  - 0.5×10^{19} n/cm^2 (E > 0.1 MeV)
  - 4×10^{19} n/cm^2 (E > 0.1 MeV)
  - 20×10^{19} n/cm^2 (E > 0.1 MeV
- Post Irradiation Evaluation (PIE) begun
- Complementary ion irradiation study and PIE completed.

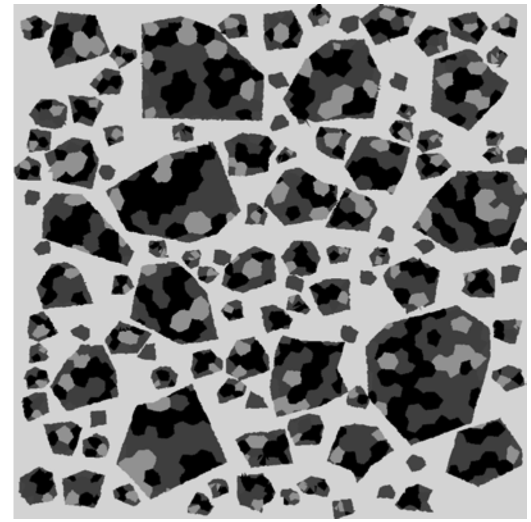
Toward modeling of realistic microstructures for concrete irradiation



- Circles only
- Homogeneous control
    - particle size,
    - volume fraction,
    - and spacing



- Various shapes available
- Possibility to mix different shapes
- Possibility for embedded inclusions
- Control over particle elongation
- Control over particle orientation
- Material properties can be adjusted



Multi-phasic aggregates
- using Voronoi diagrams
- Control phase size
- Control phase fraction
- Control over grain boundary
- Highly complex microstructures

Increasing complexity

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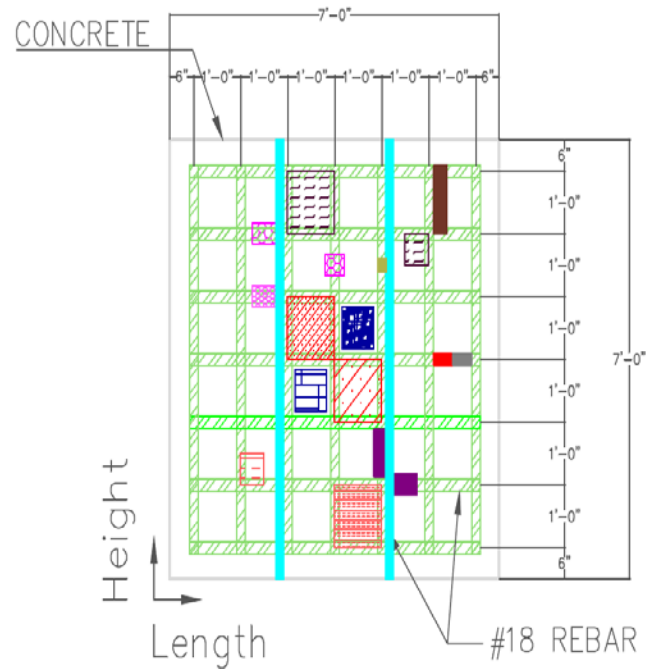
Improved NDE Test Techniques

- Thick specimen representative of a NPP Containment fabricated
- Linear Array Ultrasonic data collected and analyzed
- Ground Penetrating Radar data collected
- Results
  - Frequency banded SAFT yields improved results over “normal” SAFT
  - Second harmonic analysis not as useful as hoped based on thin specimen analysis
    - More attenuation due to thicker depth and thicker rebar suspected as major causes
  - Some defects under deep cover still difficult to locate

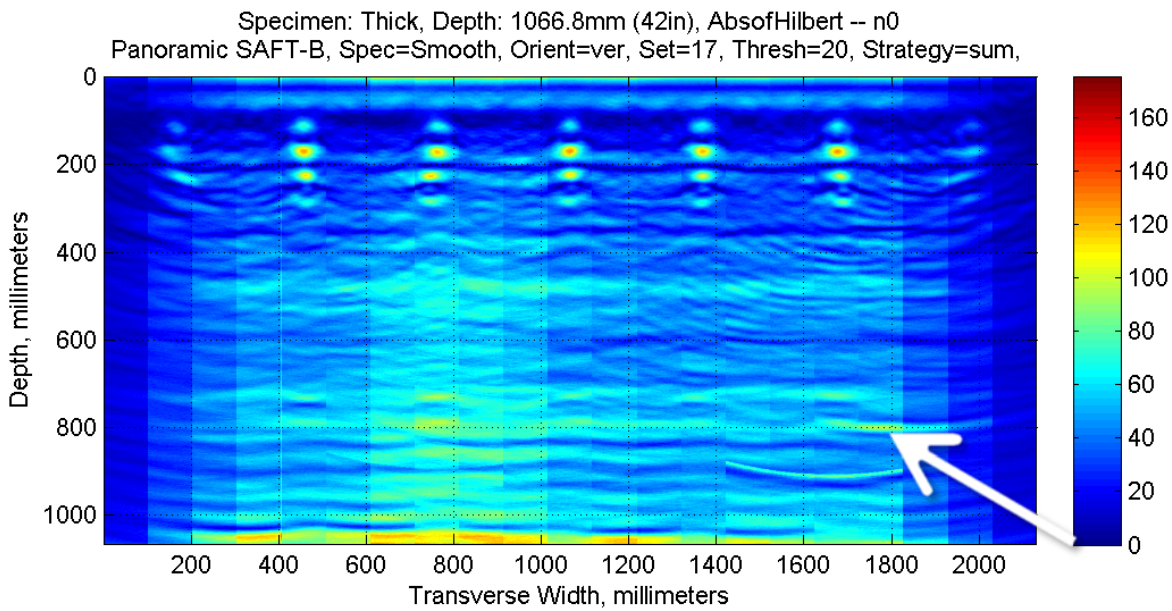
Light Water Reactor Sustainability

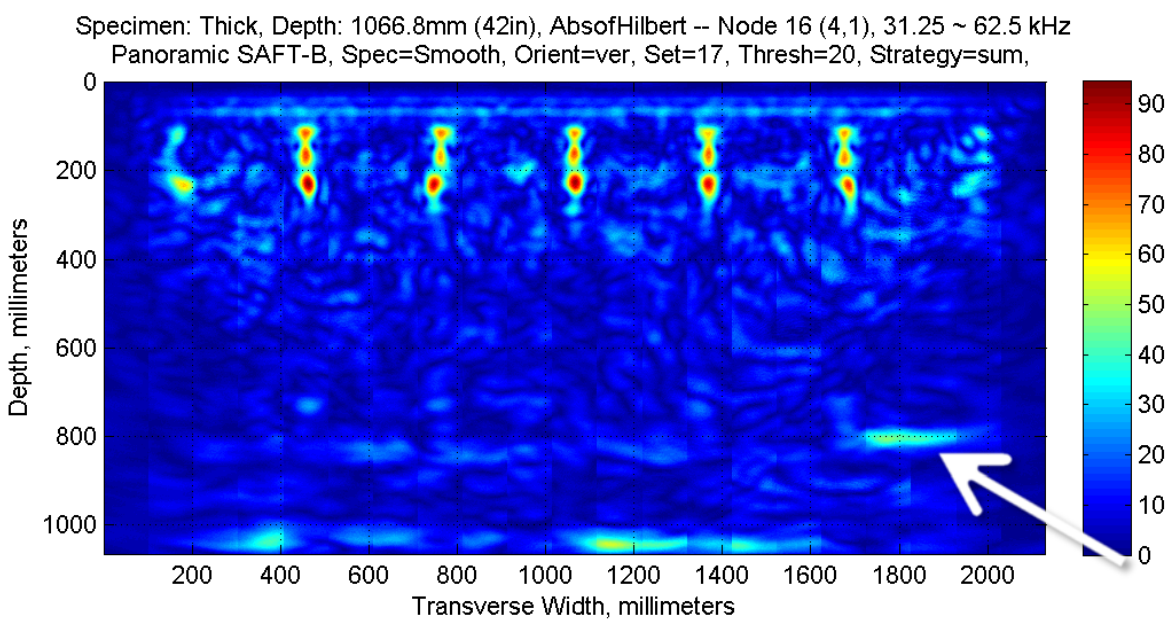
Twenty Defects were Inserted into the Large Concrete Specimen

DEFECT TABLE		
ID NUMBER	DESCRIPTION	LABEL
D1	POROUS HALF CYLINDER (NO COVER)	
D2	POROUS HALF CYLINDER (COVER)	
D3	POROUS HALF CYLINDER (NO COVER)	
D4	POROUS HALF CYLINDER (COVER)	
D5	POROUS HALF CYLINDER (COVER & CRACK)	
D6	PVC	
D7	PVC	
D8	DISSOLVING STYROFOAM (THICK)	
D9	STYROFOAM (THICK)	
D10	STYROFOAM (THIN)	
D11	PLEXIGLASS	
D12	DISSOLVING STYROFOAM (MEDIUM)	
D13	STYROFOAM (MEDIUM)	
D14	PLEXIGLASS	
D15	DISSOLVING STYROFOAM (THIN)	
D16	LUMBER (2X4)	
D17	GLOVES	
D18	DEBOND DUCT TAPE (ONE LAYER)	
D19	DEBOND DUCT TAPE (MULTI-LAYER)	
D20	MOVING REBAR	



Lumber (2x4) – deep cover





At 32" deep, normal SAFT does not detect the "defect"

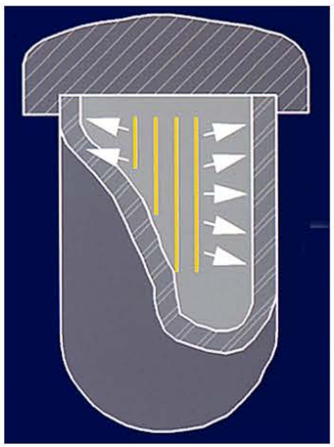
Frequency banded SAFT clearly identifies the "defect"

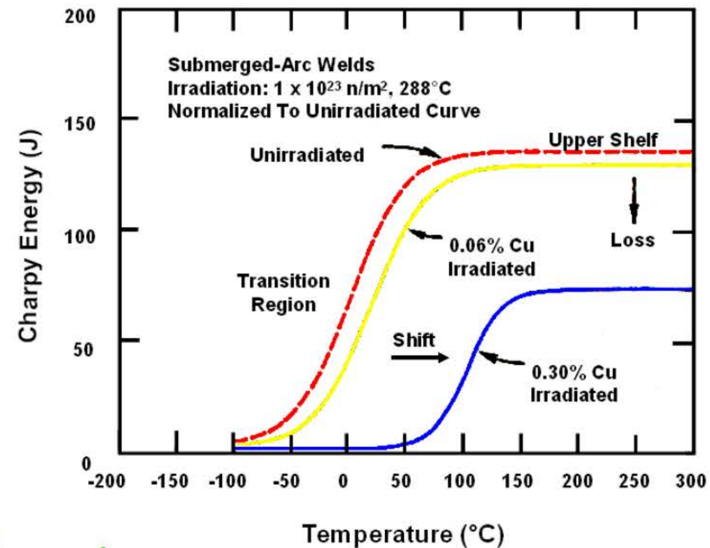
Reactor Pressure Vessel Steels

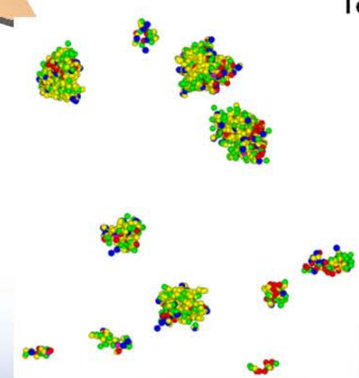
Reactor Vessel Integrity Assessments Must Account for Potential Degrading Effects of Neutron Irradiation

Irradiation Causes Ductile/Brittle Transition Temperature Shift and Upper Shelf Energy Loss — Copper, Nickel, etc. Increase The Effect

Neutron Embrittlement of RPV

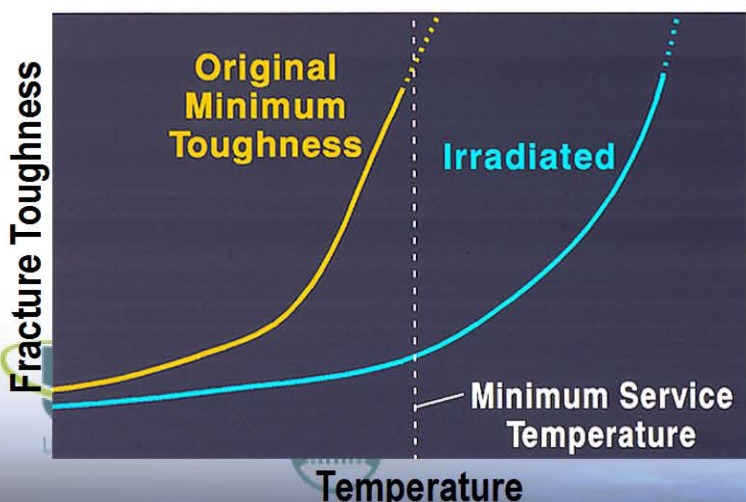






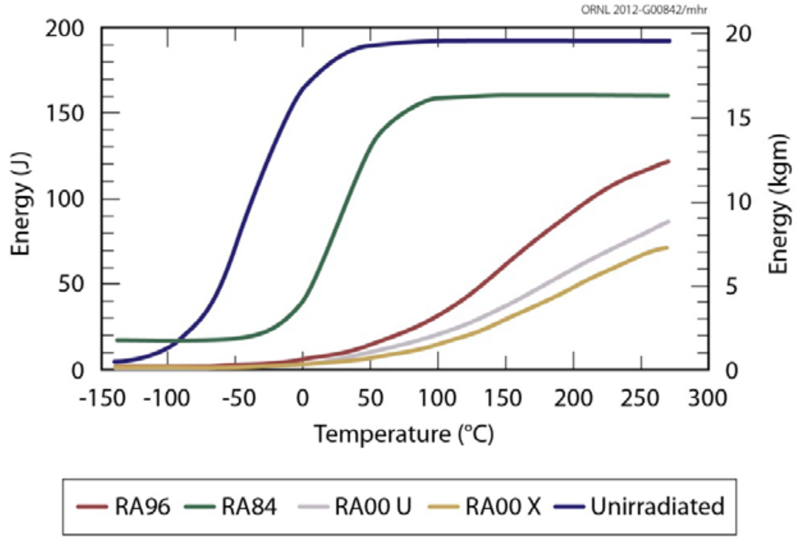
Irradiated Microstructures: Precipitates and Matrix Damage

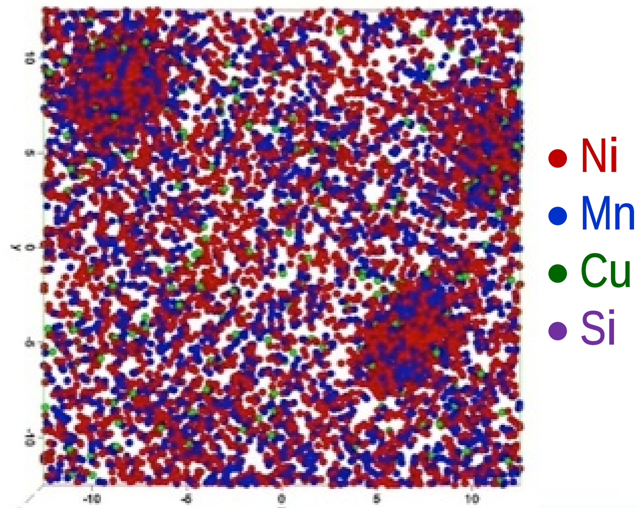
Precipitates ~2nm in diameter



"Late Blooming Phases" have been the focus of materials research

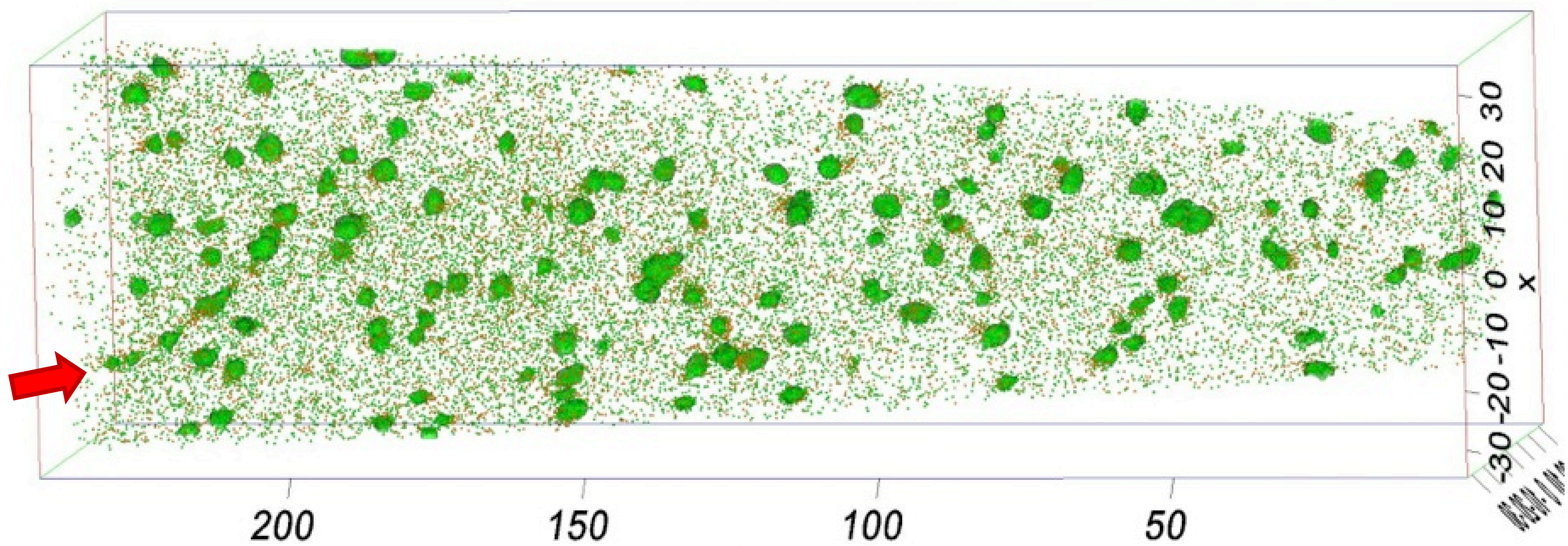
- Rapid Cu-rich precipitate hardening drives embrittlement of the reactor pressure vessel
- Modern RPV steels have low-residual Cu-levels
- Irradiation may drive phase transformations even in low Cu alloys (Odette et al.)
  - Mn-Ni(-Si-Cu) LBP that can reach large volume fractions and contribute to embrittlement
  - Could be important in low Cu steels thought to have little sensitivity to embrittlement
- RPV materials and surveillance specimens from the Ginna Nuclear Plant and from the Zion Nuclear Plants for material examination, APT, SANS, PAS



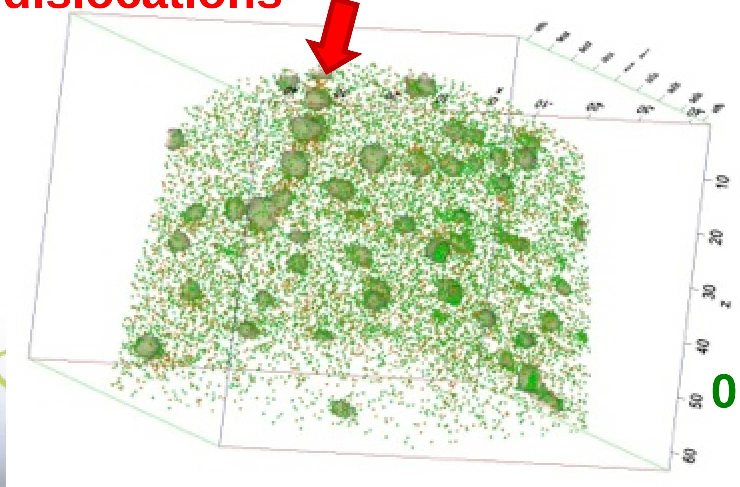


Low-copper (0.05 wt%) weld shifts 162°C at 6×10^{19}, clusters primarily of Ni-Mn-Si, very little copper.

High-copper weld in Ginna RPV exhibited Cu-Ni-Mn-Si enriched precipitates responsible for embrittlement.



dislocations

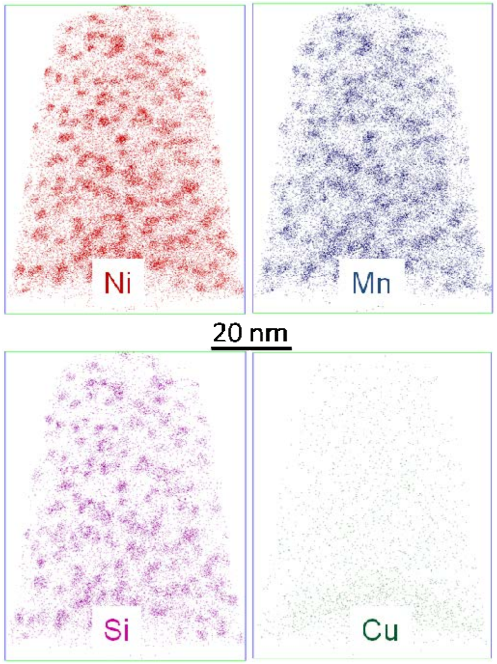


Cu-enriched precipitates and P segregation to the dislocations.

Overall concentrations (APT):
0.20% Cu, 0.60% Ni, 1.35% Mn, 0.95% Si (±0.004%)

Multiscale modeling in RPV steels is underway

- Precipitation is a leading cause of embrittlement of RPV steels and austenitic alloys under irradiation.
- To understand this phenomenon, we are developing a multiscale modeling approach that combines
  - DFT calculations
  - CALPHAD modeling
  - Mean field diffusion and precipitation theory (cluster dynamics)

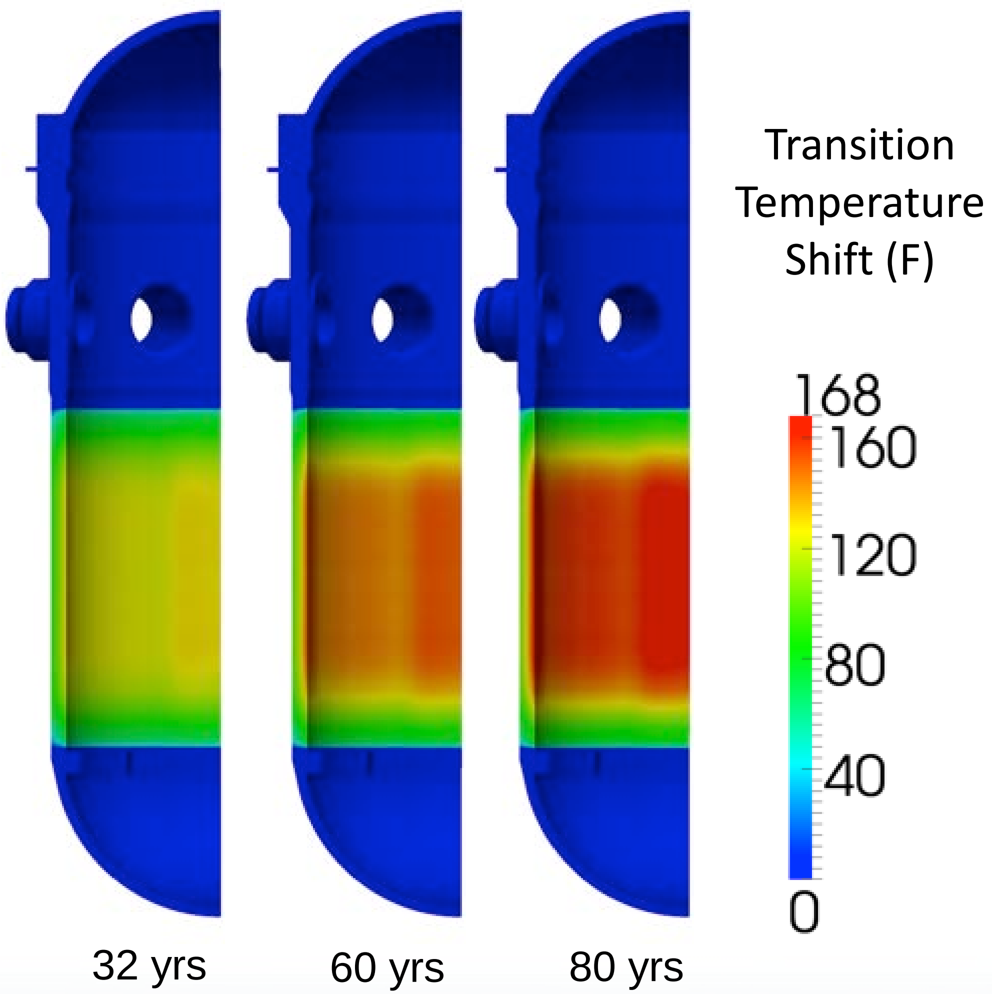


Atom maps of Ni, Mn, Si and Cu atom locations for a Cu free steels

[1] P. Wells, G. Odette, N. Cunningham, T. Milot, Y. Wu, T. Yamamoto et al. Ni-Si-Mn dominated late blooming phases in RPV steels at high fluence and flux. TMS 2013, San Antonio, TX

A new modeling tool is being developed to predict RPV degradation

- EONY model is used in a 3-D model of an RPV, Grizzly – to calculate change in temperature transition shift, over time and location.
- Application beyond 40 years is an extrapolation of experimental data. It will be updated for extended service with new mechanisms and data.
- It will incorporate weldments, heat affected zones, spatial variations in chemistry, and vessel cladding.
- Collaboration between research tasks within LWRS.



INL/ORNL/UT collaboration

The UCSB ATR-2 Irradiation Experiment Has Been Completed in the INL Advanced Test Reactor - Cooperation with LWRSP

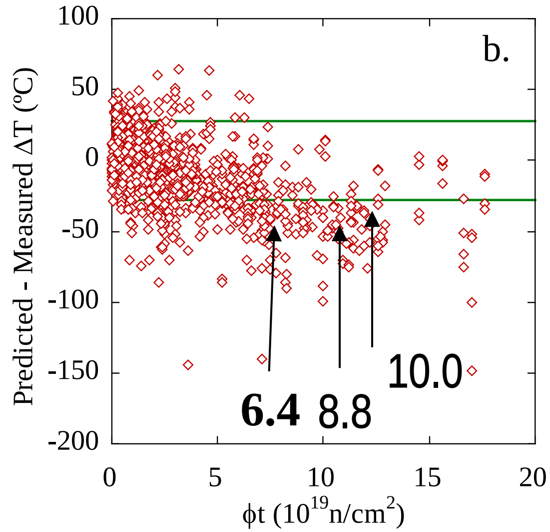
- **Embrittlement of RPV steels up to 80 years ($\phi t \approx 10^{20}$ n/cm^2) is a critical issue**

- **Current RPV embrittlement models under-predict transition temperature shift (TTS) data from <u>highly accelerated irradiations</u> – but this may be a high ϕ artifact**

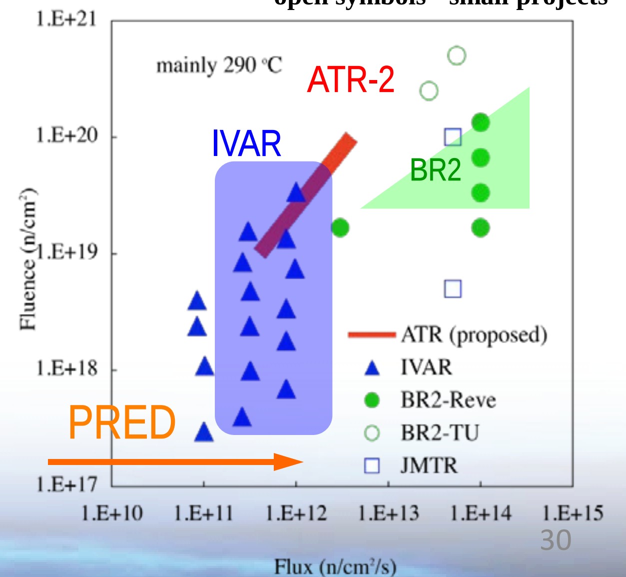
- Four Temperature zones: 250, 270, 290 and 310°C.

- **Total of ≈** 180 RPV steel alloys (~ 1600 specimens) including IVAR program (CM, L-series), split-melt alloys, and <u>nine commercial surveillance materials.</u>

- **Irradiation started June 2011, Peak/Average fluences = 1.3/1.0×10^{20} n/cm^2 .**



open symbols - small projects



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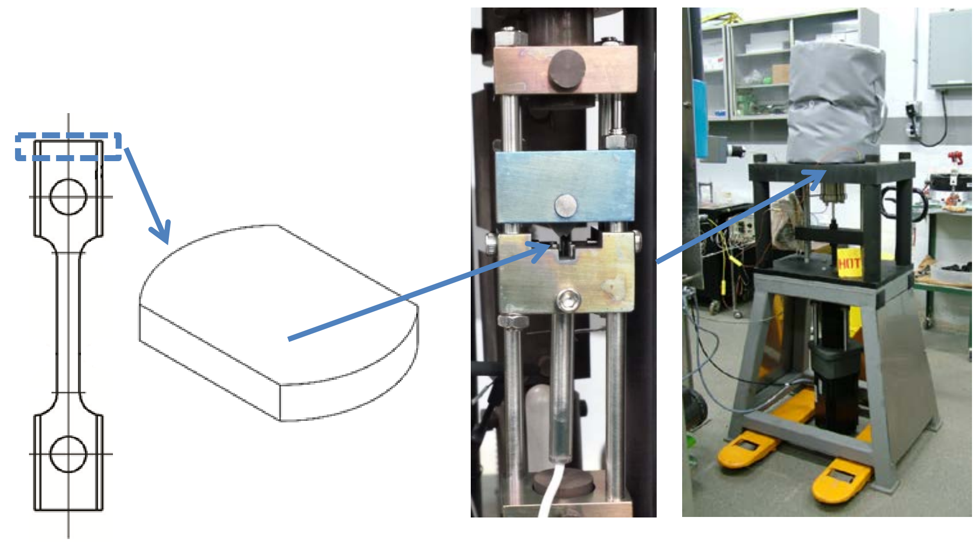
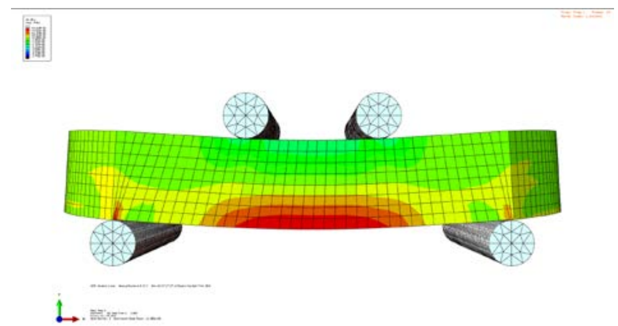
Irradiation Assisted Stress Corrosion Cracking

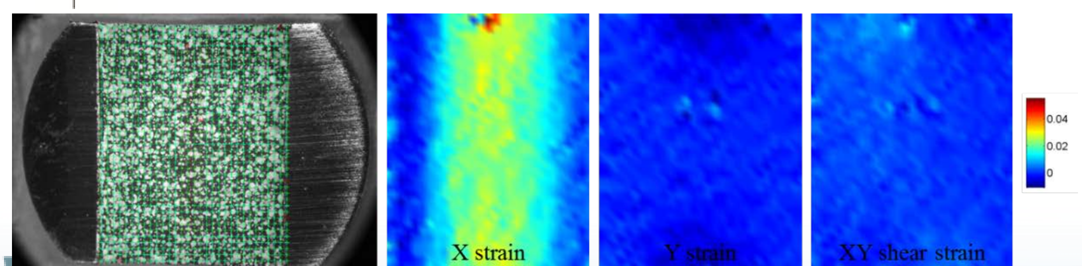
4-Point Bend Testing is being used to study localized deformation and IASCC

- **Objective: determine the relationship between localized deformation and crack initiation**
- 4 point bend tests performed in simulated LWR environment
  - Samples created from tensile bar ends
- Experiment and FEA simulation show good agreement in macroscopic strain measurement

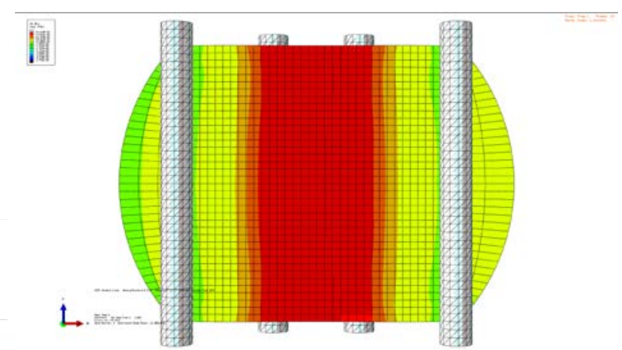




Side view of FEA simulation



DIC strain maps



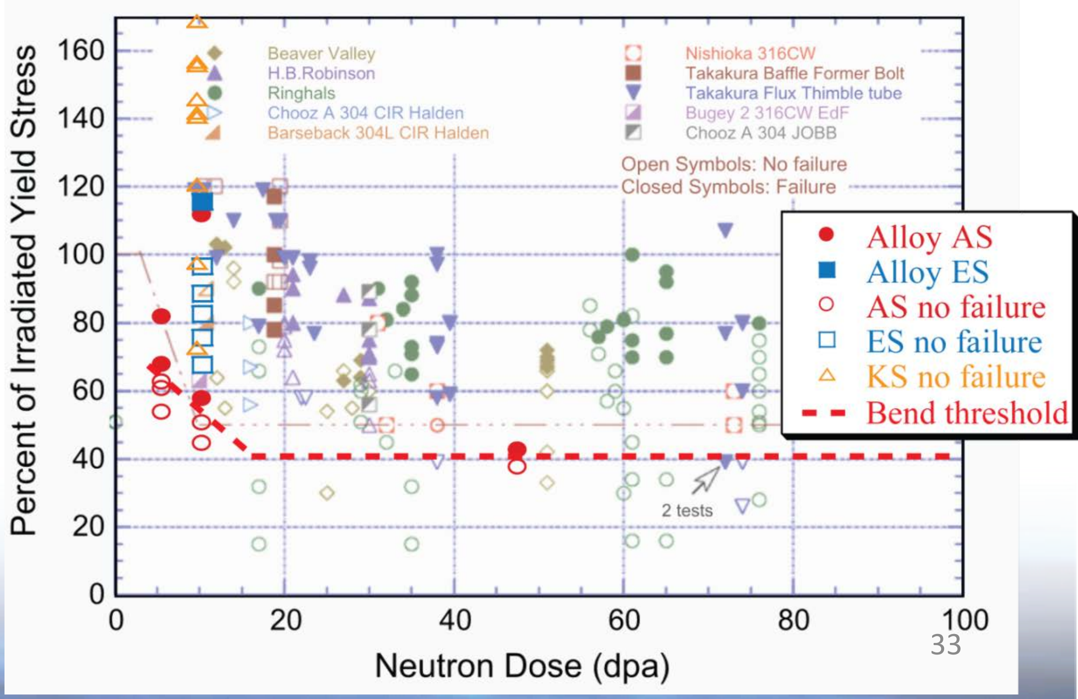
Bottom view of FEA simulation

Four Point Bend Results Reveal A Stress Threshold for IASCC Initiation that Decreases with Dose

- Crack initiation in CP 304L occurred before the yield stress at 5.5, 10.2, and 47.5 dpa due to the high susceptibility of the alloy
  - HP 304L alloys were much less susceptible, and exhibited significantly more deformation (dislocation channel formation) prior to crack initiation
- Stress threshold agrees with results from constant load test results on irradiated materials, despite differences in environment, material, and test type (see table)
  - Indicates likelihood of similar crack initiation mechanism

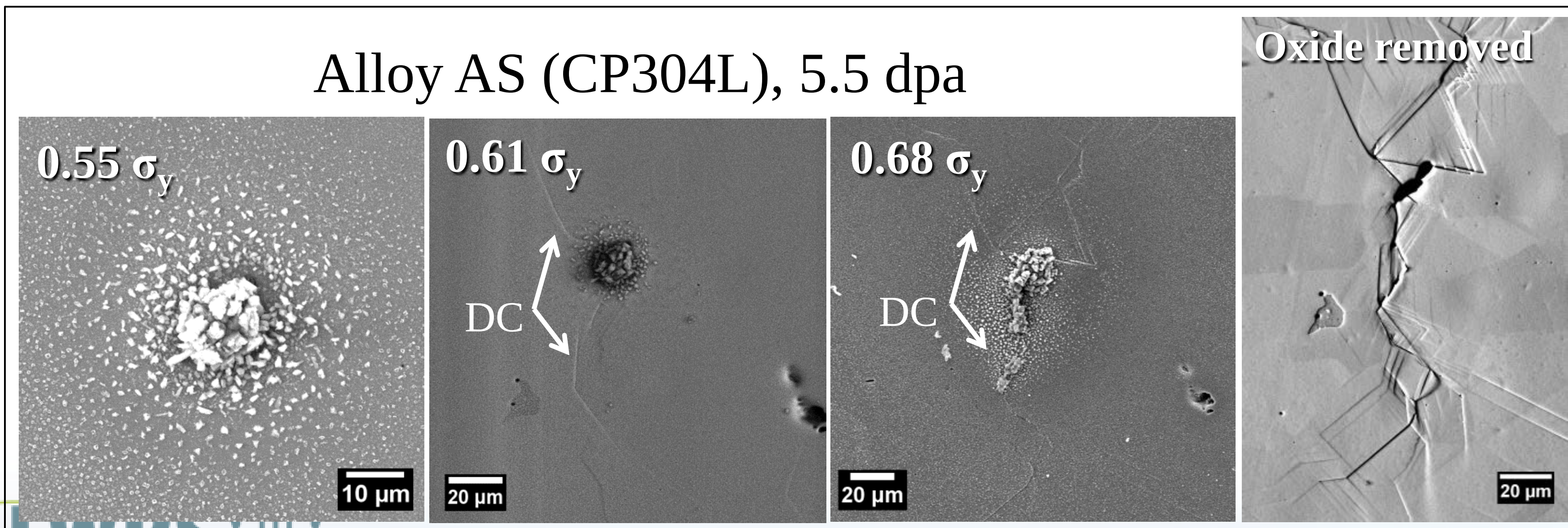
4-Point Bend	Constant Load
288°C NWC	320°C PW
Solution Annealed microstructure	Cold-worked microstructure
Dynamic, slow strain rate	Static, constant load

Chopra, O.K. *Journal of Nuclear Materials*. **409** (2011)



Large MnS Inclusions and Dislocation Channeling are the Main Factors Determining the Degree of IASCC Susceptibility of CP 304L

- IG cracks emanated from surface inclusions in 8 of 9 crack initiation sites; the only negative case having an inclusion within ~10 μm (therefore possibly interacting sub-surface, not visible during SEM examination)
- Dislocation channel interaction with these cracked inclusions was also observed at all crack sites



Modeling and Simulation

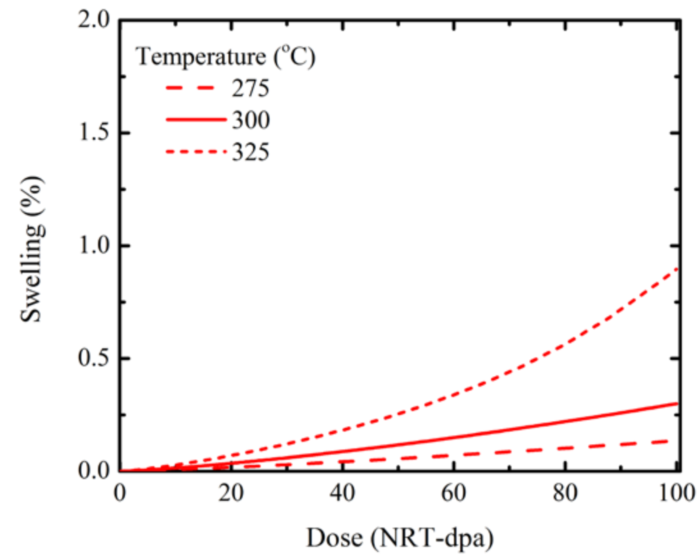
New improvements have been made for the swelling model

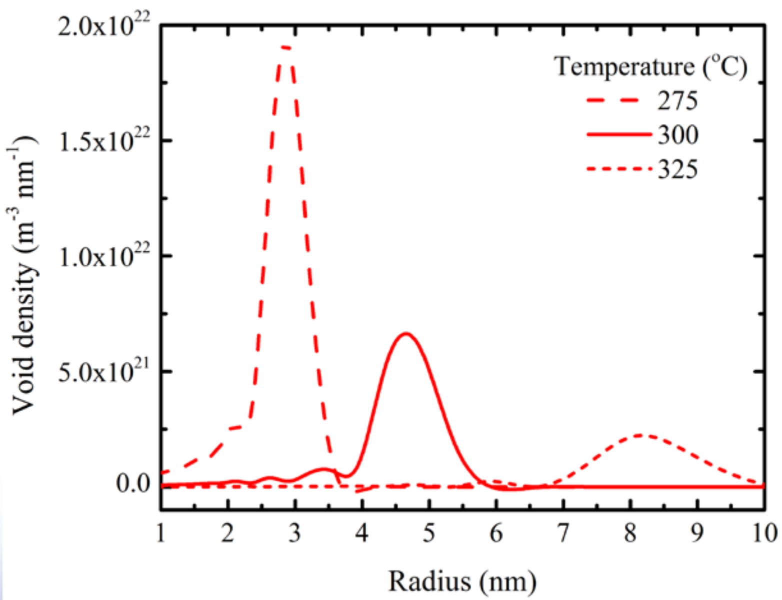
- Implement an improved grouping scheme to discretize defect cluster populations to provide greater fidelity in describing nucleation and growth of size distributions
  - reduce millions of equations to tens or hundreds of equations
  - provide an explicit bubble nucleation component based on a cluster dynamics description of He-vacancy clustering
  - improve previous description of dislocation loop evolution
- Revise reaction-diffusion components of the model to account for additional diffusion mechanisms and relevant sink strengths for extended defects, i.e. mixed one- and three-dimensional diffusion of small interstitial clusters
- Modification of the primary radiation damage source terms in terms of displacements per atom (dpa) and helium production to account for the neutron energy spectrum

LWRS
Light Water Reactor Sustainability

Typical Results: Influence of Temperature on Swelling of Components

- Swelling and microstructural predictions have been obtained using the mean-field microstructural model.
- Typical swelling results to a total dose of 100 dpa are shown in Figure 1 for temperatures of 275, 300, and 325°C. The corresponding cavity size distributions are shown in Figure 2 .





Light Water Reactor Sustainability

Low temperature behavior remains problematic in swelling model

- High vacancy supersaturations (C_v / C_v^e) lead to easy nucleation at low temperatures (recall small critical size)
- Means high potential for swelling below ~350°C
- Bullough and co-workers (1975) proposed that in-cascade formation of small vacancy clusters could effectively store vacancies at low-temperatures and provide the mechanism for the assumed low-temperature swelling cut-off
- Current model includes such a transient sink

Light Water Reactor Sustainability

Precipitation Modeling in Irradiated Stainless Steels

Project goals: Develop model of damage evolution in 316 stainless steels that includes evolving vacancy concentrations and dislocations and apply it to predict radiation enhanced precipitation to support embrittlement prediction.

Approach

Phase 1: Modeling the defect evolution to get appropriate radiation enhanced diffusion coefficients and heterogeneous dislocation loop nucleation sites

- Developing a Cluster Dynamics code (CD-defect) based on previous work [1-4] and fitting to experiments to estimate sink (loop, void) and point defect evolution under irradiation

Phase 2: Modeling the precipitation kinetics

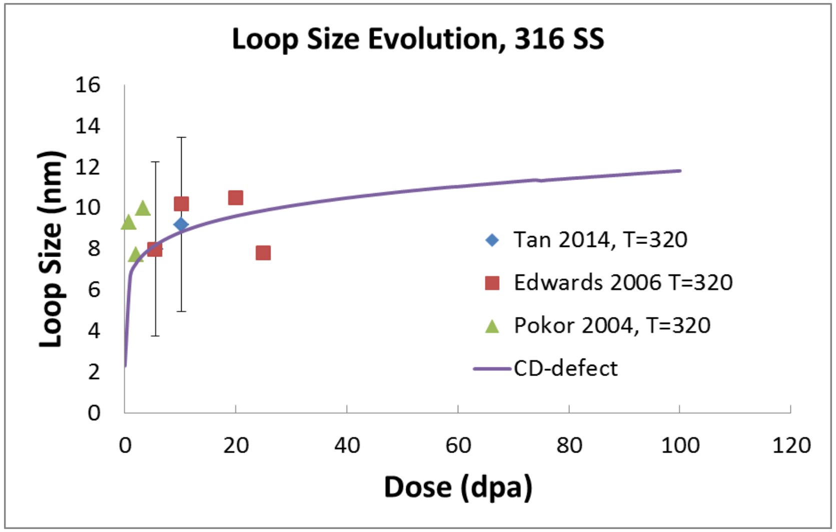
- Predicting precipitation kinetics by combining the CD-defect predictions with classical nucleation theory precipitation model based on OCTANT [5] CALPHAD and MatCalc [6] databases/tools.

[1] A. Duparc et al. JNM, 2002.
[2] C. Pokor et al. JNM, 2004.
[3] A. Gokhman , F. Bergner. 2010.
[4] J. Gan et al. JNM, 2001.
[5] Ying Y, Busby J. JNM, 2014.
[6] Kozeschnik E, Buchmayr B, 1999.

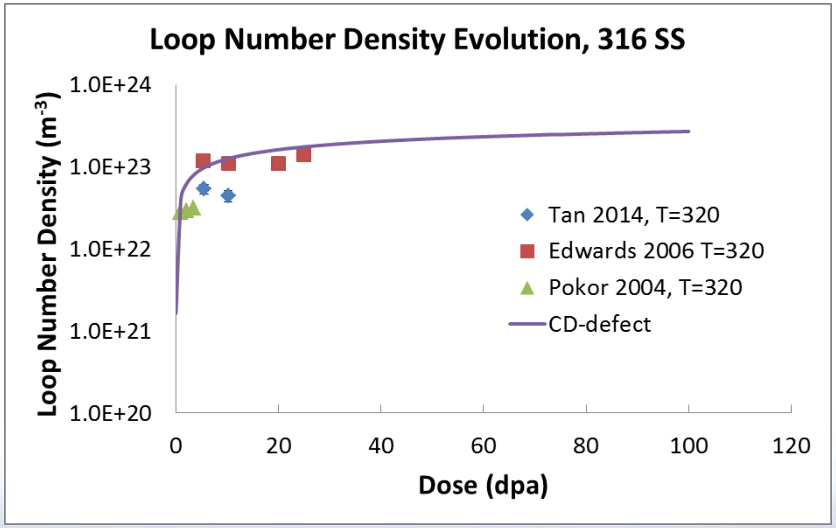
Frank loops growth is important to capture

- Phase 1: CD-defect tool
  - Material parameters are selected from literature [1,2].
  - In-cascade clustering is modeled based on molecular dynamics simulation [3].
  - The CD-defect code successfully models the experimental loop size (top fig.) and density (bottom fig.).

Successful model for predicting dislocation evolution and vacancy concentration to input into precipitation model



Loop size evolution of 316SS under irradiation



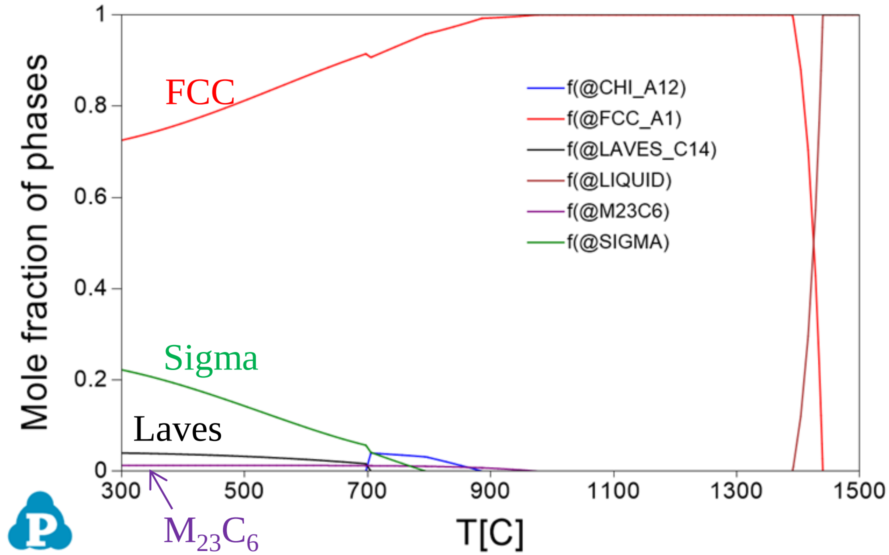
Loop number density evolution of 316 SS under irradiation

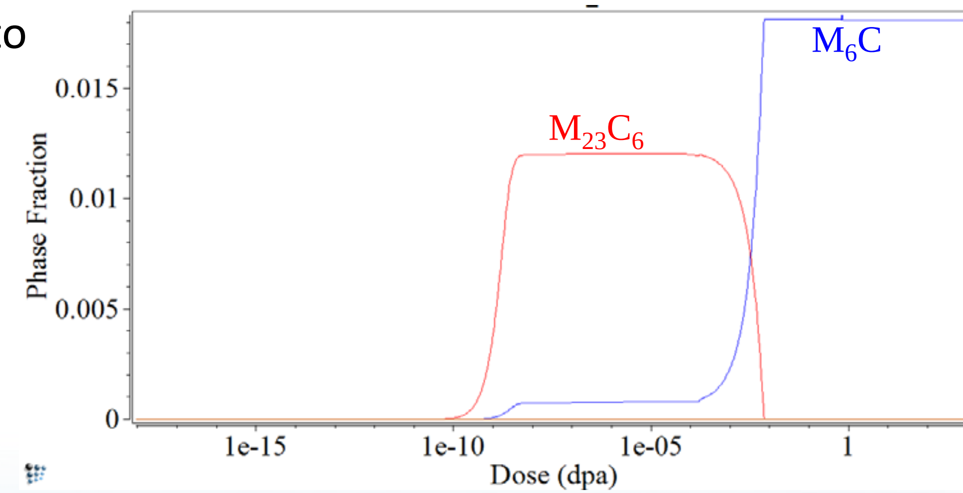
1. C. Pokor et al., JNM, 2004
2. N. Ghoniem et al., phys. stat. sol., 1979
3. W. Phythian et al., JNM, 1995

Stable and metastable phases can be predicted

- Phase 2: Modeling precipitation kinetics
  - $M_{23}C_6$, Laves, and Sigma are stable phases at ~300 °C (top fig.) but Laves and Sigma are kinetically inhibited phases under LWR condition.
  - $M_{23}C_6$ is the fastest emerging phase but, it dissolve at expense of M_6C later (bottom fig.). Dominant M_6C reported in experimental observations [1-4].
  - G-phase and γ’ are radiation induced precipitation phases and γ’ is predicted to be the dominant phase in segregated regions under LWR condition.

Precipitation model results qualitatively match experiments – further validation in progress





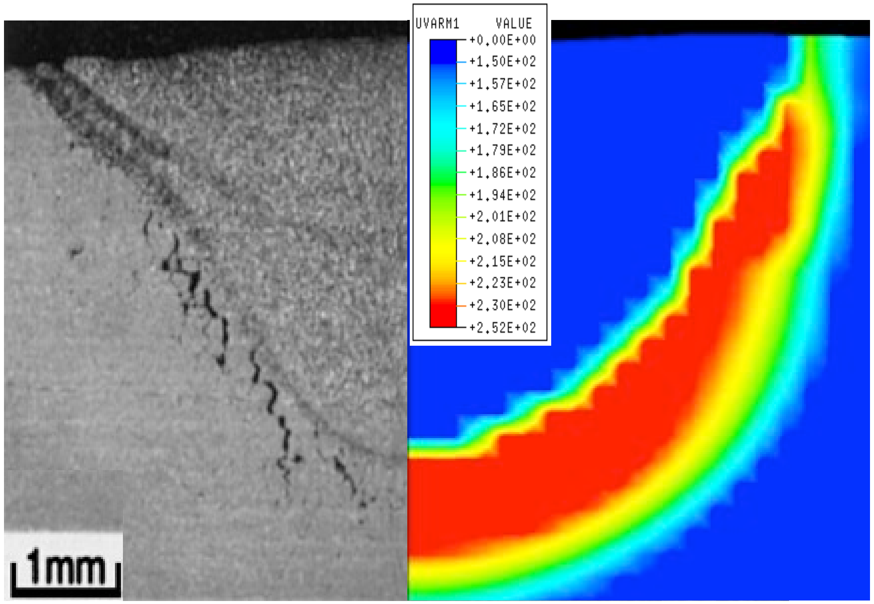
Precipitation in 316 SS with matrix composition.

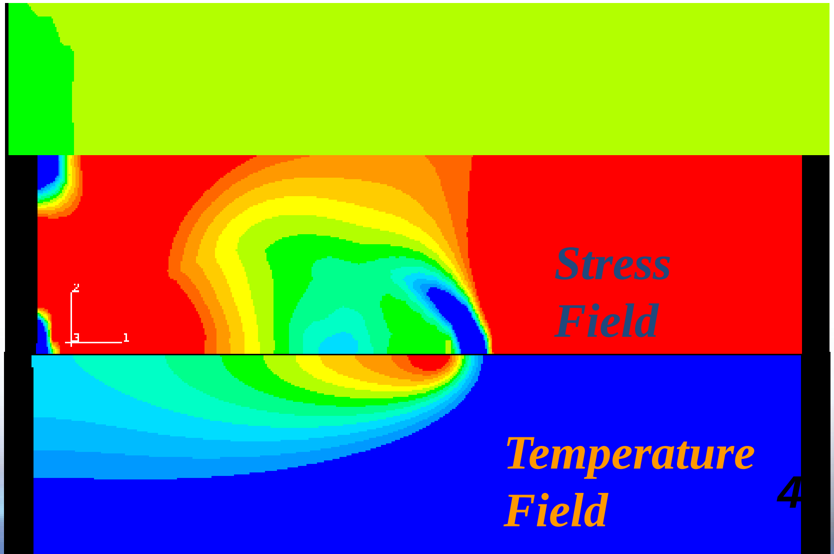
1. A. Renault-Laborne et al. ASTM, 2014. 2. N. Hashimoto et al. JNM, 2000.
3. Maziasz, P., JNM, 1982. 4. Maziasz, P., Scripta Metallurgica 1979.

Mitigation techniques

Advanced welding R&D may provide solutions to long-standing areas of concern

- Residual stress-modeling provides insights into long-term performance and cracking resistance
- Current research in advanced weldments is jointly funded by DOE and EPRI
  - Survey of present art of hybrid welding processes
  - Advanced computational model for hybrid welding processes
  - Hybrid laser weld processing model to optimize the weldability of irradiated materials
  - Experimental methodology for direct measurement of transient high-temperature stress history during welding
- Technology is being developed with the direct expectation of transfer to industry in the near term

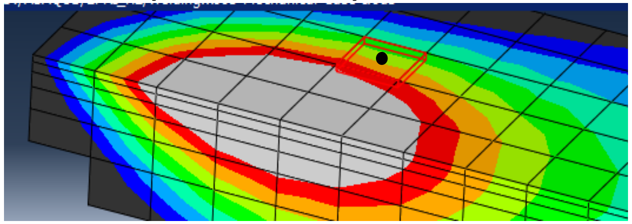


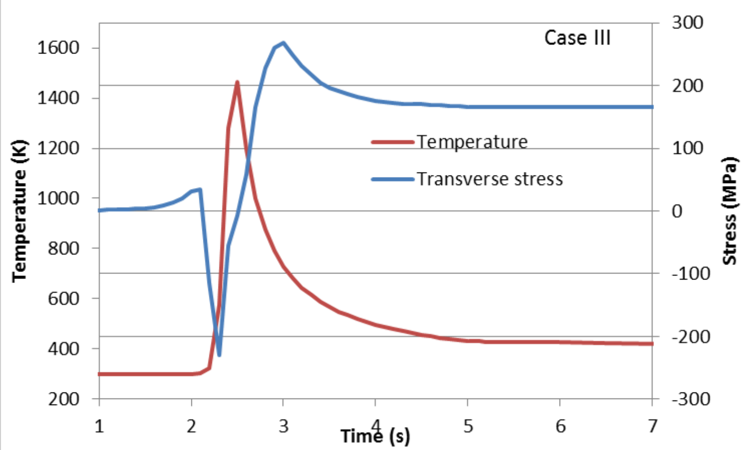


FE Thermal and Mechanical Modeling and Experimental Validation

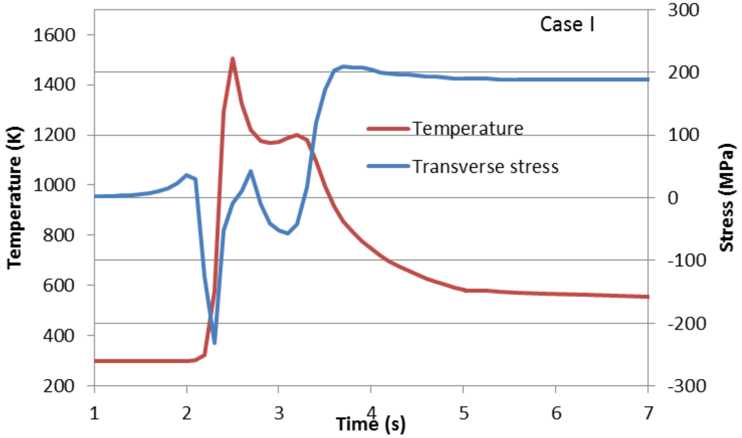
- FE models were developed to proactively manage the thermal input and control the stress distribution during the welding process.
  - With proactive thermal management, the tensile stress adjacent to the weld pool at temperatures above 1000K can be effectively reduced.

Modeling: adjacent to weld pool on top surface





Conventional method



Proactive thermal management

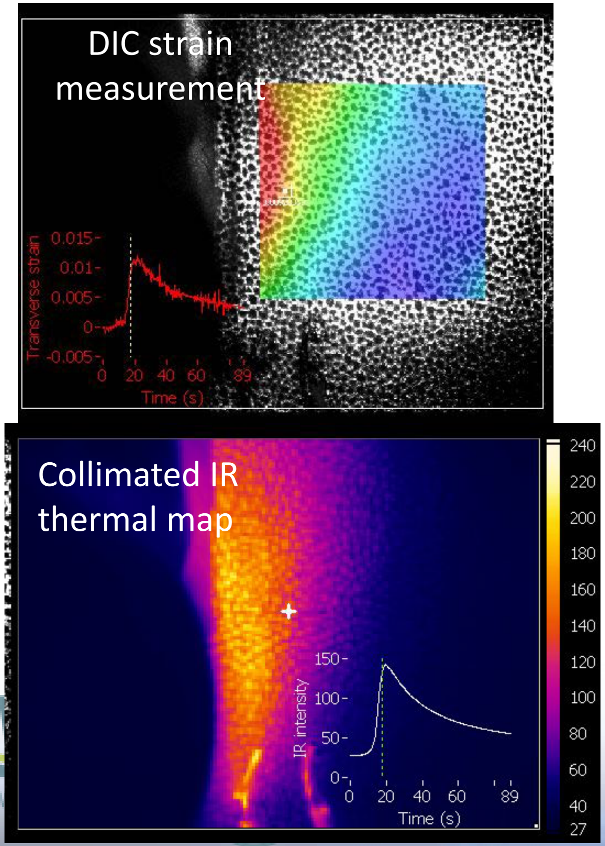
Light Water Reactor Sustainability

Recent Publication: Chen, J, Yu, X, Miller, RG, and Feng, Z. (2015) *In situ strain and temperature measurement and modeling during arc welding.* Sci Tech Welding Joining, **20**(3):181-188..

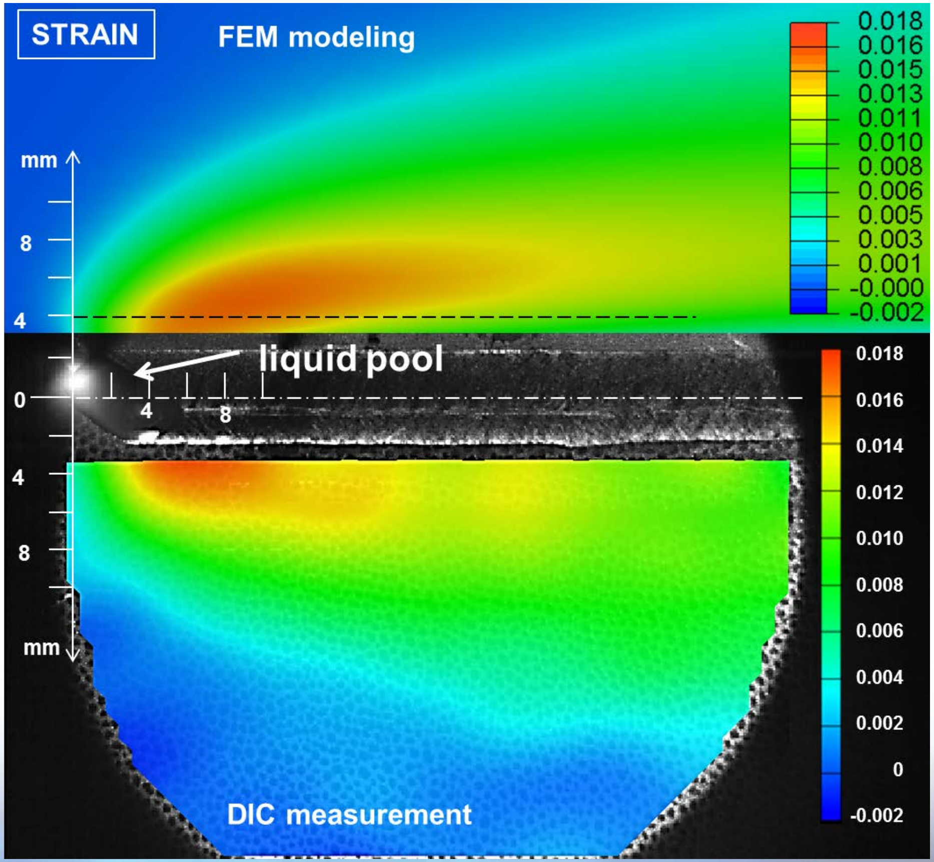
FE Thermal and Mechanical Modeling and Experimental Validation (cont'd)

- An in-situ DIC strain and IR thermography measurement technique was developed to validate FE modeling.

Experiments

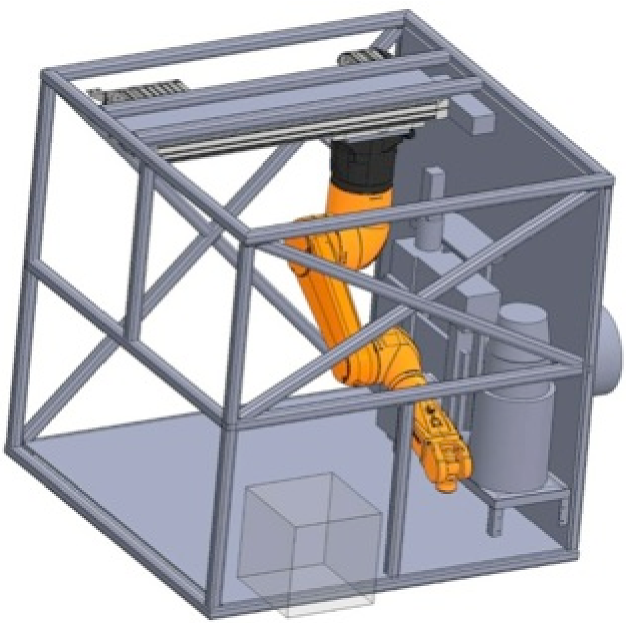


Experiment vs. Modeling

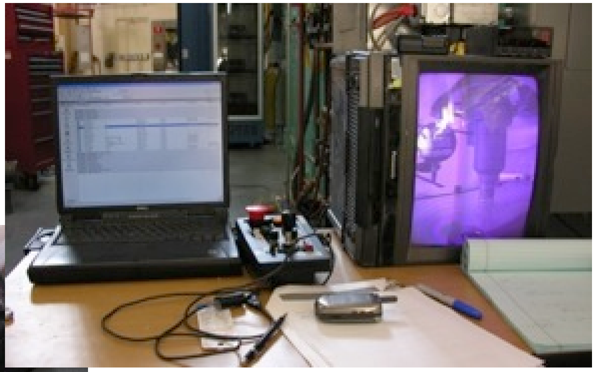


Design and Construction of A Dedicated Welding Hot Cell:

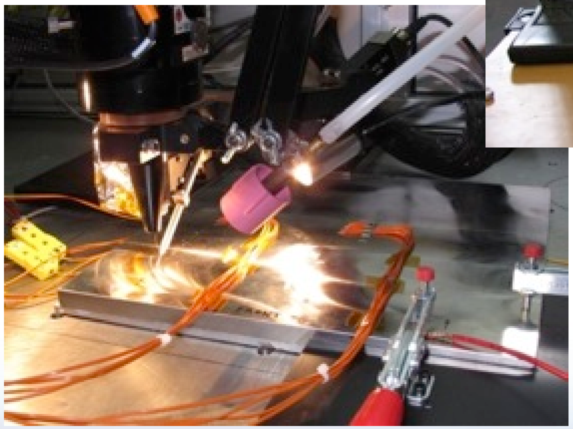
- First of its kind in the US. Part of an "one-stop" facility for R&D on irradiated materials to support DOE NE programs and industry's needs.
- Cost-shared with EPRI
- Switchable between different welding processes: laser welding, arc welding, and friction stir welding systems. Both LW and FSW can be remotely operated to reduce contamination issues of welding equipment
- In-situ temperature and stress measurement capability through remote optical system and unique measurement techniques
- System design has been completed. Individual hardware are being procured and tested



Exposed view of concept design of welding hot cell with robotic manipulators and friction stir welding system

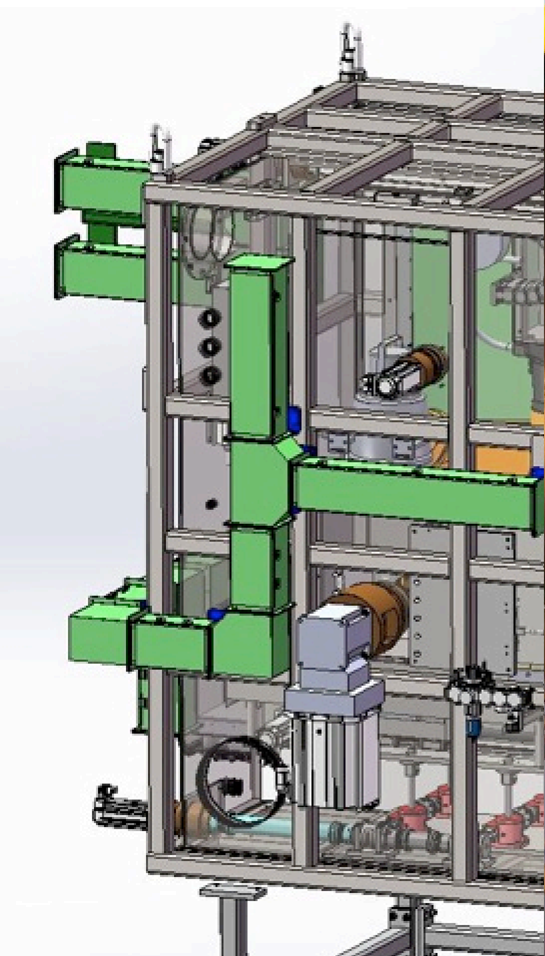


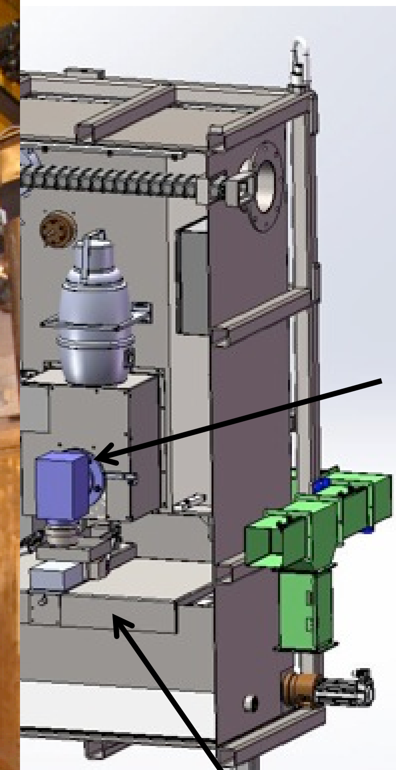
Remotely operated FSW system to be integrated in the hot cell



Laser welding system under testing and to be integrated in the hot cell

Welding Hot Cell Cubicle Equipment Integration





The LWRS Program Website Provides a Range of Program Information (www.inl.gov/lwrs)

Light Water Reactor Sustainability

Reliability, availability, productivity, component aging, safety and security of existing nuclear power plants.

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What's new

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U.S. DEPARTMENT OF ENERGY
Nuclear Energy

LIGHT WATER REACTOR SUSTAINABILITY PROGRAM: INTRODUCTION

The Light Water Reactor Sustainability (LWRS) Program is the primary programmatic activity that addresses Objective 1 (develop technologies and other solutions that can improve the reliability, sustain the safety, and extend the life of the current reactors) described in the U.S. Department of Energy Office of Nuclear Energy's 2010 Research and Development Roadmap. For the purpose of the LWRS Program, "sustainability" means the prudent use of resources – in this case, our nation's commercial nuclear power plants. Sustainability is defined as the ability to maintain safe and economic operation of the existing fleet of nuclear power plants for a longer-than-initially-licensed lifetime. It has two facets with respect to long-term operations: (1) manage the aging of plant systems, structures, and components so that nuclear power plant lifetimes can be extended and the plants can continue to operate safely, efficiently, and economically; and (2) provide science-based solutions to the industry to implement technology to exceed the performance of the current labor-intensive business model.

Operation of the existing plants to 60 years, extending the operating lifetimes of those plants beyond 60 years and, where practical, making further improvements in their productivity is essential to realizing the Administration's goals of reducing greenhouse gas emissions to 80% below 1990 levels by the year 2050.

The following LWRS Program research and development pathways address Objective 1 of the 2010 Nuclear Energy Roadmap:

- *Materials Aging and Degradation*
- *Advanced Instrumentation, Information, and Control Systems Technologies*
- *Risk-Informed Safety Margin Characterization*
- *Reactor Safety Technologies*