LWRS Program Materials Research
Life Beyond Eighty (LBE)
Stakeholder Engagement Meeting

Materials Research Pathway
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Oak Ridge National Laboratory

Virtual from Oak Ridge, TN

November 5, 2020

US DOE Light Water Reactor Sustainability Program

Materials Research Pathway (MRP) Life Beyond Eighty (LBE) Stakeholder Engagement Meeting

November 5, 2020

Time (EST)	Activity	Lead		Organization
1:00 – 1:10	Welcome	Tom and Frank		ORNL
1:10 – 1:15	LWRS TIO Director and US DOE Federal Manager	Bruce Hallbert Alison Hahn		INL DOE-NE
1:15 – 1:25	Introductions	Tom Rosseel		ORNL
1:25 - 1:40	LWRS MRP Overview and LBE	Tom and Frank		ORNL
1:40– 2:00	Reactor Pressure Vessel Aging at Extended Operation	Mikhail Sokolov		ORNL
2:00 – 2:10	Discussion	All		
2:10 – 2:30	Emerging materials issues of austenitic internals for PWR operation to 80 years and beyond	Frank Garner Maxim Gussev		Radiation Effects Consulting ORNL
2:30 – 2:40	Discussion	All		
2:40 – 2:55	Coordination of EPRI Issue Programs' Research Activities with LWRS Materials Pathway Projects	Mike Burke Emma Wong		EPRI
2:55 – 3:05	Discussion	All		

US DOE Light Water Reactor Sustainability Program

Materials Research Pathway (MRP) Life Beyond Eighty (LBE) Stakeholder Engagement Meeting

November 5, 2020

3:05 – 3:20	Break	All		
3:20 – 3:35	Life Beyond 80: Concrete Aging	Yann Le Pape		ORNL
3:35 – 3:45	Discussion	All		
3:45 – 4:00	Reliable Use of Old Cables in Extended Operation	Leo Fifield		PNNL
4:00 – 4:10	Discussion	All		
4:10 – 4:25	Second / Subsequent EMDA: What, When, and How?	Tom Rosseel		ORNL
4:25 – 4:35	Discussion	All		
4:35 – 5:00	Path Forward	All		
5:00	Adjourn	Tom and Frank		ORNL

LWRS
Program Goal and Objectives

- **Goal**
  - **Enhance the safe, efficient, and economical performance of our nation's nuclear fleet and extend the operating lifetimes of this reliable source of electricity**.
- **Objectives**
  - Enable long-term operation of the existing nuclear power plants
  - Deploy innovative approaches to improve economics and economic competitiveness of LWRs in the near-term and in future energy markets
  - Sustain safety, improve reliability, enhance economics
- **Research and development focus areas**
  - Plant modernization
  - Flexible plant operation and generation
  - Risk-informed systems analysis
  - **Materials research**
  - Physical security

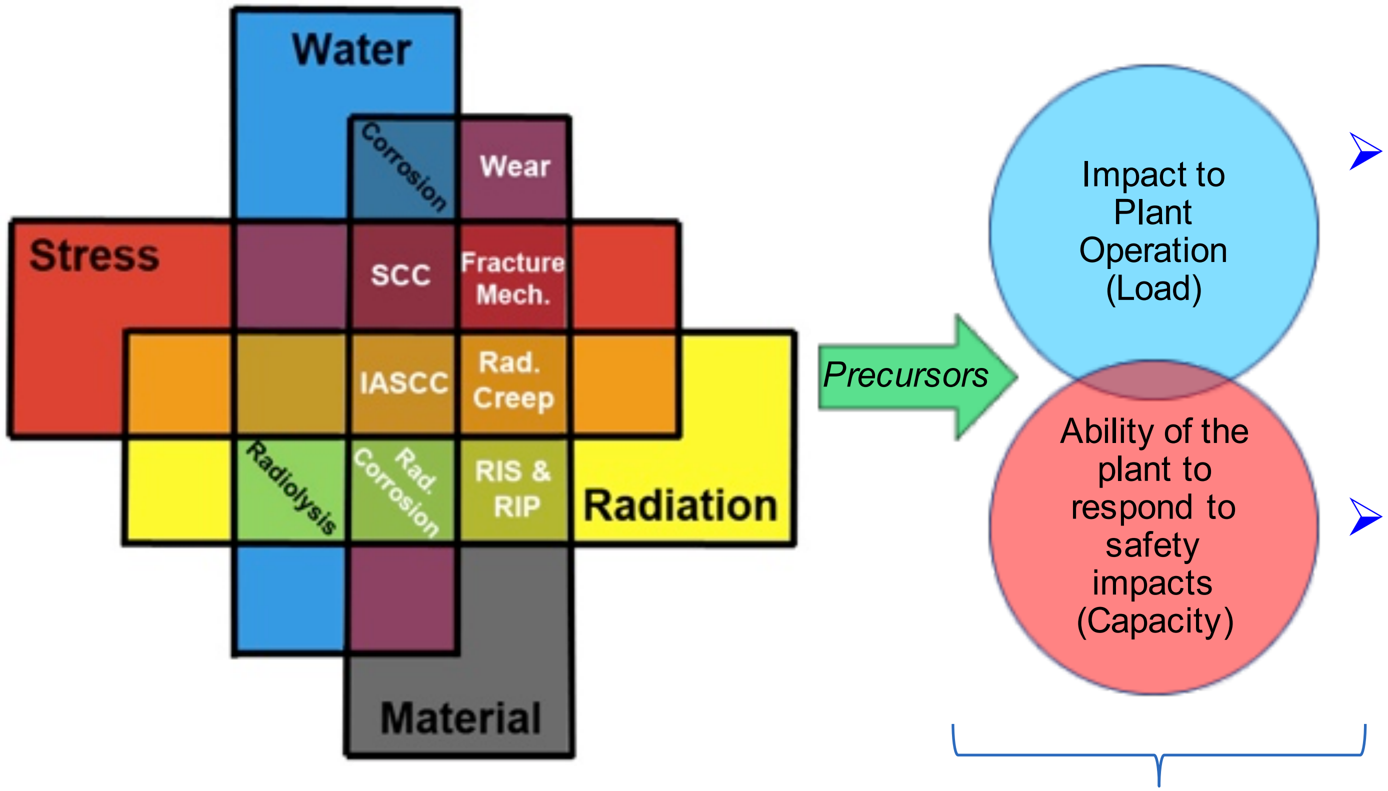
DOE's program for LWR RD&D

Nine Mile Point (Courtesy of Exelon)

LWRS Pathway Goals

Plant Modernization	Enable **plant efficiency improvements** through a strategy for long-term modernization
Flexible Plant Operation & Generation	Enable **diversification and increase revenue** of light water reactors by deploying systems to extract electrical and thermal energy to produce non-electrical products
Risk Informed System Analysis	Develop **significantly improved safety analysis methods and tools** to optimize the safety, reliability, and economics of plants
Physical Security	Develop and provide **technologies and the technical bases to optimize physical security postures** to maintain protection and improve efficiencies
Materials Research	Develop the **scientific basis to understand and predict long-term behavior of materials** including detecting and **characterizing aging mechanisms and components** essential to **safe and economically sustainable NPP operations**

Materials Research Pathway: Goals and Objectives



Potential impacts on economics, reliability and safety

- To **develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and**
- To **provide data and methods to assess performance of systems, structures, and components essential to safe and economically sustainable nuclear power plant operations.**

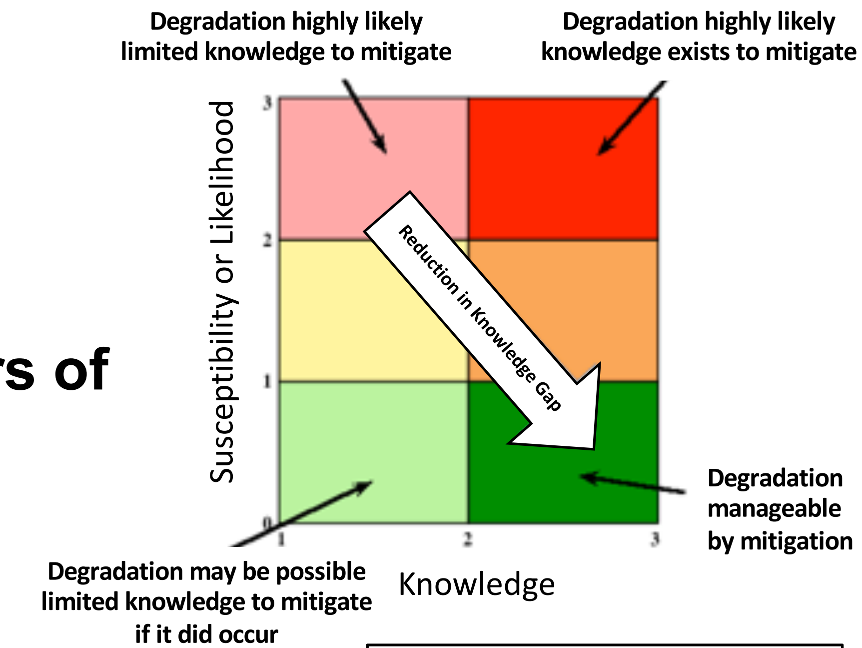
Materials in Extreme Environments

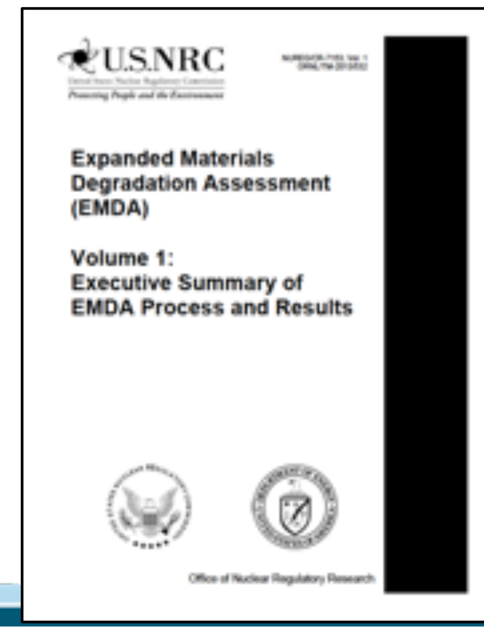
Research Needs Assessment

Expanded Materials Degradation Assessment (EMDA):

NUREG / CR-7153 (joint DOE / NRC effort) 2011 - 2013

- **Expanded scope of the Proactive Materials Degradation Assessment (PMDA, NUREG / CR-6923) from internals to identify degradation effects and scenarios beyond 60 years of operation.**
- **Expert panel from the nuclear community (industry, EPRI, national labs, universities, international, & NRC)**
- **PMDA findings used as inputs to develop Generic Aging Lessons Learned (GALL).**
- **Addressed gaps but does not rank in terms of priority.**
- **EMDA: RPV, Internals, Concrete, and Cables**




U.S.NRC

Expanded Materials Degradation Assessment (EMDA)

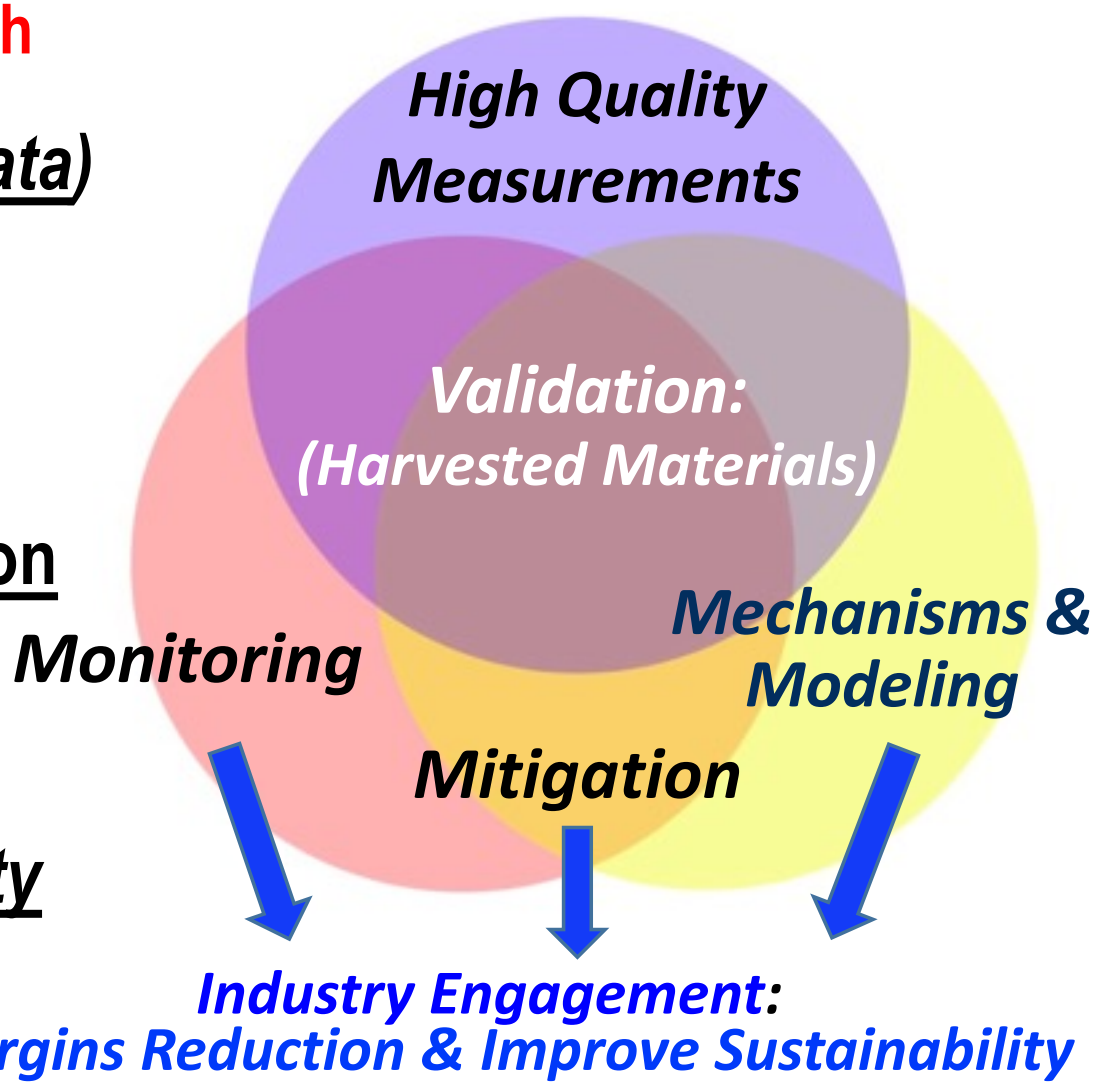
Volume 1: Executive Summary of EMDA Process and Results

Office of Nuclear Regulatory Research

Addressing aging management knowledge gaps requires a multifaceted research approach

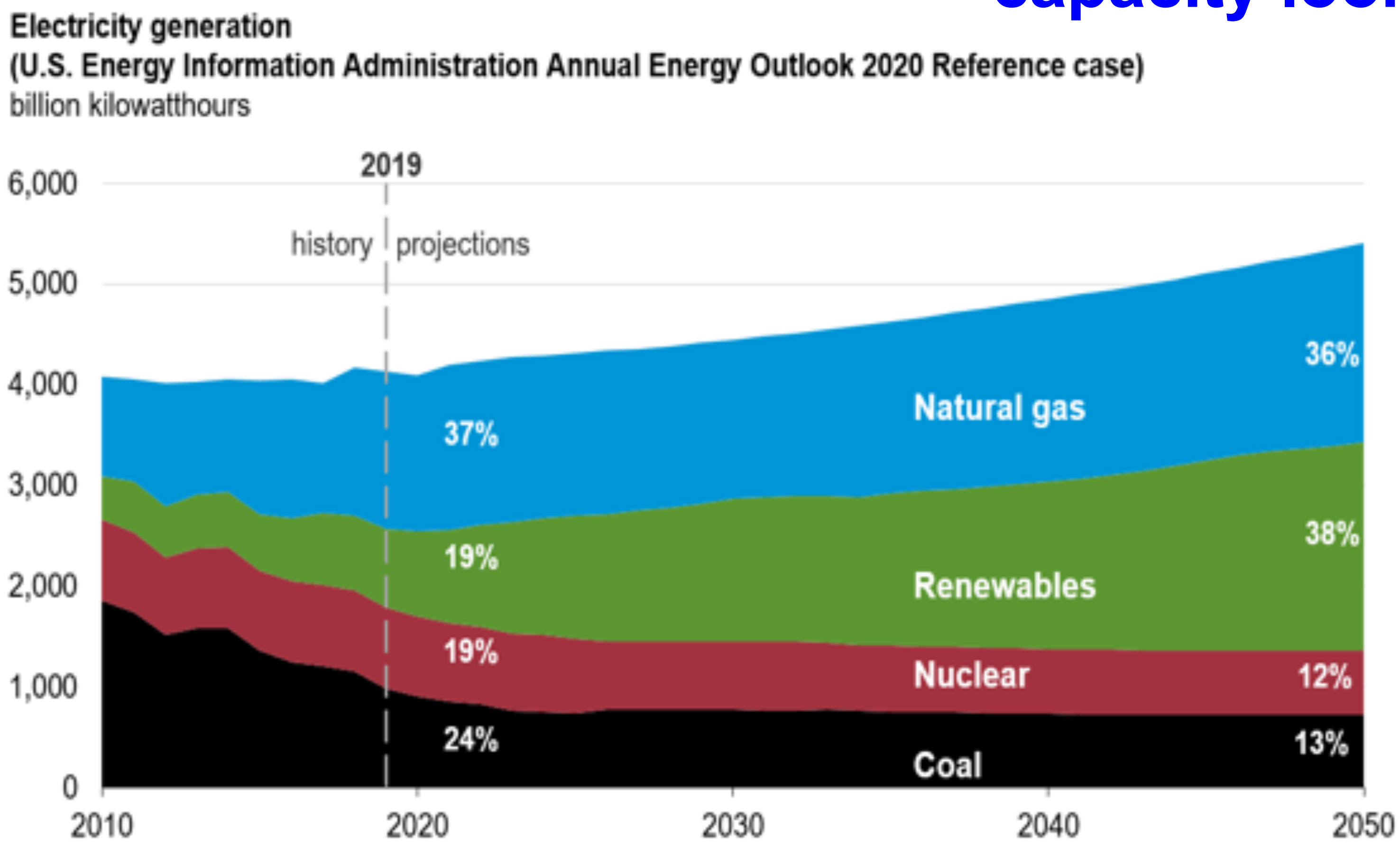
Guided by sound nuclear materials research approach

- ***Measurements of degradation*** ***(high quality data)***
  - ***Structure and properties of materials under stress***
- ***Mechanisms of degradation (scientific understanding)***
- ***Modeling and simulation to predict degradation***
- ***Monitoring degradation (non-destructive examination)***
- ***Mitigation strategies for economic productivity***

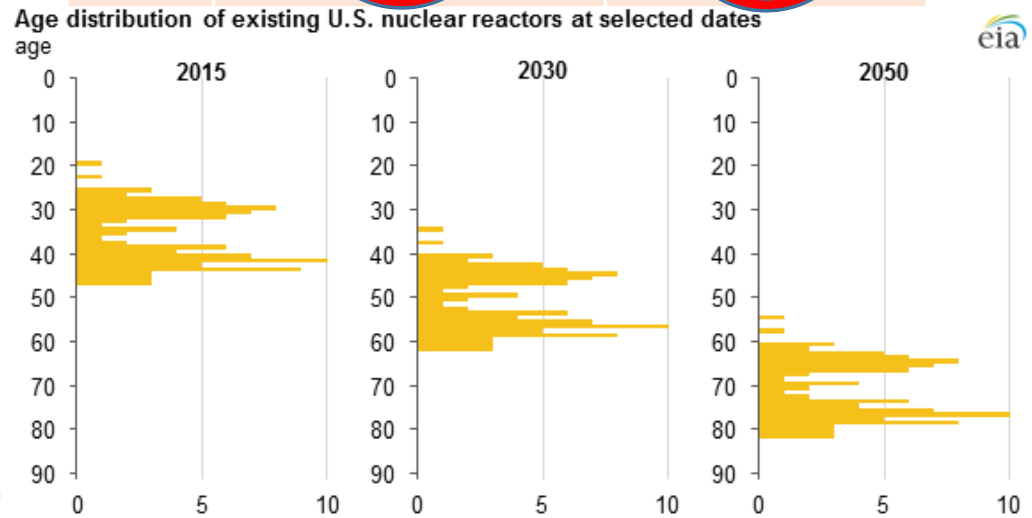


Each Materials Research Pathway task applies one or more of these methods to address aging management and life-time extension knowledge gaps

What will US electrical power generation and capacity look like in 2050?



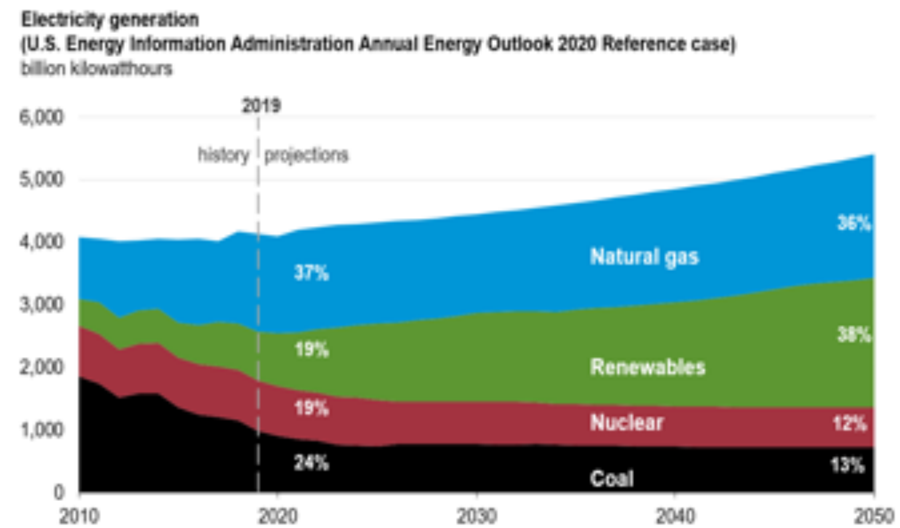
	Nuclear electricity generation (BkWh)	Nuclear Capacity (GW)
2019	807	98.1
2050	642	78.5



- Based on the US Energy Information Administration (EIA) predictions, **by 2050 nuclear capacity and electricity generation (including new builds) will decrease to ~ 80% of 2019 levels**
- Based on the age distribution of existing US nuclear reactors, **by 2050, 50% of US nuclear fleet will be within 10 yrs. of 80 years of operation** and, therefore, **without a LBE plan, the US could lose 50% of its nuclear capacity due to closures and limited new builds, resulting in ~30 GW capacity shortage in 2060**

What about SMRs, Advanced Reactors, and Renewables?

- What is the current outlook in the US for advanced reactors with passive safety systems? (**Can costs and time to build be reduced?**)
- Do we know how many SMRs and other advanced reactor concepts will be operational? (**How long will it take to assess success?**)
- What is the path forward to **increase the capacity of advanced reactors / SMRs** by 2050?
- Is the **electrical capacity of renewables** under or over predicted? (**30-year estimates are questionable**)
- Can we predict the size of a carbon tax? (not likely)



What are the LBE known unknowns?

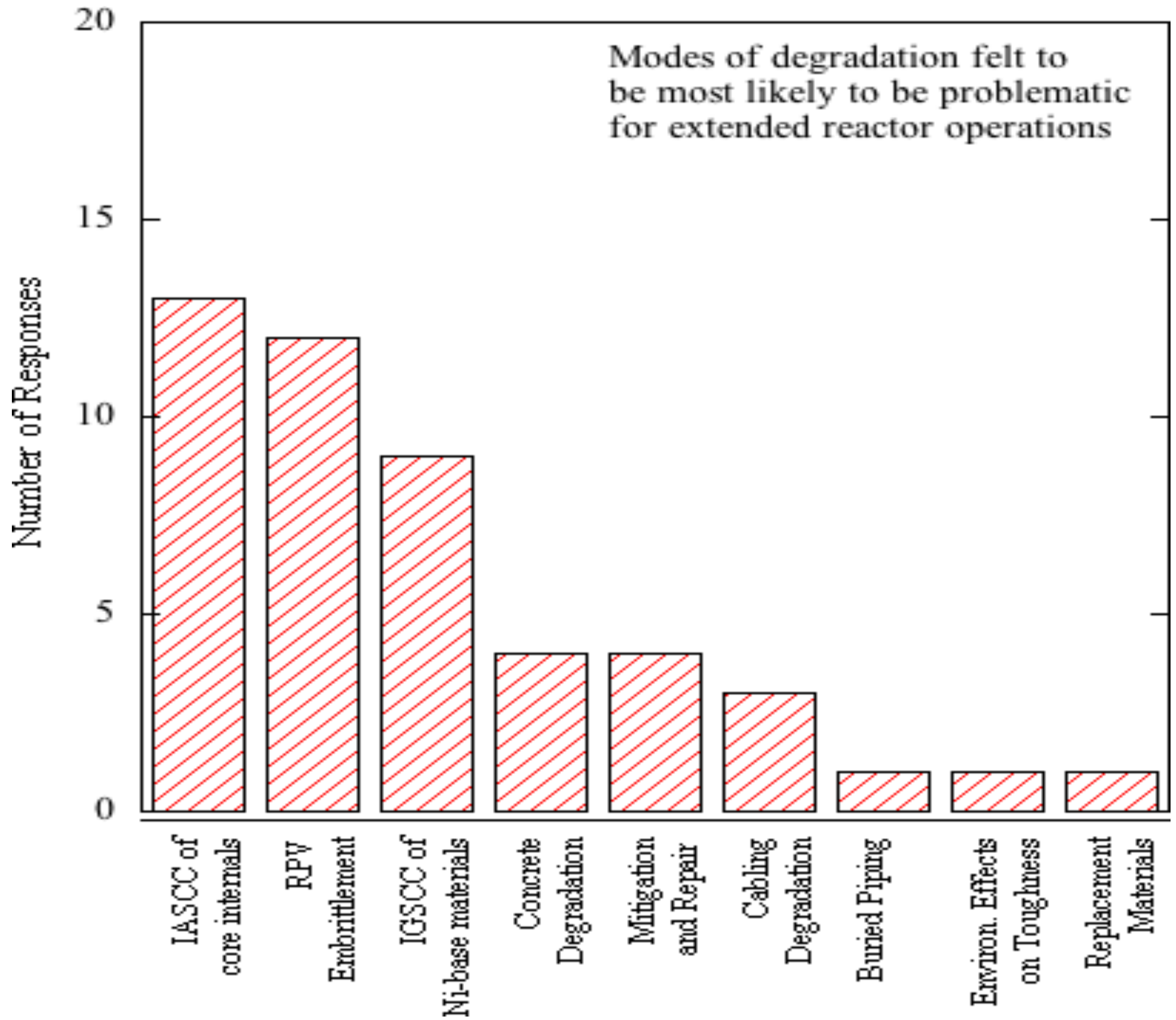
- Degradation modes that **are already occurring and will grow more severe** during extended lifetimes
- Degradation modes at LBE for which **there is limited or no mechanistic understanding** and for which **long-term research is needed**
- Degradation modes for which **follow-on work can complement ongoing national or international research**
- Areas for which **technical progress can be made in the near term.**

What are the LBE unknown unknowns?

- **Degradation modes** for which there is **little or no supporting data** and that **may be problematic for** extended lifetimes
- **Unidentified degradation modes that may already be occurring** and **may grow more severe during extended lifetimes**
- **Improved / advances** in **NDE technologies and methods:**
- Will **sensors / methods better identify degradation?**
- Will real time **monitoring be feasible?**
- Will **advances in mitigation methods be useful?**
- Will advanced **replacement alloys be cost effective?**
- Will **weld repair methods be successful in the field?**
- **Other issues / Suggestions / Discussions / Path Forward**

Identification and Prioritization of Research Activities

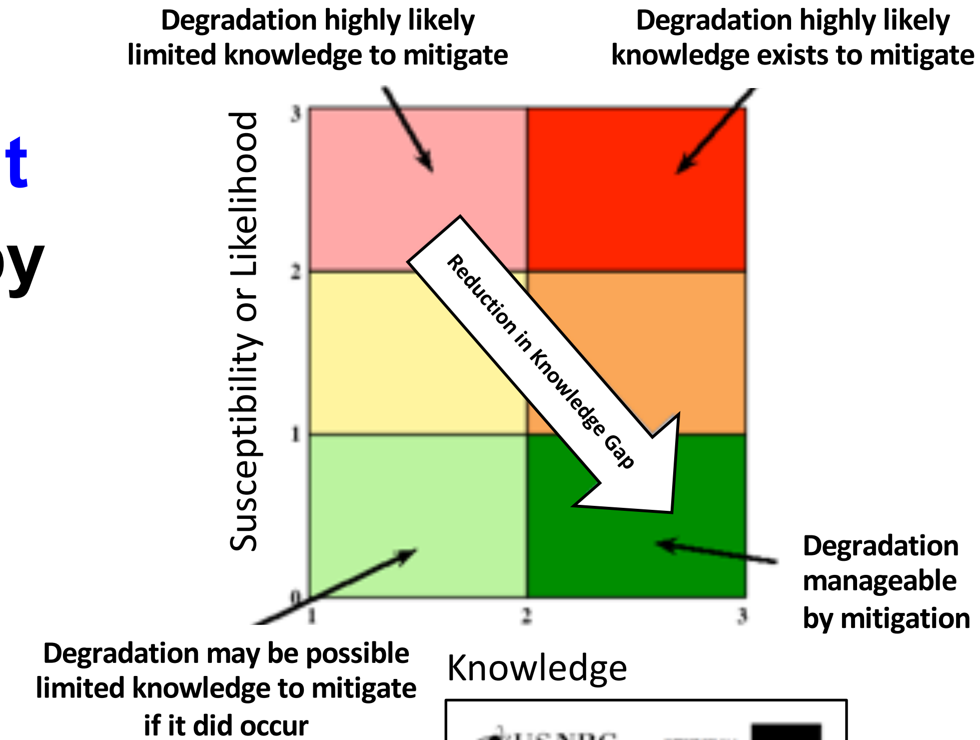
- In **2008**, three national laboratories, two universities, two nuclear reactor vendors, a nuclear power plant utility, and nine key experts from EPRI participated in research discussions **to address extended NPP operations** at EPRI Charlotte.
- Goal was to **identify, formulate, and prioritize the competing needs** in a collaborative manner with, DOE NLs, industrial and regulatory partners.
- Teams identified an **initial list of the most pressing research tasks.**

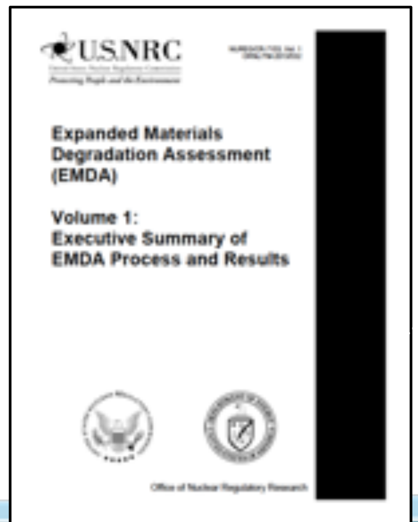


Summary of modes of degradation that are the most likely to be problematic for long- term operation of NPPs.

Establishing LBE Research Needs Based on an Expert Panel Consensus

- In FY 22, initiate a Subsequent or Second Expanded Materials Degradation Assessment (SEMDA) and publish a gap analysis report by 2024:
- Expert panel from the nuclear community (industry, EPRI, national labs, universities, international, & NRC)
- Address gaps but not rank in terms of priority.
- SEMDA: RPV, Internals, Concrete, Cables, Mitigation, NDE / On-line Monitoring, Harvesting





U.S.NRC

Expanded Materials Degradation Assessment (EMDA)

Volume 1: Executive Summary of EMDA Process and Results

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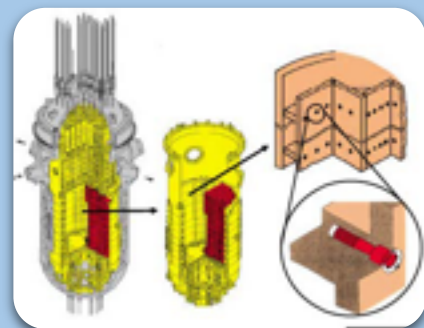
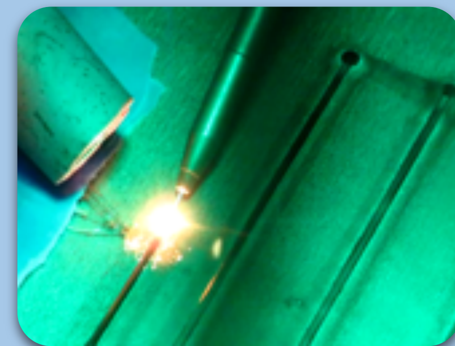
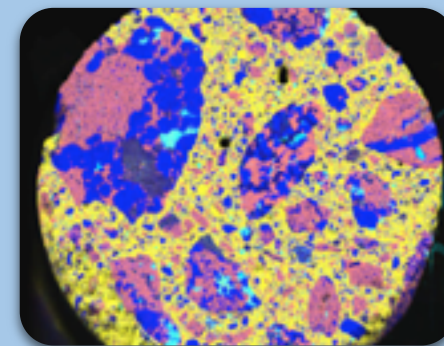
Materials Research Pathway: Results, Outcomes, Collaborations







Reactor Pressure Vessel

Core Internals and Piping

Mitigation Technologies

Concrete Degradation

Cable Aging Degradation

Materials Research Key Activities

The research outcomes from this program will be used by utilities, industry groups, and regulators to inform operational and regulatory requirements for materials subjected to long-term operation conditions

- **RPV: Develop a predictive model for embrittlement validated through experiments, surveillance samples, and ex-service materials.**
- **Corrosion: Understanding mechanisms of IASCC failure and SCC initiation of stainless steels and Ni-base alloys – predict and develop mitigation strategies.**
- ***Concrete degradation: Develop a fully coupled thermo-hydro-mechanical-chemical-radiation damage model for predicting the performance of concrete structures.***
- ***NDE: Develop better condition monitoring techniques for cables & concrete structures.***
- **Cable degradation: Understand modes, predict performance and evaluate rejuvenation strategies and replacement materials.**
- **Mitigating Damage: Develop Advanced Replacement Materials. (Joint Project with EPRI)**
- **Mitigating Damage: Development of procedures, techniques and computational modeling for advance weld repair of irradiated steels. (Joint Project with EPRI)**
- ***Harvesting service-aged materials: Assess and improve codes, standards, & predictive models***

Materials Research: What's Next for extended Operation of NPPs?

- Complete development of predictive degradation models
- Refine predictive models through Codes and Standards evaluations for use by the nuclear industry
- Continue / improve LWRS engagement with stakeholders (utilities, vendors, NRC,…) to solve critical sustainability issues

- Extended Operation (LBE): How should we prepare for a possible need to provide electrical capacity from the existing LWR fleet?

Stakeholder Engagement Meeting LBE Objectives (1)

Initiating a plan for LWRS MRP LBE Research Options

- **Develop objectives and goals based on issues presented today**
- **Develop a path forward to assess goals**
- **Develop Key Performance Indicators**

Obtain feedback from LWRS MRP Stakeholders (Utilities, Vendors, EPRI, US NRC, University Researchers, and MRP staff) on:

- How would our **Stakeholders assess our MRP LBE Research Options**?
- **Industry Needs: What should LWRS MRP be addressing (key priorities and direction) for an LBE path forward?**
- How can the LWRS MRP **collaborate on LBE with our Stakeholders**?

Stakeholder Engagement Meeting Objectives (2)

Establishing a path forward:

- **Identity knowledge gaps (SEMDA?)**
- **Review and identify key priorities and timelines to reach goals**
- **Establish partnerships with stakeholders** (PWROG, BWROG, EPRI, NRC, and universities) **to develop research plan**

Sustaining National Nuclear Assets

http://lwrs.inl.gov

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Reactor Pressure Vessel Aging at Extended Operation - Thermal Annealing of Reactor Pressure Vessels

Mikhail A. Sokolov

Materials Science and Technology Division,
Oak Ridge National Laboratory

Subsequent License Renewal Has Been Driven by Ability of Critical Materials to Satisfy Current Regulatory Positions

- For Turkey Point Units 3 and 4, the circumferential beltline RPV weld is the critical component and it has passed acceptance, by mostly, applying alternative PTS rule.
- It is highly likely scenario that most Units that are applying for SLN be using this approach as well.
- While this is the very valuable point, once you used up the margin provided by AltPTS rule, the Power Units are out of safety margin.
- The only proven technology that has been existed and been proven in the US to extend life of PRV (the same as life of the Power Unit itself) is thermal annealing

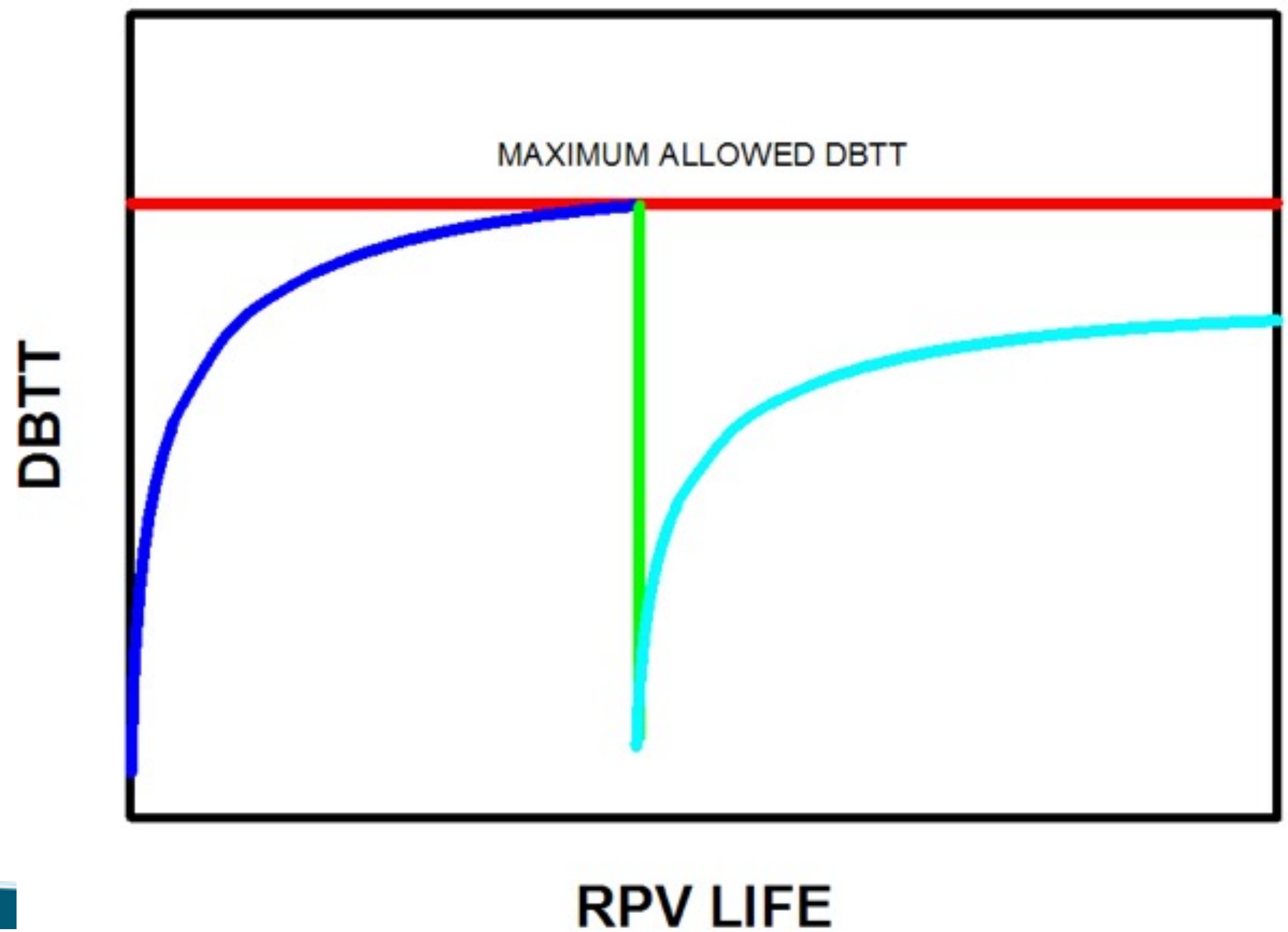
What is or is not RPV Thermal Annealing?

- It is not a traditional metallurgical anneal at temperatures up to about 1000^{o}C
- It is a localized area heat treatment of the RPV (at lower temperatures between 340^{o}C and 500^{o}C) and for a long holding time (up to 168 h)
- Material mechanical properties are partially or fully recovered, but resulting fine scale microstructure is different
- Re-irradiation response can be different due to the different starting microstructure

Vessel Annealing

- Thermal annealing is the only proven option that can recover irradiated beltline material transition temperature shift and recover upper shelf energy properties lost during radiation exposure and extend RPV service life
- Beltline region is heated to 340 to 560°C
- Amount of mechanical property recovery is function of:
  - Difference between the irradiation and thermal anneal temperatures
  - Time of annealing
  - Material chemistry
  - Degree of pre-existing irradiation damage

RPV Thermal Annealing



Vessel Annealing – Wet Anneal

- Two basic types of annealing
  - Wet anneal
  - Dry anneal
- Wet anneal is performed at temperatures < 650°F (343°C)
- Reactor coolant water is generally heated by the RCPs
- Wet annealing is not as complicated from an engineering viewpoint because primary water temperature is controlled by pump heat up to the vessel design temperature of 343°C
- Wet anneals have been successful on two test reactors, SM-1A (US Army, Alaska, 1967) and BR3 (Belgium, 1984) and they operated near 260°C for a short time after annealing

Vessel Annealing – Dry Anneal

- Dry anneals are performed at higher temperatures than wet anneals
  - Use air as the heating medium inside of radiant can
  - Electric-resistance heating source
- Dry annealing requires removal of core internal structures and primary water so that a radiant heating source can be inserted near the vessel wall to locally heat the embrittled beltline region
- Engineering difficulties of dry anneal process are quite complex and may need plant-specific evaluations to assure that other portions of the plant (eg., concrete) are not harmed by the high annealing temperatures

VVER-440/1000 RPV Thermal Annealing

Reactor	Year	Temperature/time	SS clad
Novovoronezh-3	1987	430±20 °C/168h	No
Armeniya-1	1988	450±50 °C/168h	No
Greifswald-1	1988	475±15 °C/150h	No
Kola-1	1989	475±15 °C/150h	No
Kola-2	1989	475±15 °C/150h	No
Kozloduy-1	1989	475±15 °C/150h	No
Kozloduy-3	1989	475±15 °C/150h	Yes
Greifswald-2	1990	475±15 °C/150h	No
Greifswald-3	1990	475±15 °C/150h	Yes
Novovoronezh-3 (re-anneal)	1991	475±15 °C/150h	No
Novovoronezh-4	1991	475±15 °C/150h	No
Kozloduy-2	1992	475±15 °C/150h	No
J. Bohunice-2	1993	475-503 °C/160h	Yes
J. Bohunice-1	1993	475-496 °C/168h	Yes
Loviisa-1	1996	475±15 °C/100h	Yes
Rovno-1	2010	475±15 °C/150h	Yes
Balakovo-1 (VVER-1000)	**2018**	**560C/100h**	**Yes**

Vessel Annealing Recovery Results in US

- Annealing recovery test results on US RPV steels found:
  - Annealing at 850°F (454°C) resulted in complete recovery of USE, and 75% or more recovery of the Charpy 41 J transition temperature shift
  - Annealing at 343°C provided significantly less benefit
  - EPRI report TR-106001, Dec. 1995 reported results of some irradiation embrittlement and re-annealing studies for Yankee Rowe related materials
    - Annealing at 454°C resulted in recovery of 80-100% of the transition temperature and 100% recovery of the USE
    - Annealing at 343°C resulted in about a 40% recovery in transition temperature

Dry Annealing Evaluations

- 454°C is regarded as an optimum “dry” annealing temperature
- In 1980s a study at INL assessed annealing feasibility for US RPVs including alternative heating methods
- In 1995 a study was conducted to determine if thermal annealing of the reactor vessel in Westinghouse 3 and 4 loop plants is feasible
  - Thermal and stress analyses determined that stress, temperature and dimensions of the vessel and its associated components remain within acceptable limits
  - Conclusion: there are no major technical impediments to thermal annealing the vessels studied
- In 1990s number of research on effect of annealing on recovery of irradiated RPV steels was performed as ORNL and UCSB as part of HSSI Program

Marble Hill Demonstration Project

- In 1990's a joint DOE/industry-sponsored Annealing Demonstration Project (ADP) was conducted at the Marble Hill facility (a partially completed Westinghouse plant) to demonstrate feasibility
  - Nozzle-supported four loop Westinghouse design vessel -- canceled plant (unirradiated vessel)
  - Indirect gas-fired heating method was chosen
  - DOE funding lost after demonstration was completed, but EPRI funded writing the Marble Hill report (EPRI TR-104934) and a final NRC report (NUREG/CR-6552) was also later published

Midland Demonstration Project

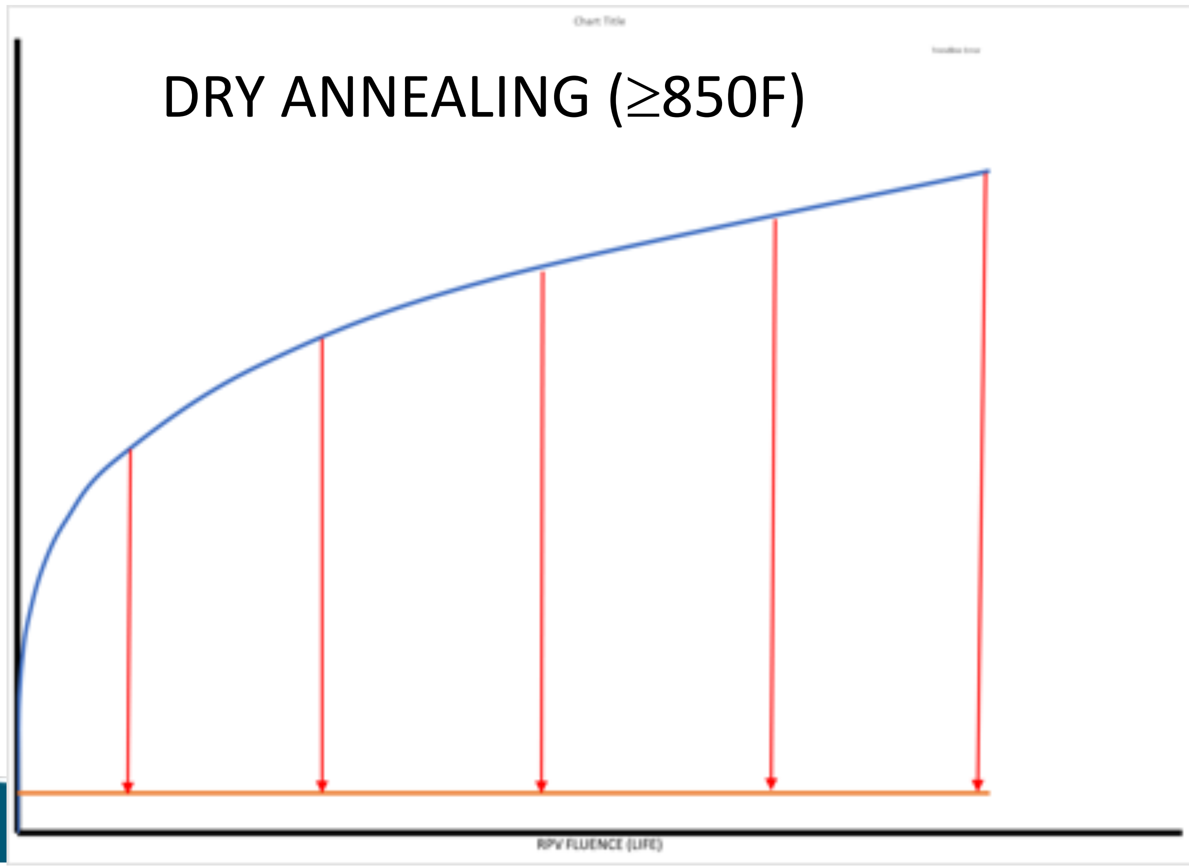
- Skirt-supported Babcock & Wilcox-design vessel
- Electric resistance heating method (Russian technology and experience)
- Project approximately 50 percent complete when DOE funding eliminated
- Electric resistance heater fabricated and tested, but never shipped to the US from Russia
- Demonstration never completed

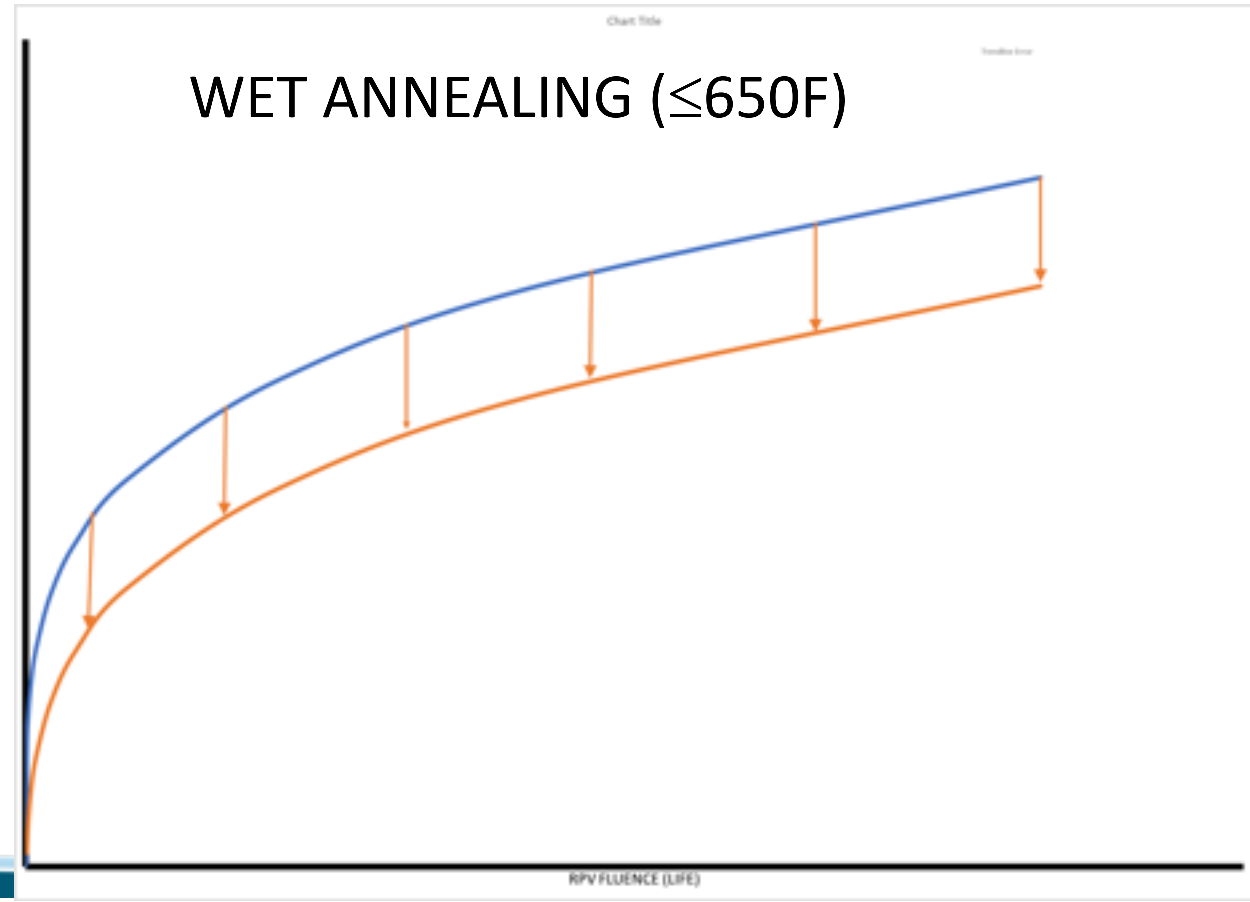
Why the Demonstration? – Palisades NPP

- Palisades was limited to operate until 1999 based upon PTS concerns for the most-limiting weld metal heat (W5214 axial welds); other welds: 27204 and 34B009
- Material chemistry variability issue as a result of sampling of retired steam generators containing two of the same key welds (W5214 and 34B009)
- Planned to anneal in 1998 to recover properties and continue operation to at least 2011 and hopefully beyond
- Developed a supplemental surveillance program to assess material recovery and re-embrittlement trends for all beltline welds and the surveillance plate material
- Annealing canceled due to revised fluence estimates; also concern about public hearings when authorized to anneal

Supplemental Surveillance Program

The supplemental surveillance program can be achieved by re-use of previously tested surveillance specimens using small specimen test techniques that are already exist





Concluding Remarks

- Technology exists and has been proven for performing thermal annealing on VVER-440/1000 and PWR RPVs in the beltline region
- Decision to anneal may involve more than technical or cost issues
  - When annealing is technically warranted, it should be seriously considered and not discarded as a "last resort"
  - Even if thermal annealing is not technically needed, a decision to anneal could benefit extended long term operation
    - Guarantee that the fracture toughness properties are improved
    - Re-embrittlement rate should be reduced

BACKUP SLIDES

Guidance for Thermal Annealing

- 10 CFR 50.66, "Requirements for thermal annealing of the reactor pressure vessel"
  - Permits thermal annealing of LWRs
  - Requires a plan for conducting the thermal annealing be submitted at least three years before fracture toughness criteria are exceeded
  - Reg Guide 1.162 describes the format and content of an acceptable Thermal Annealing Report (TAR) and addresses the metallurgical and engineering issues that need to be addressed in an application to perform a thermal annealing

ASTM Activities

- ASTM E 509 revised in 1997 to provide expanded guidance on thermal annealing and the necessary supplemental material surveillance programs; ASTM E 509 was further updated in 2003
- Splitting of old ASTM E 185 into two new Practices (ASTM E 185-02 on Surveillance Program Design and ASTM E 2215-02 on Testing of Surveillance Capsules); both emphasize and encourage fracture toughness Master Curve testing in addition to CVN testing
- Small specimen test techniques may be applied to annealing applications using other ASTM guides
  - ASTM E 636 on Supplemental Test Techniques
  - ASME E 1253 on Charpy Specimen Reconstitution

ASME Code Case N-557

- "In-Place Dry Annealing of a PWR Nuclear Reactor Vessel (Section XI, Division 1)"
- Provides Code guidance for assuring design conformance after performing a thermal anneal heat treatment
  - Limits magnitude of thermally induced stresses in nozzle region
  - Effectively limits the maximum temperature of annealing to 505^{o}C
  - Passed in 1995 in anticipation of Palisades NPP thermal anneal
- Technical basis published by EPRI in TR-106967

Key Annealing Issues as Related to Long Term Operation

- Based on NRC Regulations and guidance – evidence of dose rate on annealing recovery at low annealing temperatures (less than 427^{o}C)
- Based on guidance in ASTM E 509
  - Re-embrittlement rate and surveillance during extended life (including any effect of dose rate)
  - Potential enhancement of P segregation and intergranular fracture
- Based on ASME Code Case – must minimize thermally induced stresses in nozzle region, which effectively limits maximum temperature of annealing to 505^{o}C

Emerging materials issues of austenitic internals for PWR operation to 80 years and beyond

F. A. Garner

Texas A&M University

Radiation Effects Consulting

M. N. Gussev

Oak Ridge National Laboratory

Overview of presentation

- Within the current 40-year licensing period of PWRs a number of materials issues of first-order importance have been addressed for austenitic internals, focusing primarily on IASCC and embrittlement.
- Second-order issues have long been recognized but previously have not been considered to be life-limiting.

  Void swelling and irradiation creep

  Transmutation (gaseous and solid)

  Helium-hydrogen synergisms

  Radiation-induced phase instabilities

  Undefined magnetic phases
- However, the non-linear nature of these second-order processes and their possible synergisms causes worry that they might become first-order with life extension to 60 years.
- With current focus on lifetimes of 80 years and beyond these "second-order" issues must be considered as even more worrisome.

- What is the current status of understanding for these second-to-first order effects?

Second-to-first order concerns, primarily for PWRs and secondarily for BWRs, with major focus on 304 SS

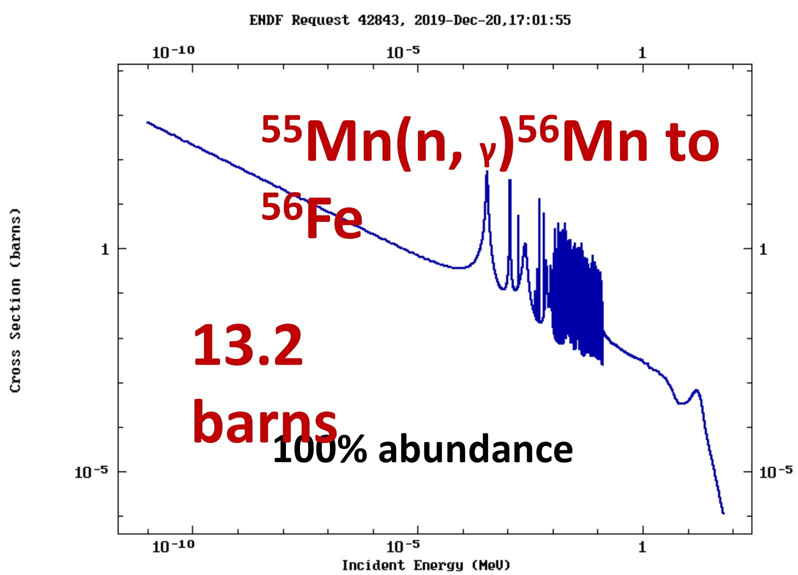
- Transmutation at high exposures (Mn, V, He, H)
- Phase instabilities (deformation martensite, Fe-rich alpha ferrite)
- Void swelling reaching 1%/dpa or not?
- Irradiation creep is higher in thermalized spectra?

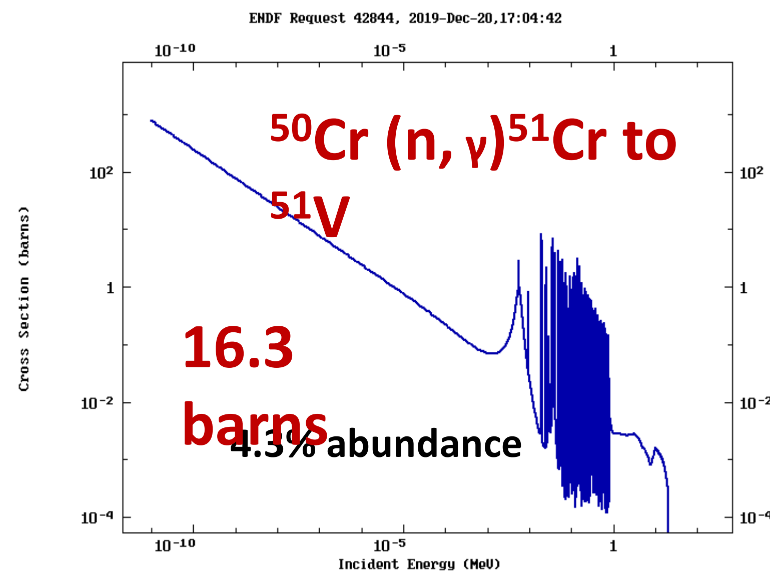
- Consequences of above on specific failure mechanisms (IASCC, embrittlement, repair welding)

Transmutation for stainless steels

Transmutation for stainless steels has been previously thought not to be a significant issue with the exception of helium produced by ^{59}Ni (n, α) reaction.

- **At higher exposures, the burnout of Mn and burn-in of V may become an issue for phase stability and IASCC, especially when combined with reverse segregation at grain boundaries.**



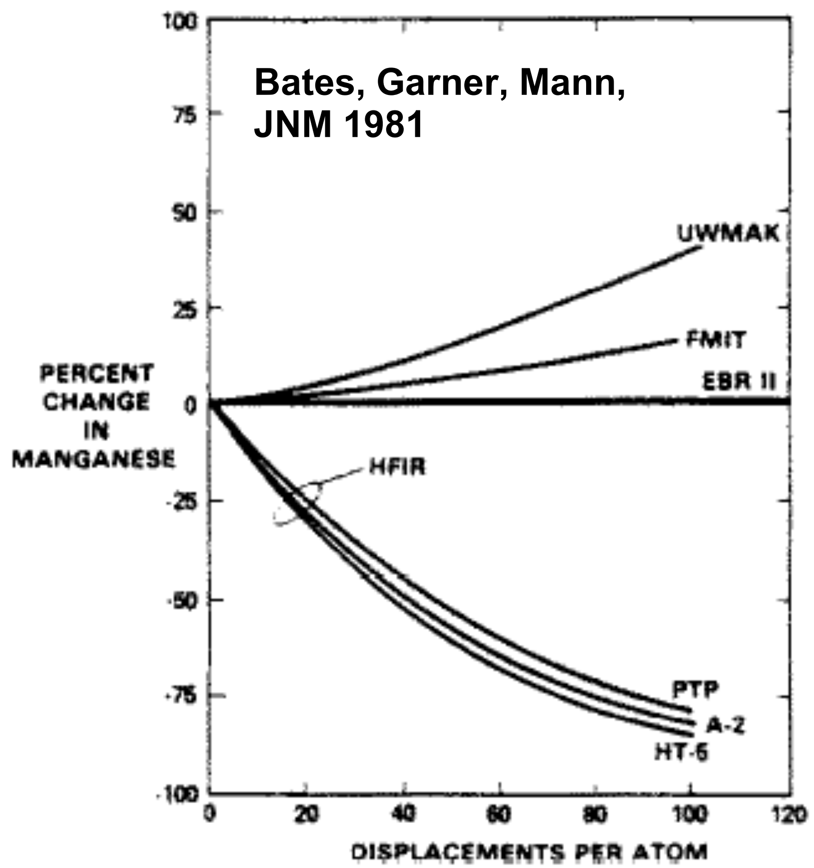


A major role of Mn in 300 series steels is to remove S from the matrix and keep it off grain boundaries where it contributes to cracking.

In 304 SS V is not deliberately added but may exist at low tramp levels.

Mn in thermal neutron spectra transmutes to Fe

New issue of importance to light water reactors



Mn has one isotope Mn-55 which transmutes to Fe-56 at 13.2 barns.

Mn is an austenite stabilizer, and its loss contributes to instability for low-nickel steels such as 304 stainless.

Primary role of Mn is to sequester S and keep it off grain boundaries.

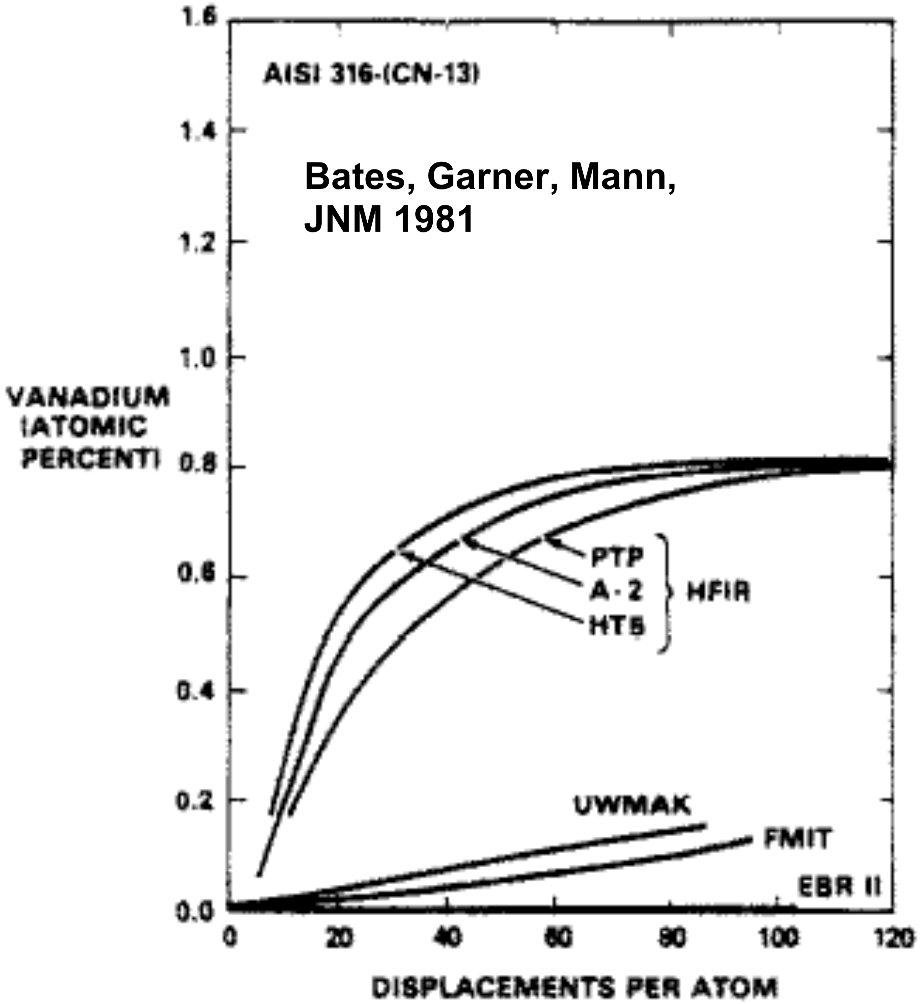
MnS precipitates are thermally very stable, but FeS precipitates are not as stable.

Mn is a fast diffuser and migrates up vacancy gradients away from grain boundaries and MnS surfaces.

Mn loss by transmutation and RIS depletion will be highest at the water-metal interface.

These losses at grain boundary intersections on the surface will encourage corrosion and crack nucleation.

Cr in thermal neutron spectra transmutes to V



Cr has four isotopes with numbers 50, 52, 53, 54 with abundances of 4.3, 83.8, 9.5 and 2.4%.

$$^{50}\mathrm{Cr}(n,\gamma)^{51}\mathrm{Cr} \rightarrow {}^{51}\mathrm{V}$$

Total burnout of ^{50}Cr produces only 0.8% V in 316 stainless steel, but with respect to carbide stability, this is a rather large amount.

Activity of carbon is one of the important factors that determine swelling, and also influences other processes such as corrosion, cracking susceptibility.

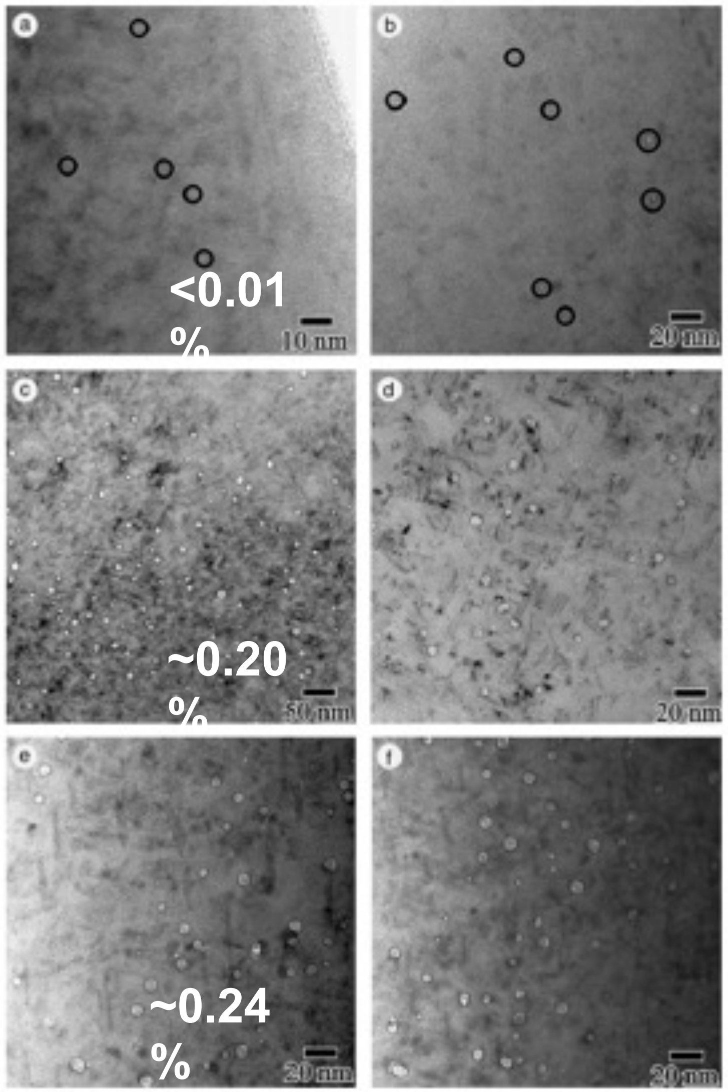
V is a carbide-forming element.

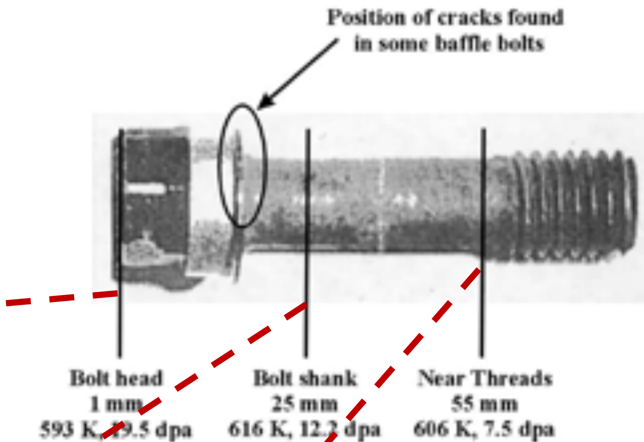
V is a ferrite stabilizer, five times stronger than the Cr it replaces.

This loss at the surface will encourage crack nucleation.

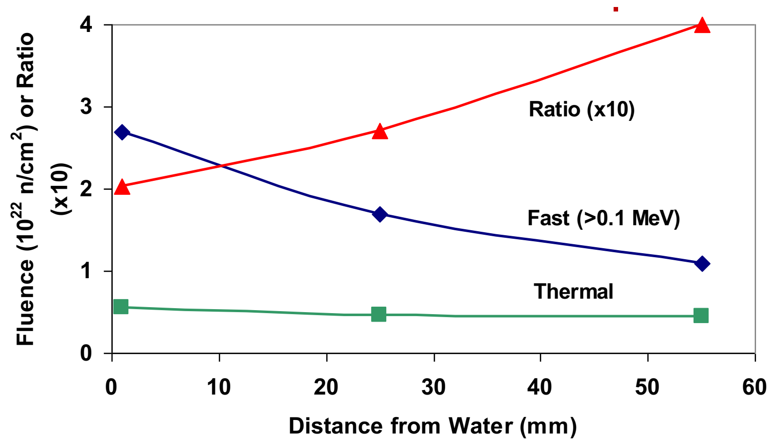
Gradient in T/F ratio along a 316 stainless steel PWR baffle bolt with low levels of void swelling

Edwards, Simonen, Garner, Greenwood, Oliver, Bruemmer, 2003





T/F ratio varied from ~0.2 at the bolt head to ~0.4 at the near threads position.



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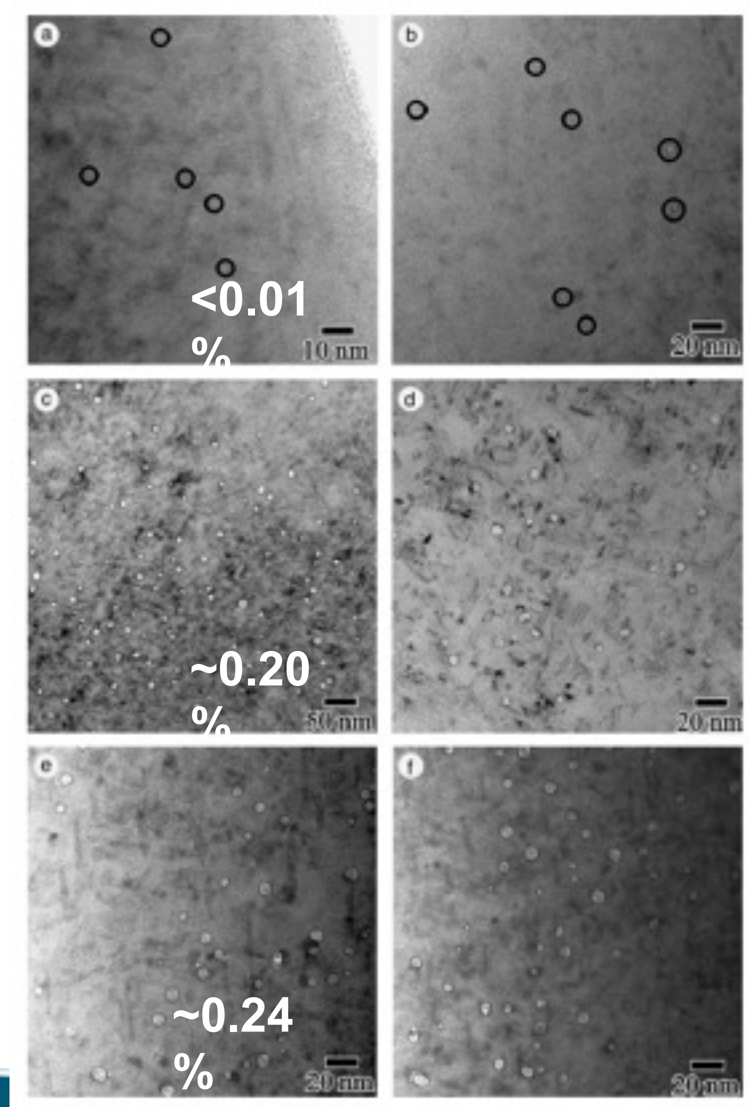
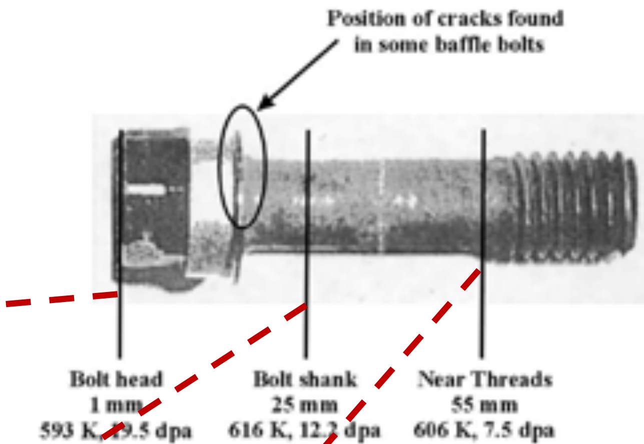
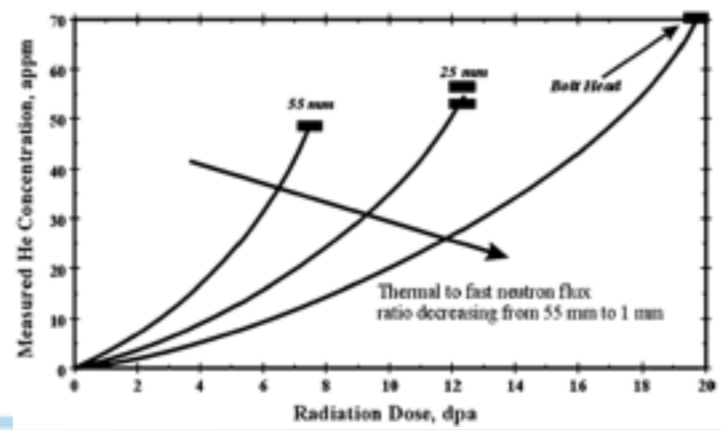




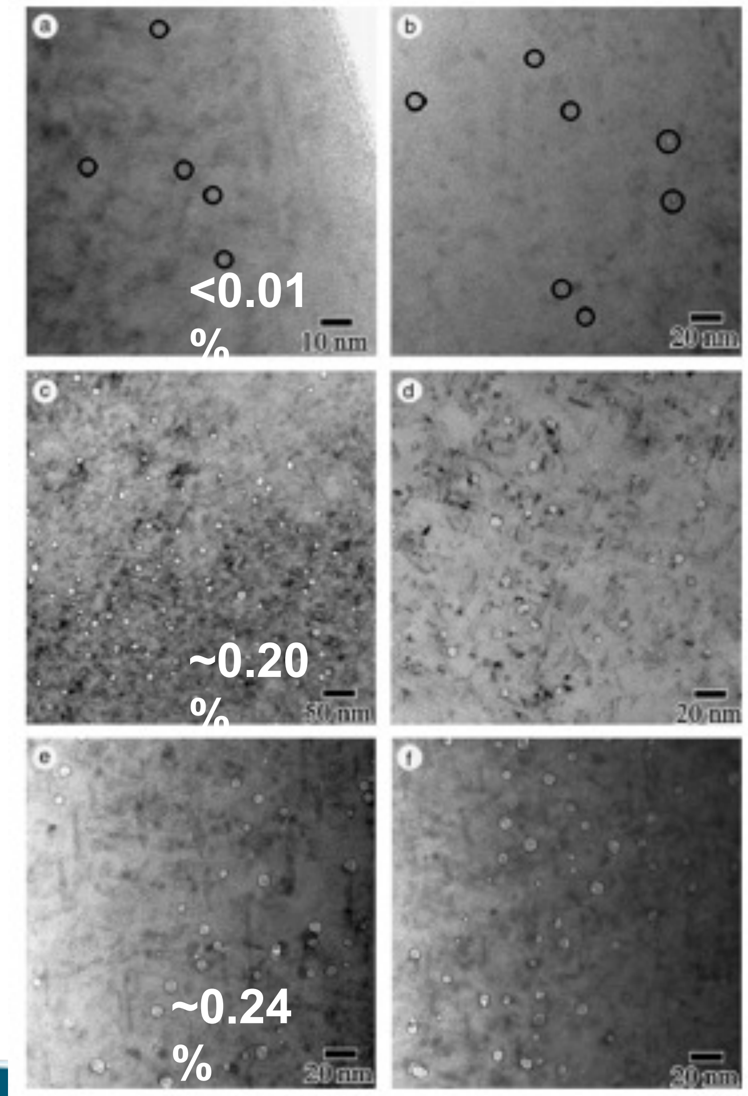
T/F ratio varied from ~0.2 at the bolt head to ~0.4 at the near threads position, producing a higher rate of helium per dpa at the lowest dpa rate.

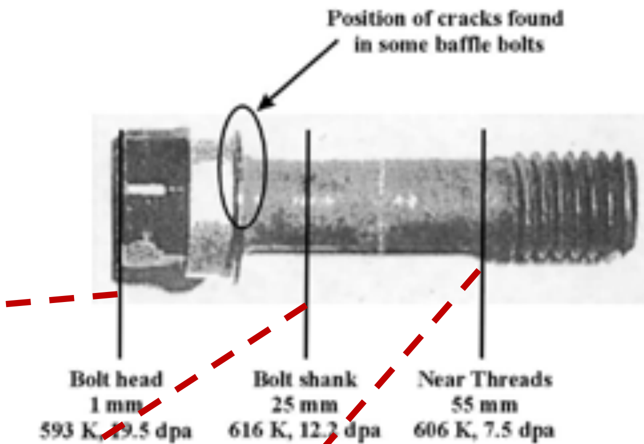
Helium , appm



Gradient in T/F ratio along a 316 stainless steel PWR baffle bolt with low levels of void swelling

Edwards, Simonen, Garner, Greenwood, Oliver, Bruemmer, 2003



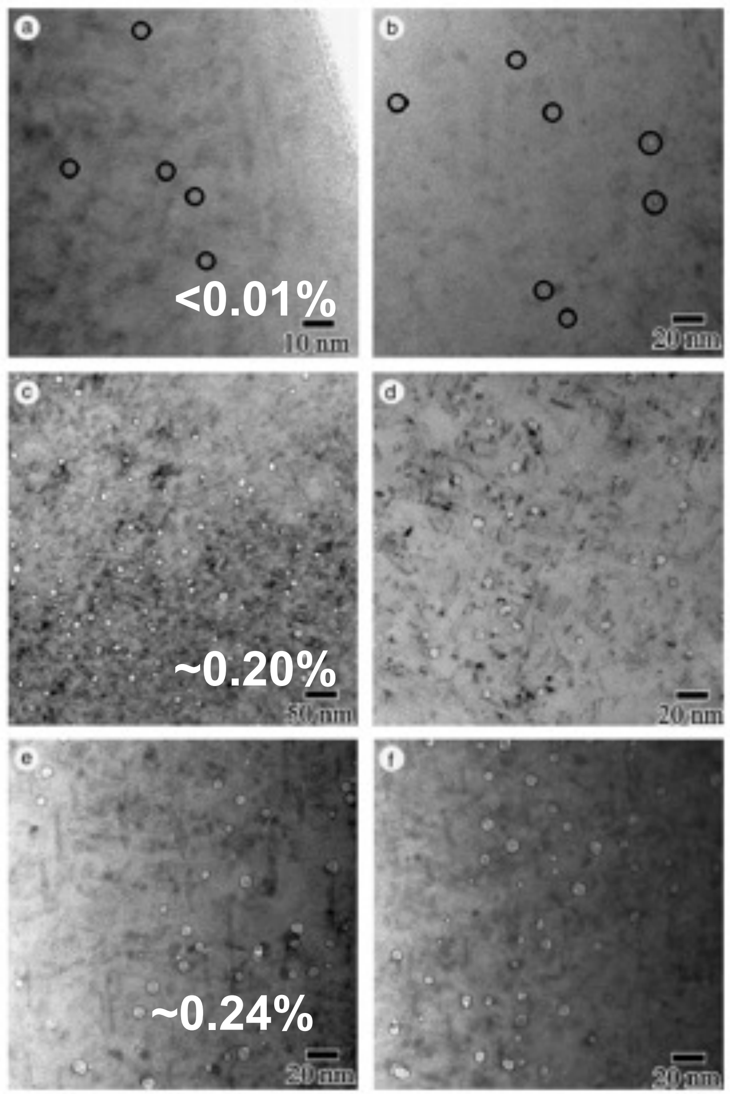


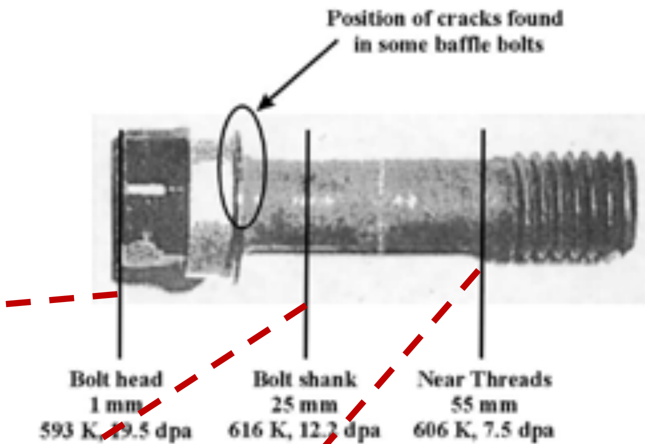
Coincident with void swelling Mn is being transmuted to Fe.

Distance from bolt head, mm	dpa	Burnout loss of Mn
1	19.5	7.1%
25	12.2	5.9%
55	7.5	5.6%

Gradient in T/F ratio along a 316 stainless steel PWR baffle bolt with low levels of void swelling

Edwards, Simonen, Garner, Greenwood, Oliver, Bruemmer, 2003



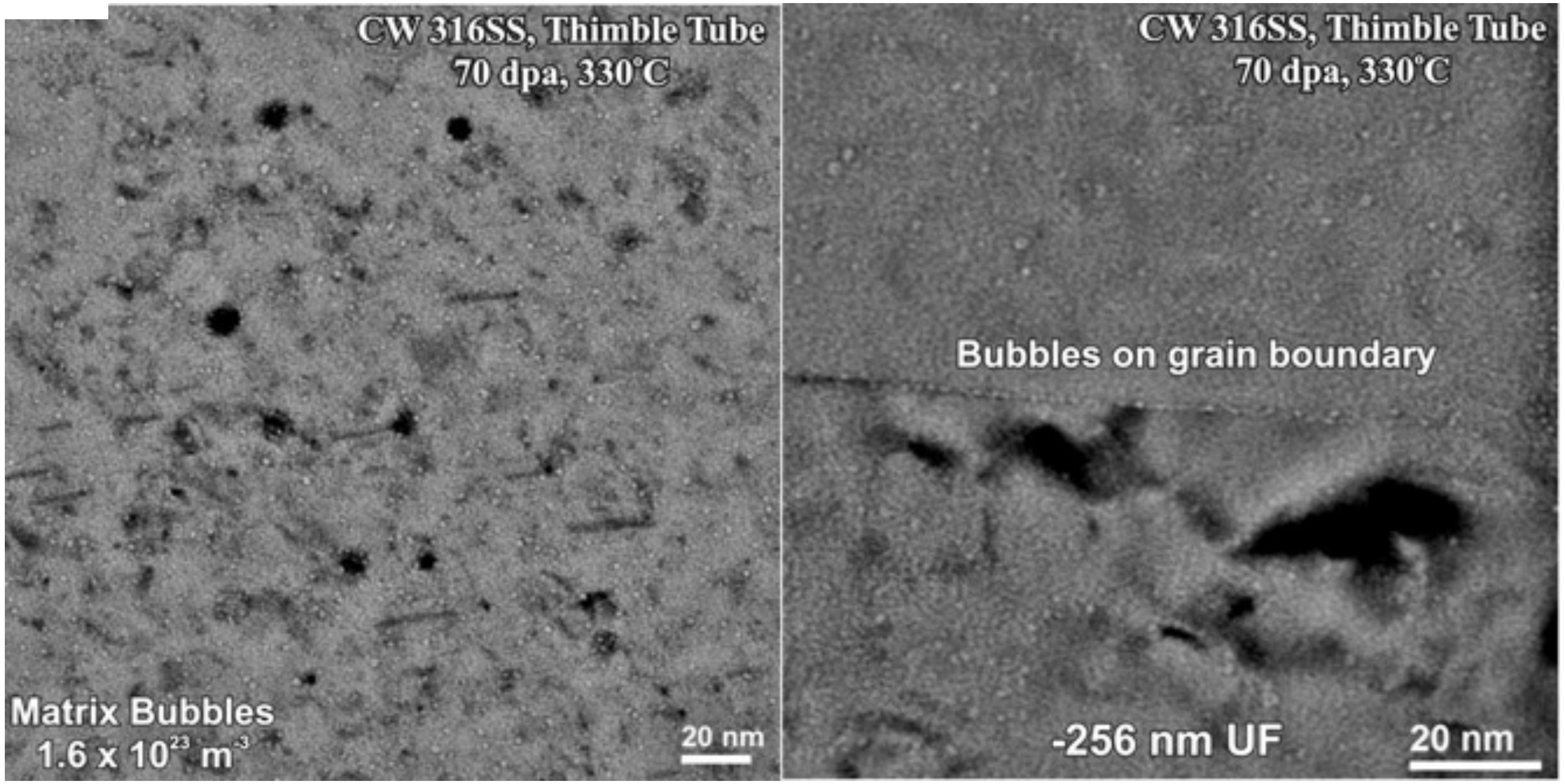


Large amounts of environmental hydrogen, in excess of transmutation, was found to be stored when significant swelling occurs.

Position (mm)	^{1}H (appm)	^{4}He (appm)
1	493, 743	71
25	720, 1260, 3660, 3710	52.7
55	1840, 3740	48.8

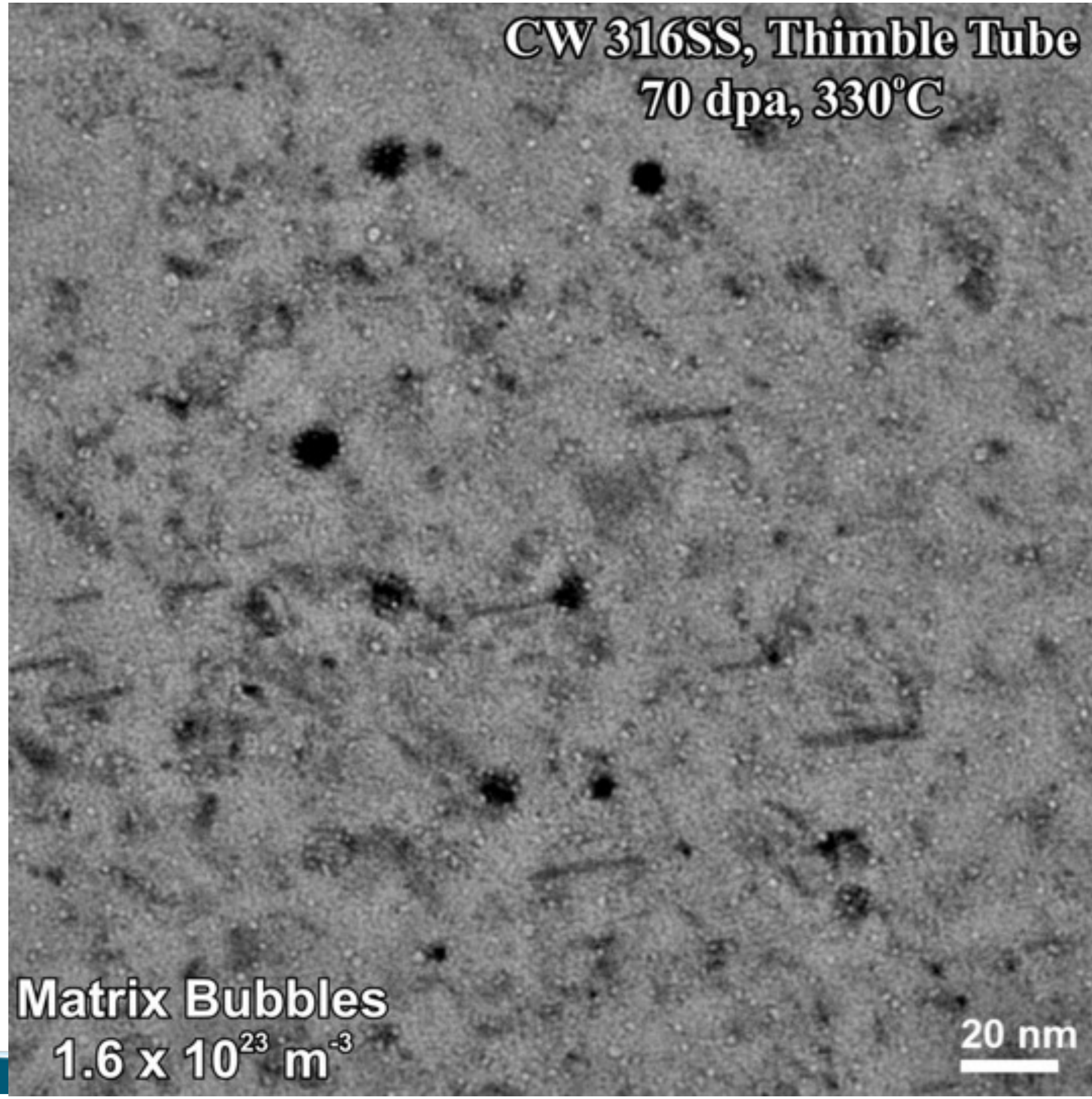
Nano-cavities in a PWR Flux Thimble Tube

Edwards, Garner, Bruemmer and Efsing, 2009



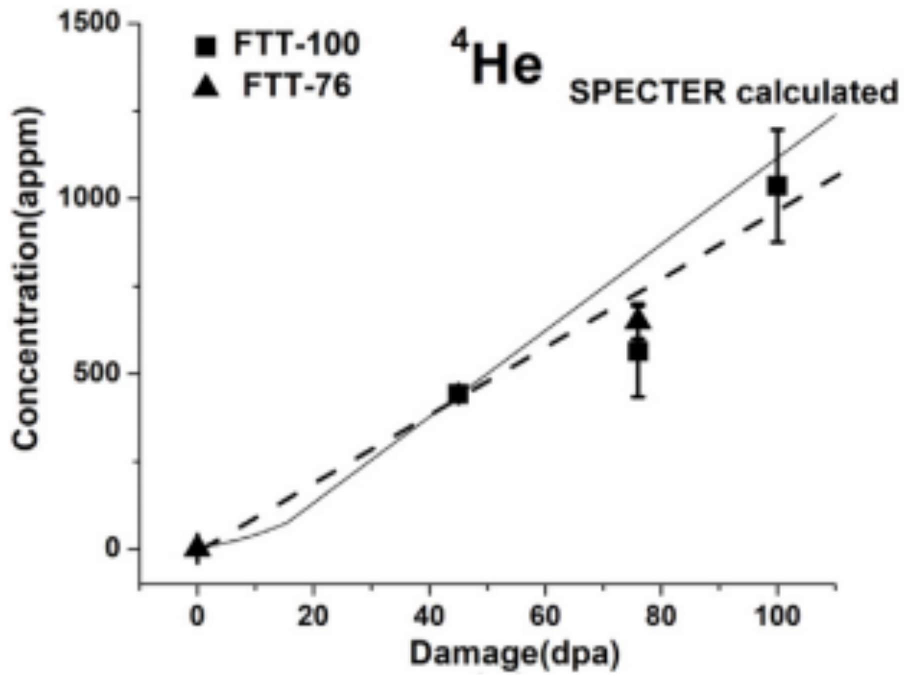
- **High density of nano-bubbles (<3 nm diameter)**
- **~600 appm He, ~2500 appm H**

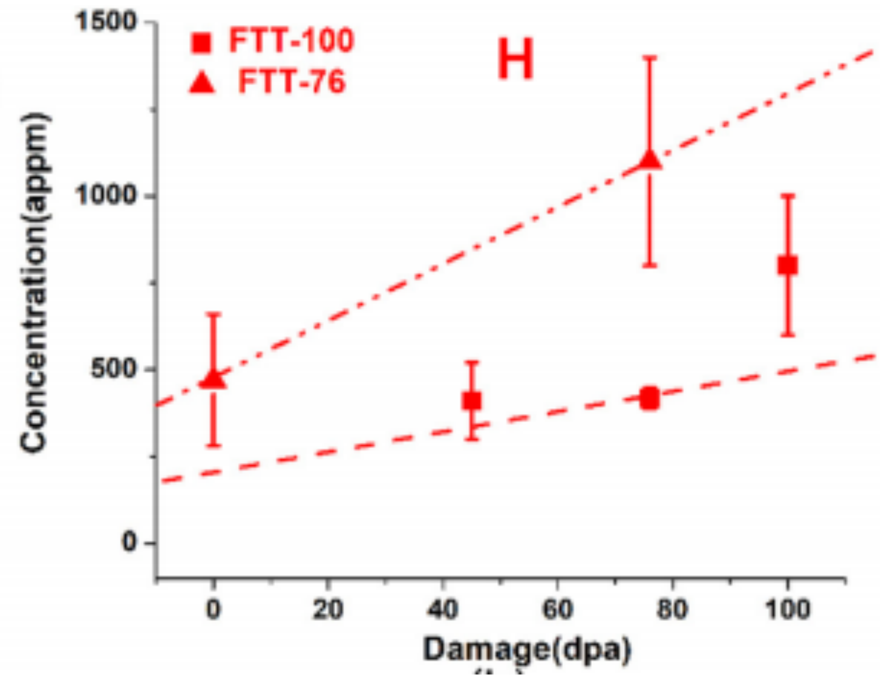
~1 dpa induced by Ni-59 reactions


CW 316SS, Thimble Tube
70 dpa, 330°C
Matrix Bubbles
1.6 x 10²³ m⁻³
20 nm

Recent measurements of He and H in PWR 316 SS flux thimble tubes to 100 dpa

Miao Song, Kevin G. Field, Richard M Cox, Gary S. Was , JNM 541, 2020





Measured helium still increasing with dpa in predictable behavior.

Hydrogen is more difficult to measure and not as predictable.

Determining Helium Content

He concentration less than 0.1 appm

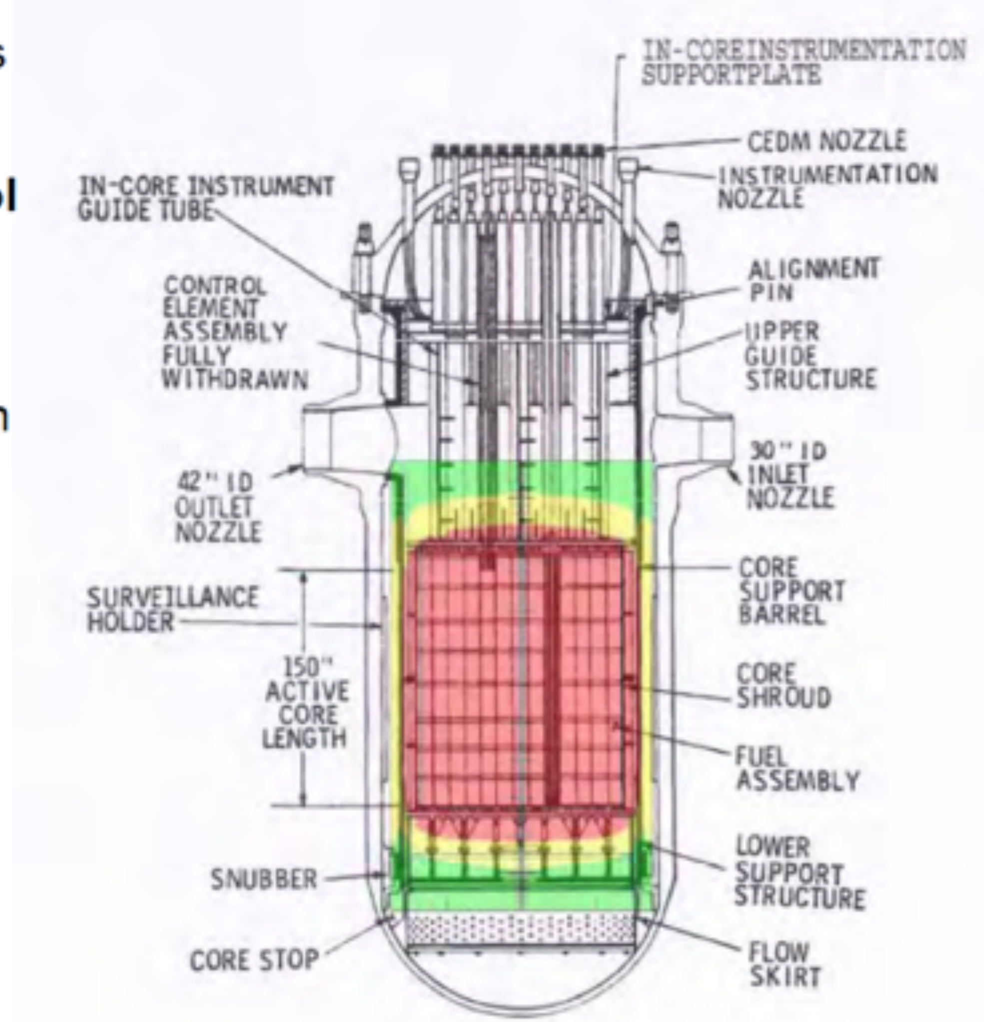
No heat input control required

He concentration greater than 0.1 appm and less than 10 appm

Heat input control required

He concentration greater than 10 appm

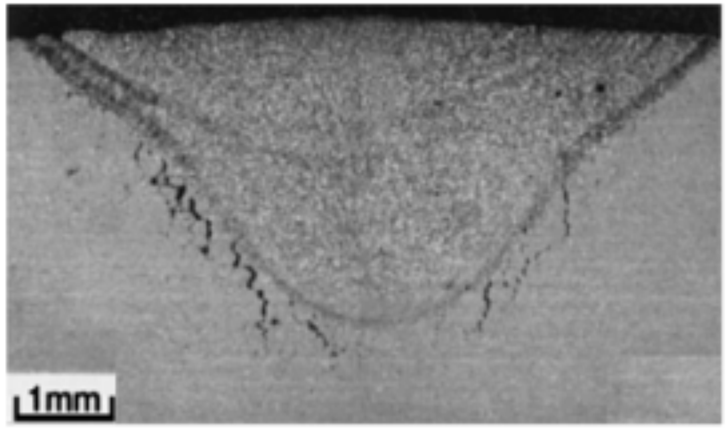
NOT Weldable with current technology



Weldability for CE reactor design at 75 ppm B and 60 EFPY

The Problem: Helium induced cracking during repair welding of irradiated materials

- Upon heating during welding, helium bubbles form and coalesce at the grain boundaries
- Upon cooling cracking in the HAZ of the weld occurs
- Conditions required for cracking depend on helium concentration, time at temperature (welding process thermal cycle) and stress state during cooling from welding



Asano et al. J. Nucl. Mat. 264 (1999)1-9

The Problem: Helium induced cracking during repair welding of irradiated materials

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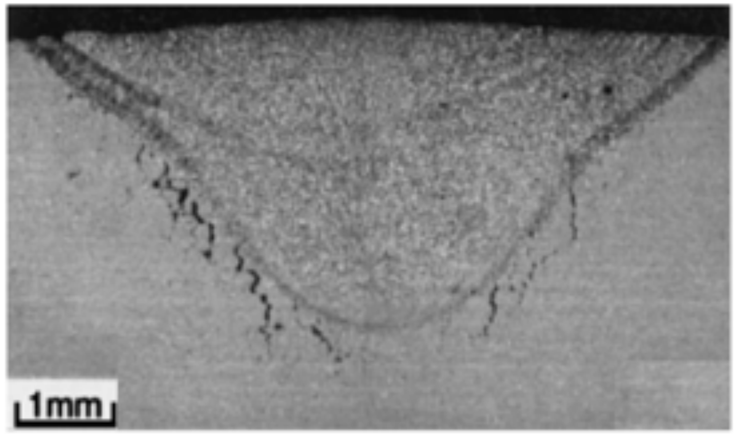
In a BWR at this dose there is very little or no swelling.

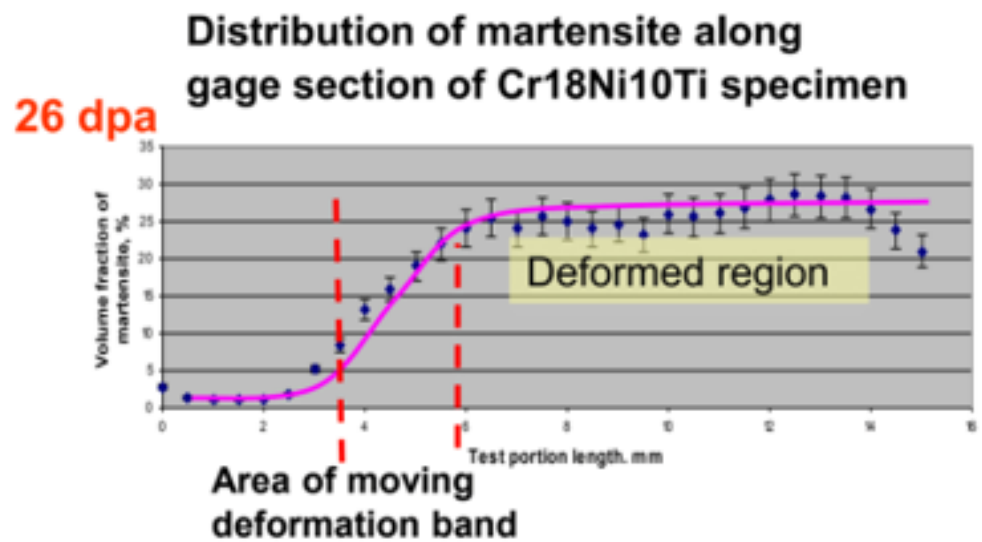


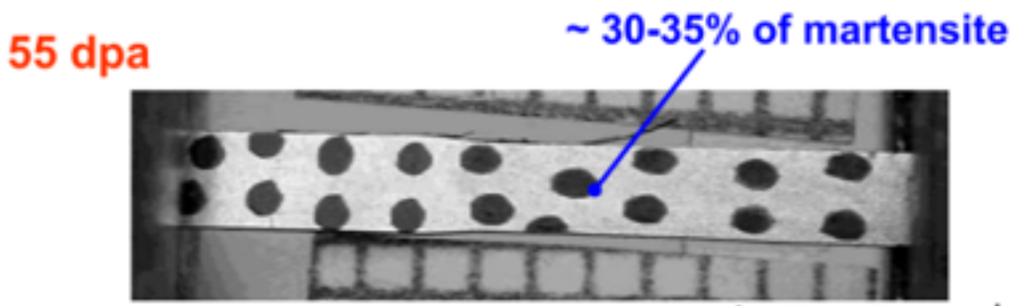
In a PWR the region where the cracks form will have voids with possible strong storage of helium and hydrogen.

Asano et al. J. Nucl. Mat. 264 (1999)1-9

Consequences of neutron-induced phase instability possibly arising from transmutation and segregation

Gussev, Maksimkin, Garner



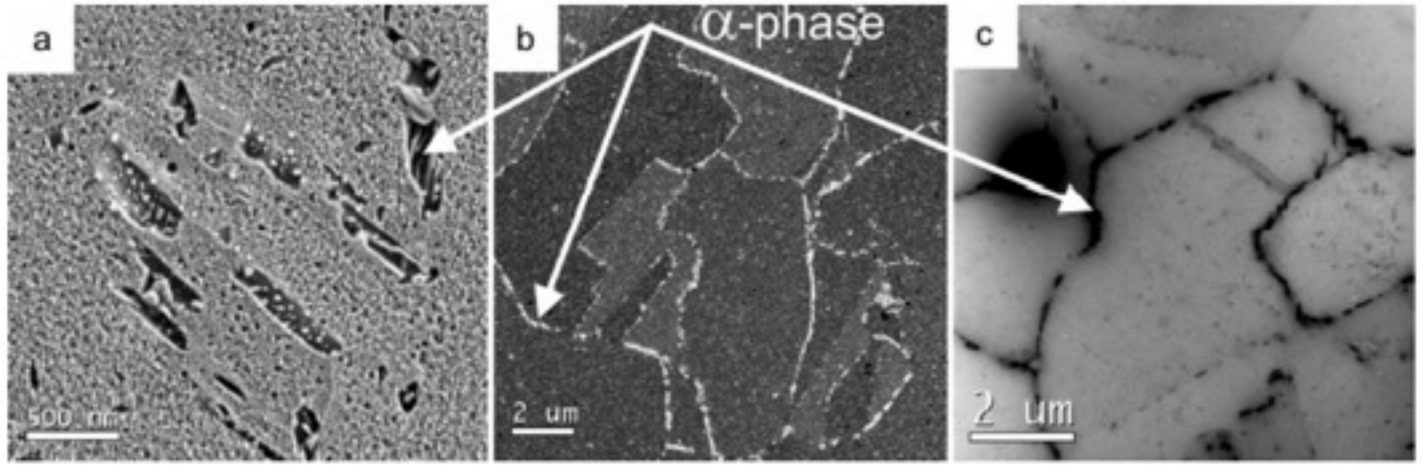


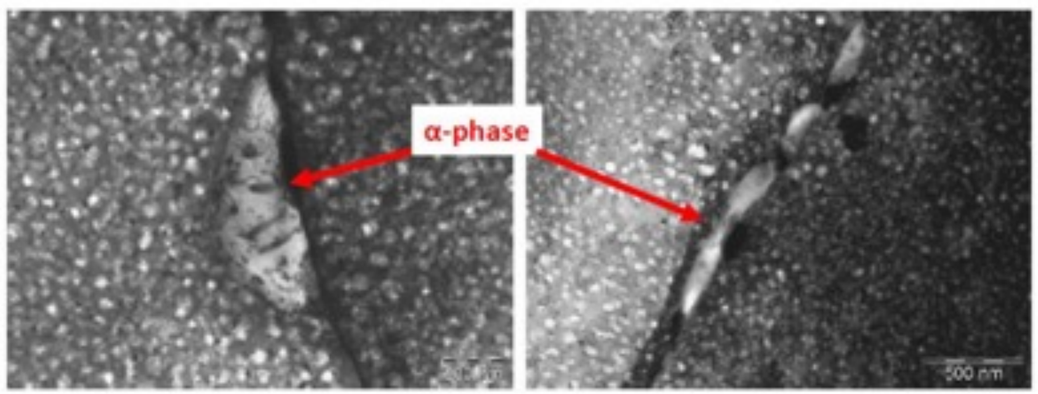
- Progressive increase in tendency to develop post-irradiation deformation-induced martensite at higher exposure levels.
- Moving deformation wave that resists necking and increases total elongation.
- Easily measured by increase in magnetic fraction.
- The cause of this phenomenon has not yet been identified but is suspected to involve transmutation and segregation.
- Potentially strong consequences on cracking and especially corrosion.

However, there is another phase instability that appears to be developing at very low dpa rates that also has a magnetic signature.

Development of Fe-rich ferrite phase in AISI 321 in BOR-60 reflector after 41 years at very low dpa rates

Gurovich (Kurchatov) and Margolin (Prometey) groups





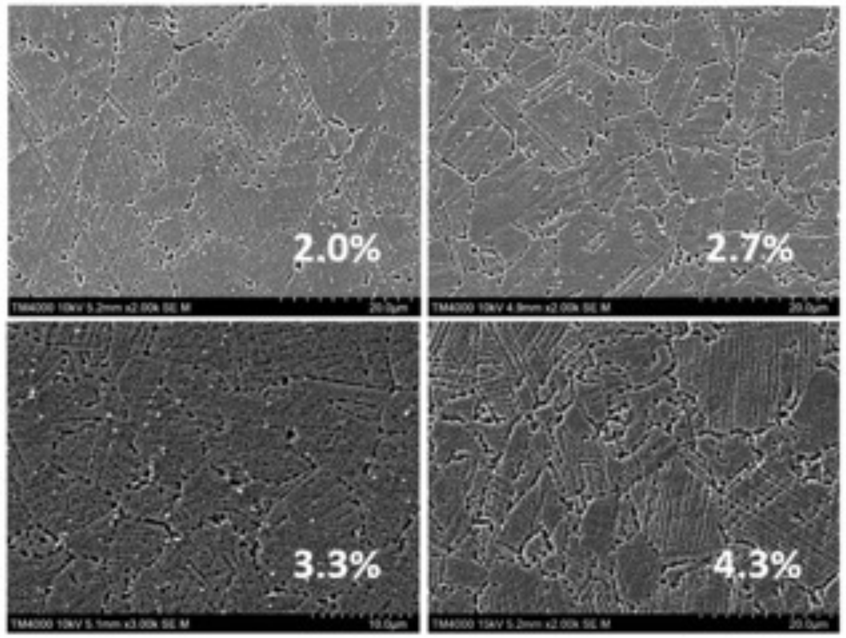
150 dpa peak
330-380°C

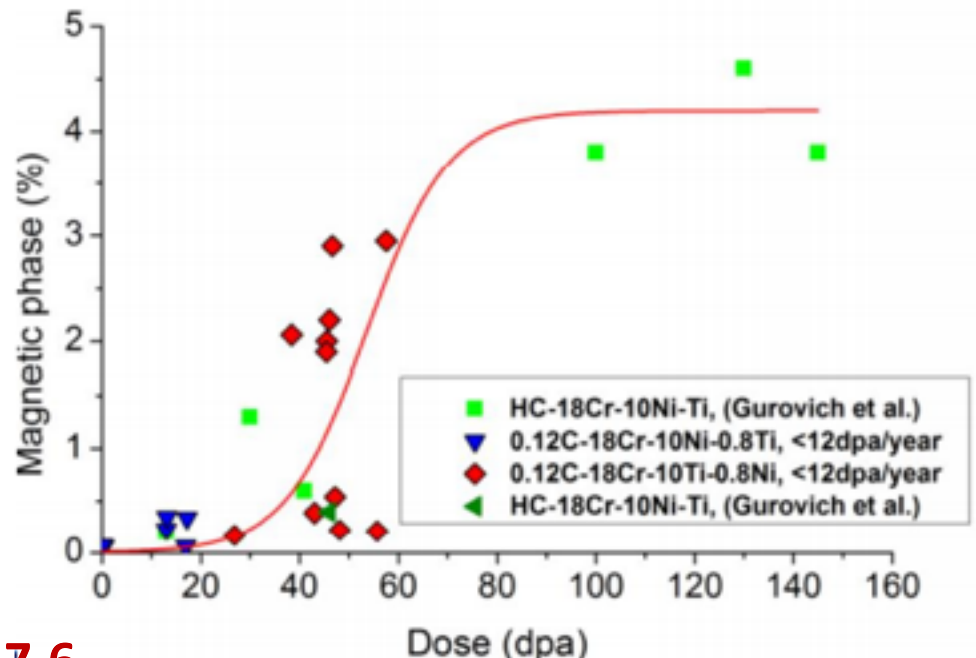
Reflector spectrum in BOR-60 has a very large epithermal neutron component that is strongly driving transmutation.

Increase in magnetism can be used to measure ferrite fraction but microscopy is difficult because ferrite is dissolved during specimen production.

Similar magnetic measurements of ferrite are being now obtained in collaboration with Maksimkin group in Kazakhstan from in-core components.

Major problem is producing specimens for TEM and STEM that retain the surface-intersecting ferrite during etching and polishing.





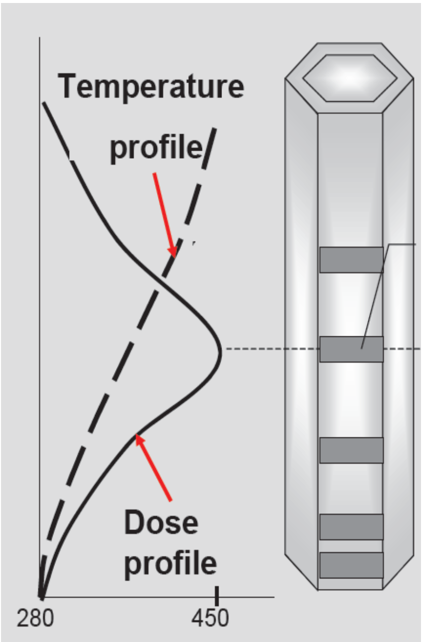
Magnetic fractions measured at 45.5–57.6 dpa at 354–406°C in BN-350 fast reactor.

This phase instability provides a potential for enhanced cracking due to easily dissolved (corroded) ferrite at surface intersecting grain boundaries.

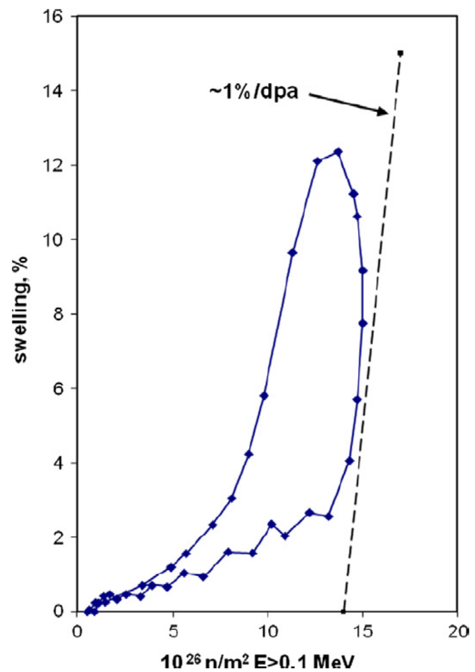
Behavior of swelling and irradiation creep at very high dose levels beyond 60-80 years?

- Swelling data for stainless steels at very high dpa levels was exclusively developed in fast reactors at high dpa rates.
- Essentially all data on 304 SS were derived from EBR-II which has an inlet temperature of 370°C, well above the temperature of most of PWR internals.
- Above ~400 °C the steady-state swelling rate is ~1%/dpa, regardless of dpa rate, temperature, stress, composition and thermal-mechanical production.
- ~1%/dpa has never been observed in any PWR component but might it be observed after a very long transient lasting ~60 years?
- Recent studies have shown that below ~370-380°C there is a transition from the high swelling rate to a lower rate on the order of <0.1%/dpa and falling with decreasing temperature.
- The transition temperature appears to be dependent on the dpa rate and can be seen in the outer reflector region of EBR-II.

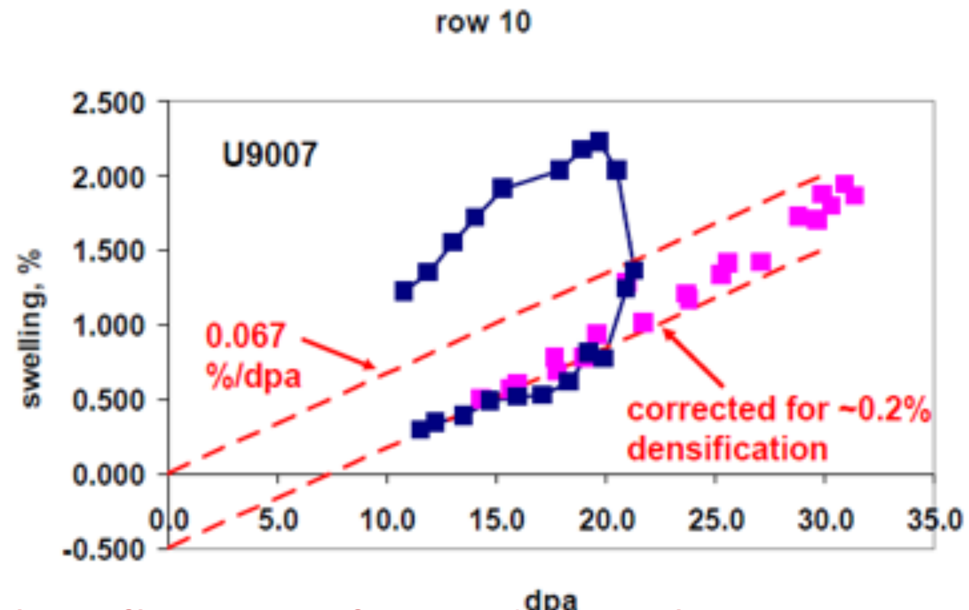
Swelling "loops" seen on two opposite sides one EBR-II Row 10 duct used to establish the current PWR swelling equation for 304 SS



Plot swelling data only as a function of dpa to produce a "swelling loop".



Loop observed for a high-flux in-core duct. Loops get thinner for some fluxes and operating temperatures.



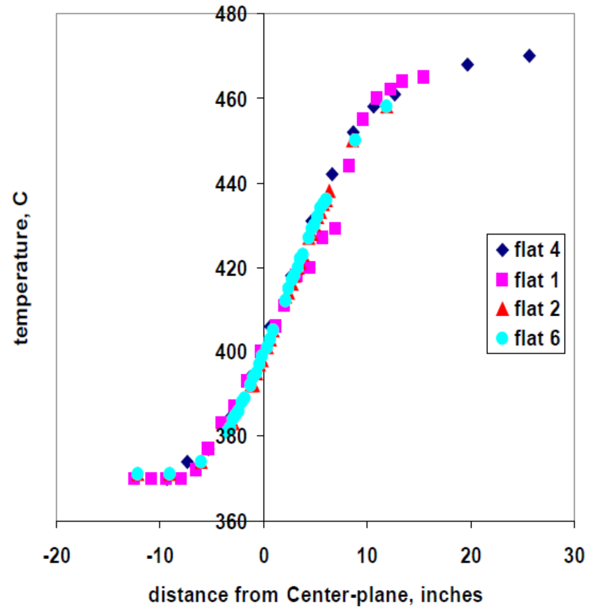
For a low-flux out-of-core duct at lower temperatures there is a ~30% gradient in dpa rate across two opposing faces.

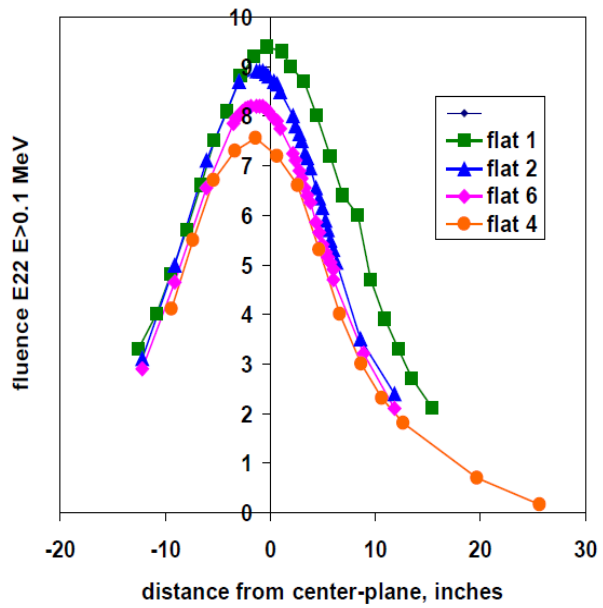
On the high-flux side the loop has zero thickness and a low swelling rate.

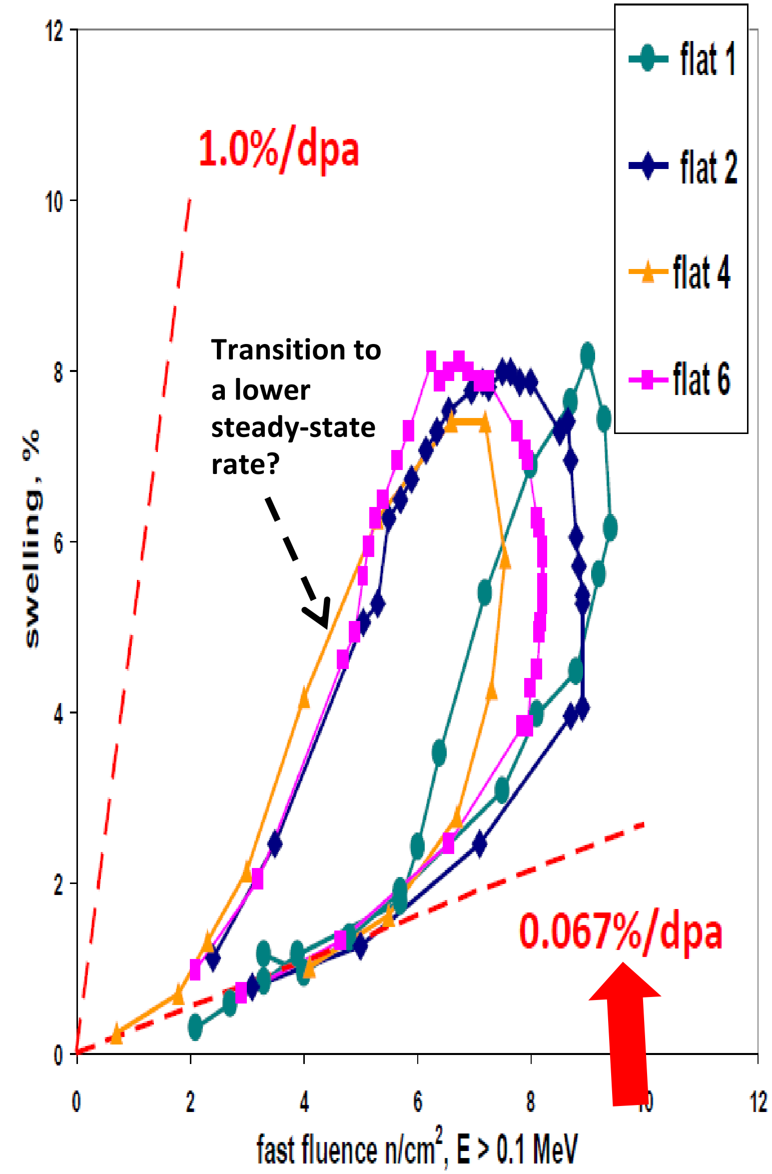
On the low-flux side there is a breakaway to a higher swelling rate at ~385°C.

Swelling observed in four faces of a hexagonal flux thimble tube in row 5 of EBR-II (dpa rates intermediate to PWR values)

Temperatures are very similar for different duct faces but there are significant differences in neutron flux.



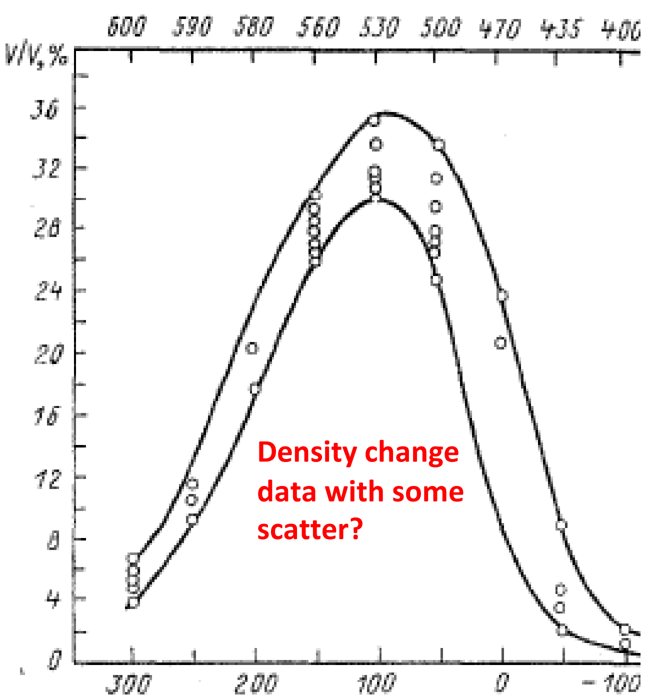


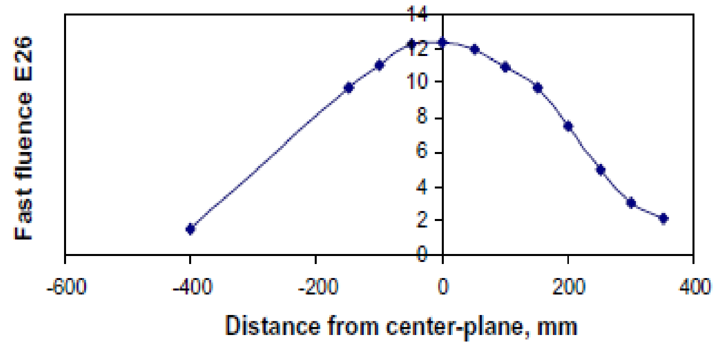


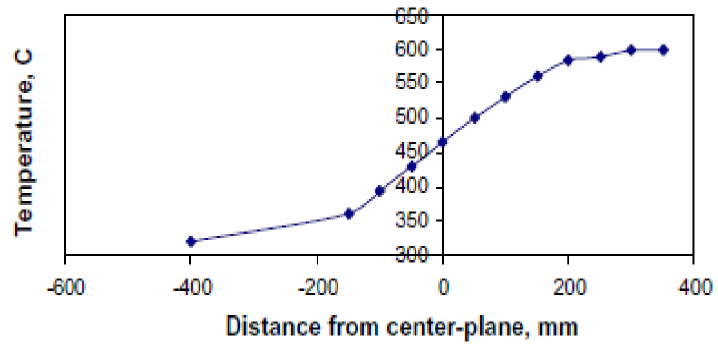
Can we see the breakaway to low swelling rate more clearly by extracting swelling profiles from reactors with much lower inlet temperatures?

BOR-60 has a lower inlet temperature of 320-330°C, depending on the season, allowing an opportunity to probe both swelling rate regimes

Neustroev and Garner, 2008







Breakaway in the 360-390°C interval

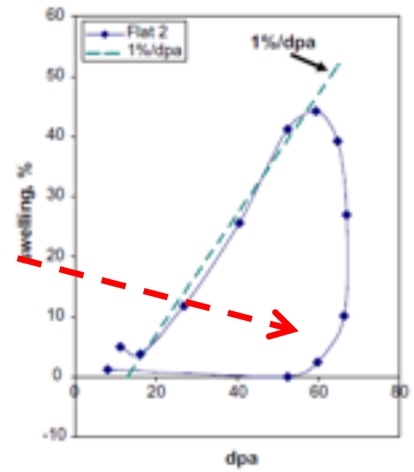


Fig. 5. Swelling plotted vs. dpa level for Flat #2.

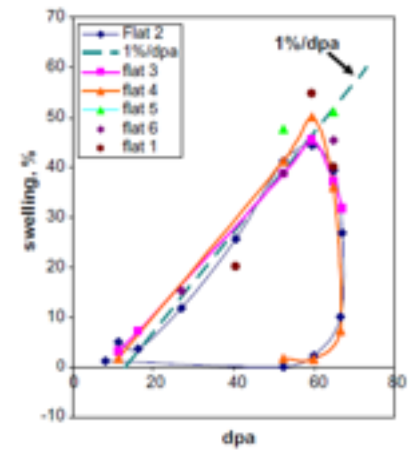
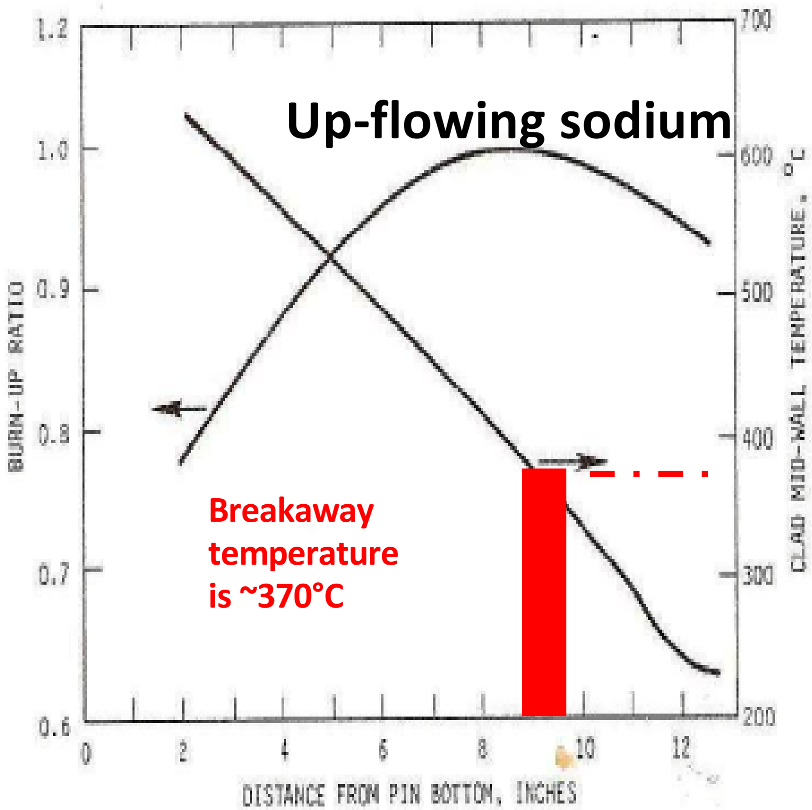
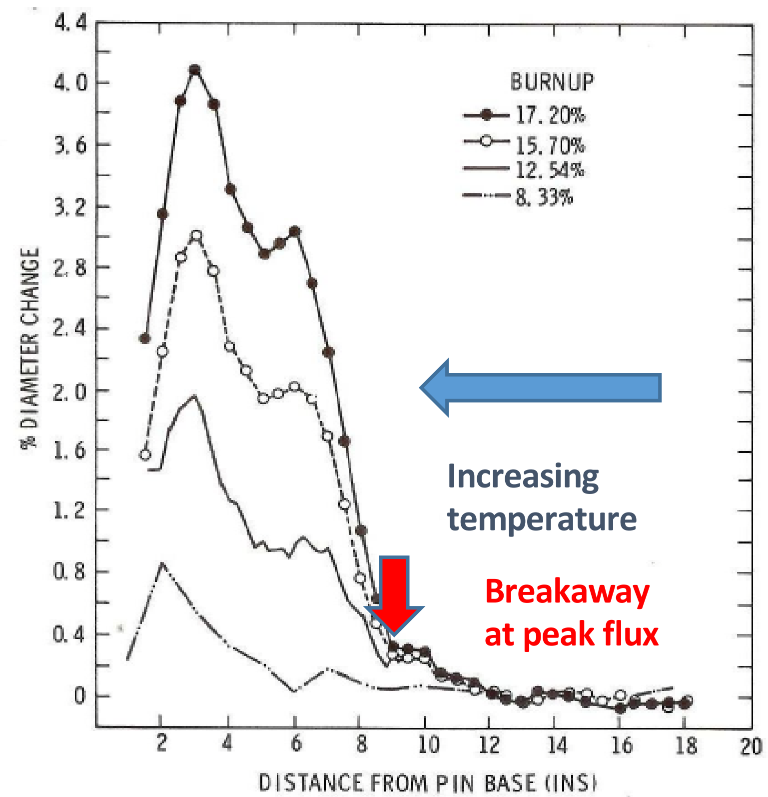


Fig. 6. Swelling plotted vs. dpa level for all flats.

Swelling of cold-worked M316 fuel cladding in DFR with inlet temperature of ~220°C

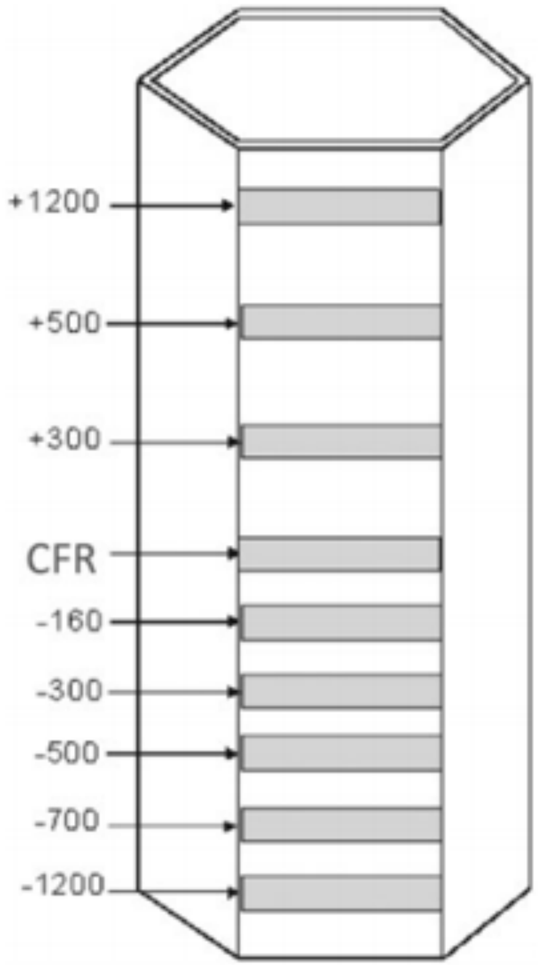
C. Cawthorne, U.S./U.K. Exchange, 1979



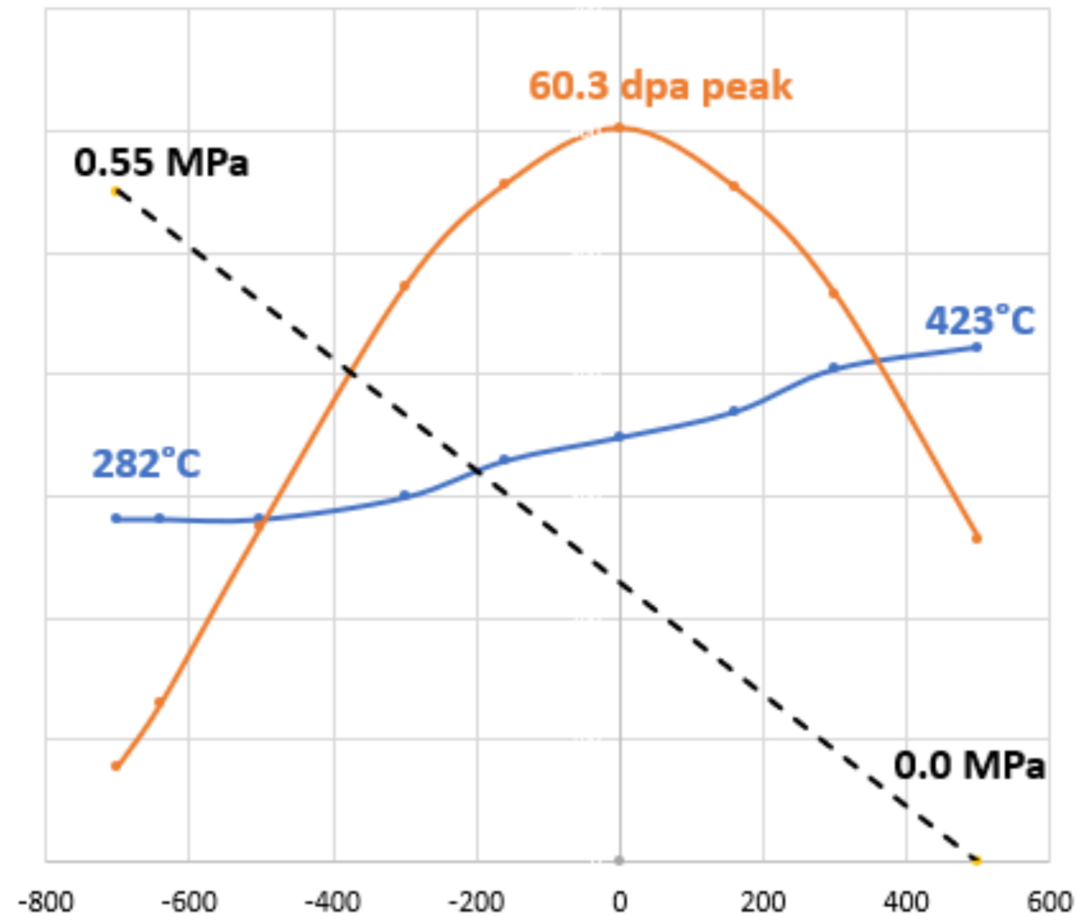


Swelling and irradiation creep of BN-350 duct with 280°C inlet temperature

Maksimkin and Garner , in progress



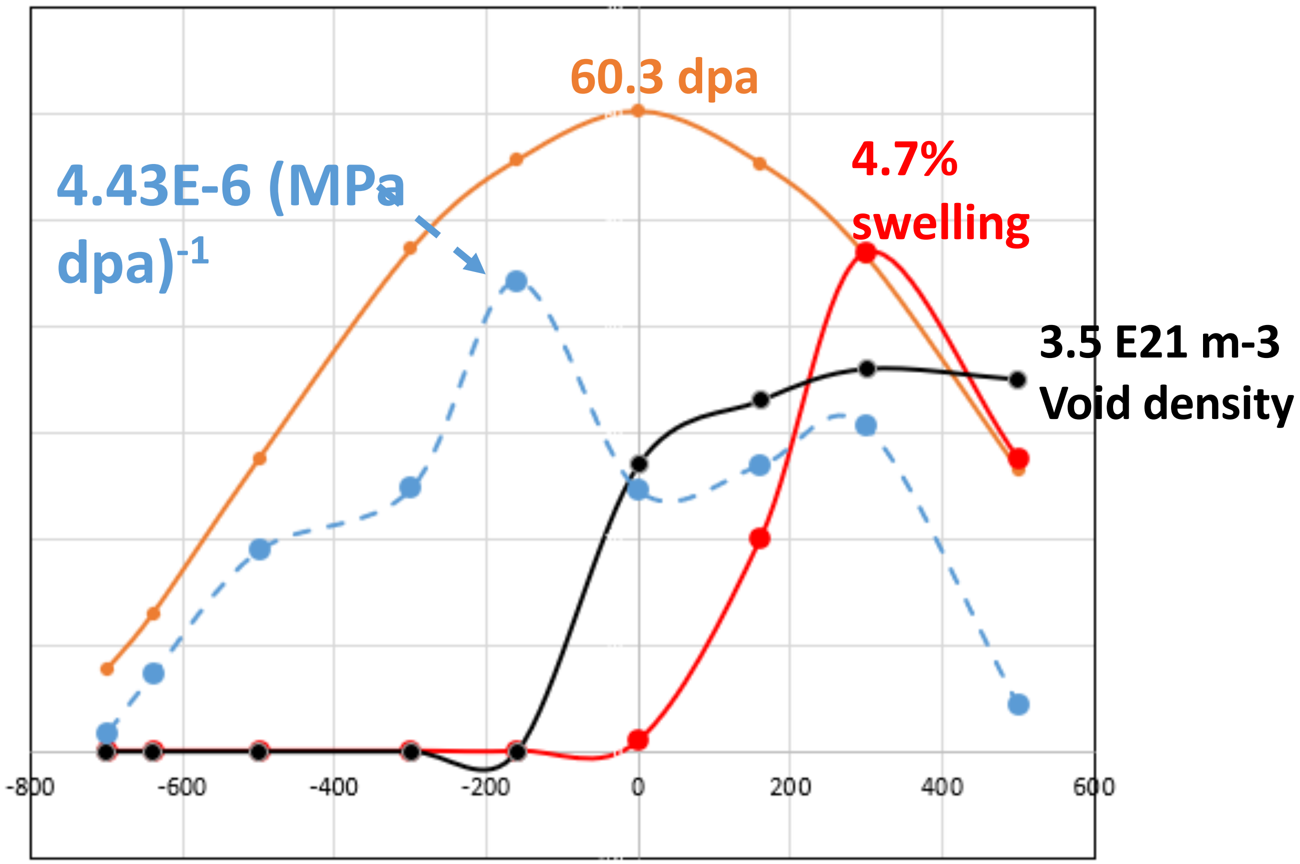
Sample cutting diagram



Distance from core centerline, mm

Swelling and irradiation creep of BN-350 duct with 280°C inlet temperature

Maksimkin and Garner , in progress



Distance from core centerline, mm

No voids below ~330°C

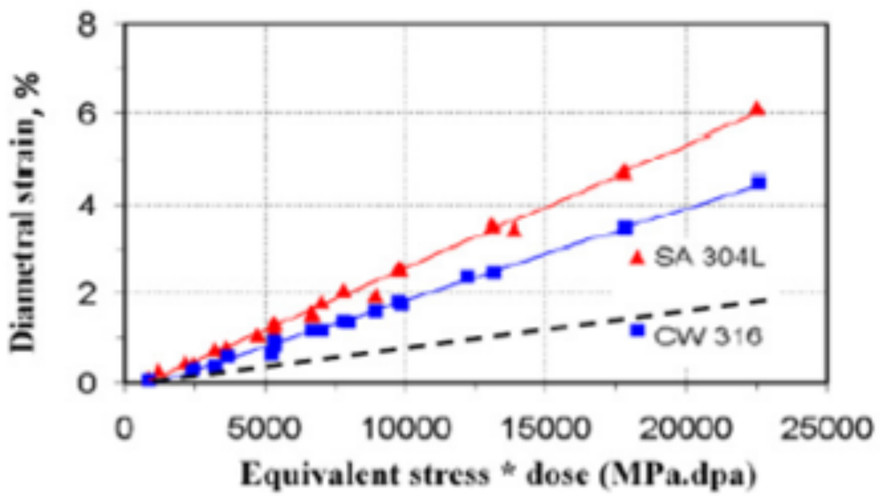
Breakaway transition from low to high swelling rate at ~365°C

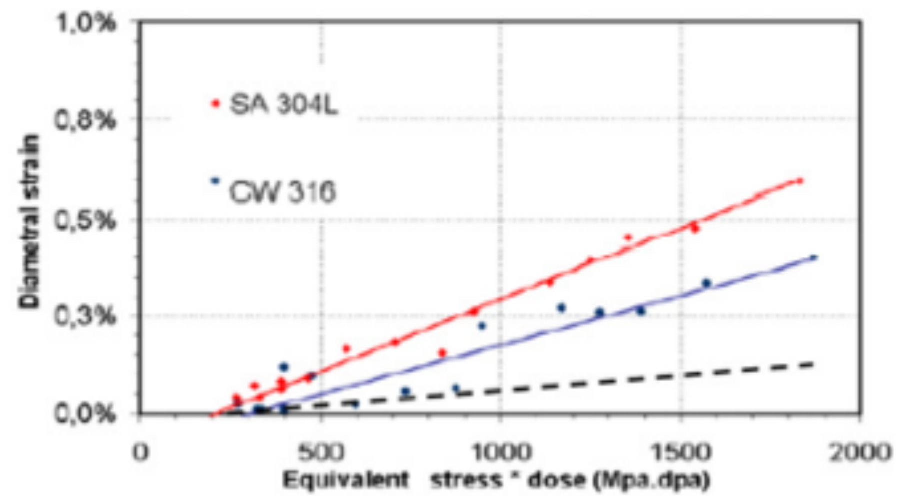
Acceleration of creep with swelling is observed as expected.

Low temperature creep observed in absence of swelling but at much higher rate than usually observed >400 °C.

Irradiation creep has been proposed to be a function of neutron spectra with LWR spectra causing more creep than fast spectra.
Various papers by Foster, Garner, Grossbeck

Actually, it is the lower temperatures in LWRs that lead to a higher creep rate. French experiments below in both fast and light water reactors show increased creep rate in absence of any swelling. Various papers by Dubuisson and Garnier.





Irradiation creep at ~330°C in BOR-60 and OSIRIS reactors, with creep rates 3-4 times larger than typical value (dotted line) observed in EBR-II and FFTF.

Conclusions

- The potential for second order processes to grow to first order importance increases with extended plant life.
- Transmutation issues associated with helium and hydrogen will continue to grow, influencing ductility and weld repair especially.
- Loss of Mn and gain of V appears to be involved in onset of ferrite on grain boundaries, producing greater tendency for corrosion and grain boundary cracking.
- Onset of deformation-induced martensite <u>may</u> also be related to Mn and V changes.
- Void swelling at <u>most</u> of the PWR-relevant temperature range does not proceed to 1%/dpa and probably never will.
- There is a breakaway transition temperature which appears to be flux-dependent in the range 360-380°C.
- Below the transition temperature the irradiation creep rate appears to increase strongly.

ELECTRIC POWER RESEARCH INSTITUTE

Coordination of EPRI Issue Programs' Research Activities with LWRS Materials Pathway Projects

LWRS Materials Research Pathway Stakeholder Engagement Meetings
- Life Beyond 80

Mike Burke, Ph. D.
Technical Executive
EPRI International Materials Research

Emma Wong
Principal Technical Leader
EPRI Innovation Department

November 3, 2020

www.epri.com

Date: Add submission date and/or revision date & #

EPRI's Mission

Advancing ***safe***, ***reliable***, ***affordable,*** and ***environmentally responsible*** electricity for society through global collaboration, thought leadership and science & technology innovation

EPRI's Approach to LTO

- Decades of aging management research
- Collaboration with the U.S. DOE, NRC Research and International partners
- NEI Initiative 03-08
- Technical basis for long term operation based on key parameters – not age
- Plant support

Living Research Programs
Technical reports are updated based upon:

Considerations for Longer-Term Operations

- Technical Basis
  - New knowledge gaps
  - Repair technologies
  - Replacements
  - Aging management programs/practices
- Impacts of LTO
  - Plant Modernization
  - Nuclear Beyond Electricity
  - Low Carbon Resource Initiative
- Market forecast

Are there potential showstoppers?

Bridging the Materials Knowledge Gap for Long Term Operations

- Driven by the need to support of member license extension efforts
- Incorporated LTO metals aging issues into EPRI Issue Management Tables
  - In accordance with NEI 03-08 compliance
  - Based on requirements of GALL/IGALL
  - LTO related gaps specifically identified information needed to support SLR
- Identified primary drivers for MRP, BWRVIP, SGMP & PSCR research
- Established coordination and collaboration for DOE/EPRI/NRC research
- Research products targeted improved assessment of degradation in
  - Reactor Pressure Vessels
  - Reactor Vessel Internals
  - Primary System Piping
  - Steam Generator Systems
- Development of improved assessments, more effective inspection strategies, mitigation and repair and replace strategies
- Direct application of EPRI developed information provides guidance for utility plant life extension submissions
- Continuing development and sharing of information will support SLR and more efficient and cost effective plant aging management methodologies

Underlying Premise and Challenge for the Materials Programs

Primary system materials integrity vital to plant performance and reliability

Reactor components operate in a harsh environment (temperature, stress, radiation, etc.)

Aging of plant system materials is complex and not always fully understood

Routine surveillances can mitigate some of these factors; however, some failures can be expected

<u>Challenge</u>: Find the next material vulnerability and address it before any failures occur

Systematic Approach to Materials Issues

Materials Degradation Matrix
3002013781

MDM

Every component, every material, every potential degradation mechanism

- Mapped to 80 years of operation
- Covering BWR, PWR, CANDU and VVER
- Published 2018

Issue Management Tables
PWR: 3002000634
BWR: 3002000690
CANDU: In Progress

IMTs

Every issue to resolve identified and prioritized

- Covers BWRs and PWR IMTs to be published 2020
- VVER in progress

The strategic approach proactively identifies aging management issues and the research activities needed to resolve them on a timely basis

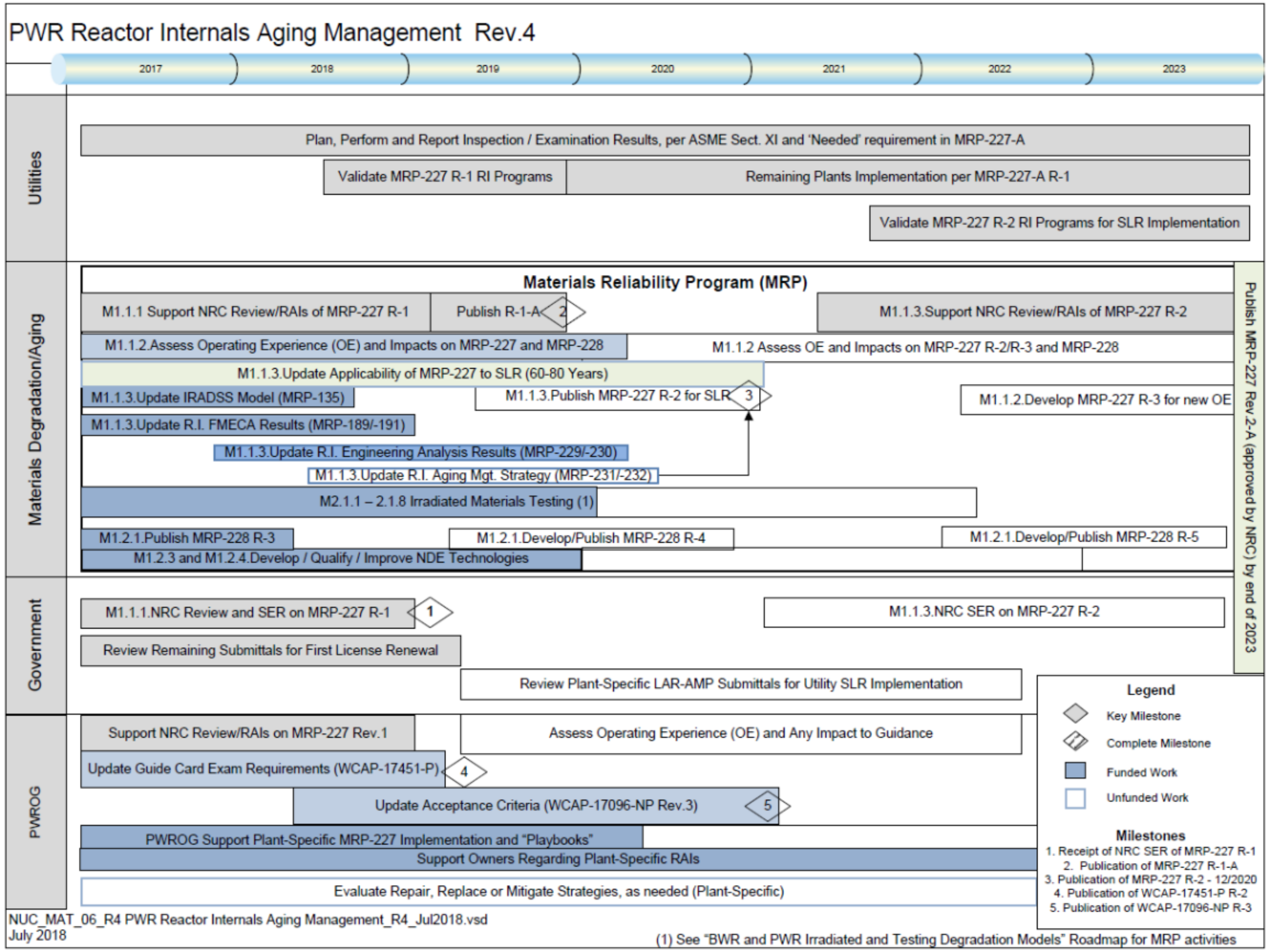
Key Strategic Issues in Metals Aging Management

- Late occurring RPV damage phenomena (PWRs) and flux effects on embrittlement correlations (BWRs)
- Irradiation effects on reactor vessel internals (PWRs)
- SCC Prediction and Modeling - including concerns regarding SCC initiation late in life and factors of improvement associated with high Cr Ni-base alloys (Alloy 600/52/152)
- Identifying key factors influencing IASCC occurrence and modeling / prediction capabilities
- Effect of extended operations on high-strength core periphery components – e.g., Alloys A-286, X-750, XM-19 (*irradiated properties including fracture toughness, tensile strength, SCC growth rates*)
- Effect of environment on fatigue life (*specifically, the capability to accurately estimate usage – not simply managing analytical margins*)

Issues are reflected in the Issues Management Tables

MRP-227 Program addresses LTO issues for PWRs

- MRP-227 document is "Pressurized Water Internals Inspection and Evaluation Guidelines" – Current Revision, R1, covers extended operation to 60 years
- Original process for first license extension >40 years
- Documentation extended and updated to take into account 60-80 years
- Recent updates take into account longer extensions
- Background documents updated to contain updated OE and research results where applicable
- "MRP-227 Process" contains background documentation – update to cover SLR
  - MRP-211 Aged Material Properties and Basis Data – Rev 1 published 2017
  - MRP-156 – IMT Consequences of Failure – Rev 0 Published 2005
  - MRP-175 Screening and Threshold Values – Rev 1 Published 2017
  - MRP 189-191 FMECA Results – Rev 2 Published 2018
  - MRP 229-230 Engineering Analyses – Rev 2 Published 2019
  - MRP 231 & 232 Aging Management Strategy – Rev 1 Published 2012
  - MRP-227 – Rev 2 to be published in 2020



BWRVIP follows analogous process for aging management of BWRs
BWRVIP-315, BWRVIP-05, BWRVIP-316

LTO Results for Managing Metals Aging

- Systematic identification of high priority metals aging issues for extended plant operations driven by original LTO program needs
- Development of comprehensive IMT documentation in IMT updates
  - Originally call out LTO related items for >40years, 60-80 years
  - Post LTO program we are identifying "open ended" aging management issues
- Development of improved information and "gap filling" programs including

  Reactor Pressure Vessels → *Coordinated and Supplementary Surveillance Capsule Programs*
  Reactor Vessel Internals → MRP and BWRVIP documents to provided guidance for Inspection and Evaluation of Reactor Vessel Internals for Life Extensions, MRP-227 process and background information process to support
  Primary System Piping → Improved information and methodologies for addressing environmental fatigue interactions
  Steam Generator Systems → Development of crack initiation and crack growth rates in key Ni-base alloys
- Coordination of research programs and sharing of results with DoE and NRC – Development of Coordinated Research Roadmaps

  e.g. EPRI/DoE ORNL development of advanced welding cell for irradiated materials

These LTO initiated projects are continuing under MRP, BWRVIP and SGMP sponsorship in support of plant aging management

Together…Shaping the Future of Electricity

Life Beyond 80: Concrete Aging

Yann Le Pape, Elena Tajuelo, Yujie Li, Amani Cheniour, David Arregui, Tom Rosseel

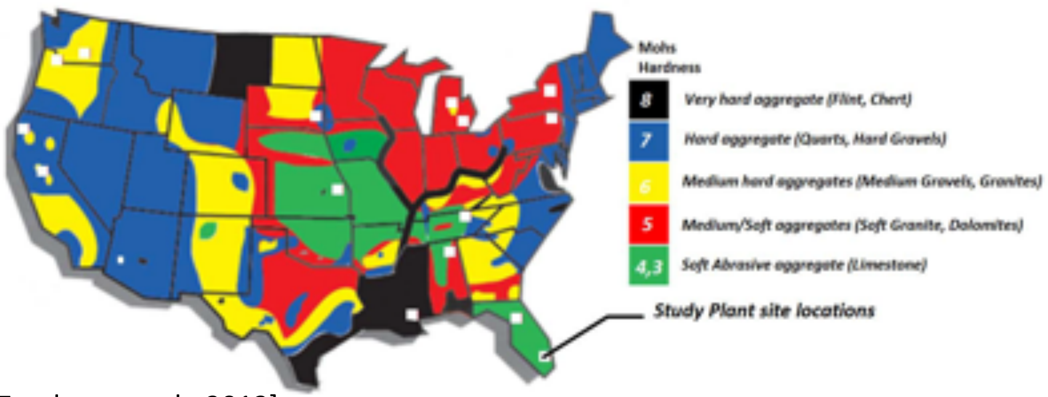
Oak Ridge National Laboratory

LWRS
LIGHT WATER
REACTOR
SUSTAINABILITY

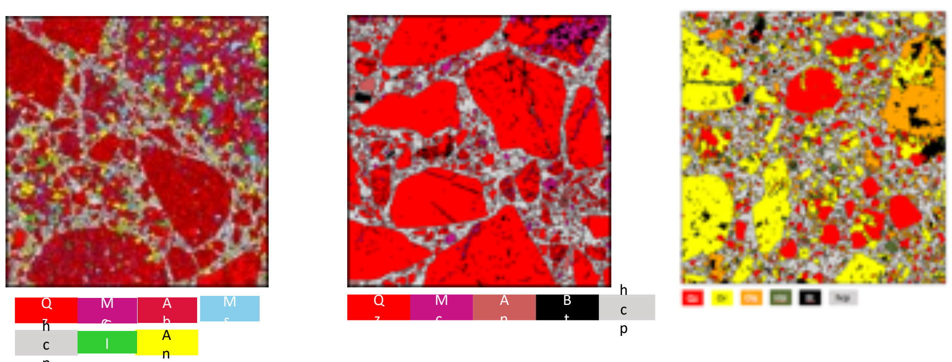
LWRS Materials Research Pathway
Concrete Stakeholder Engagement Virtual Meeting
November 5, 2020

Quick reminder: Not all concrete(s) are equal and they can be found everywhere!

US Geology and NPP locations

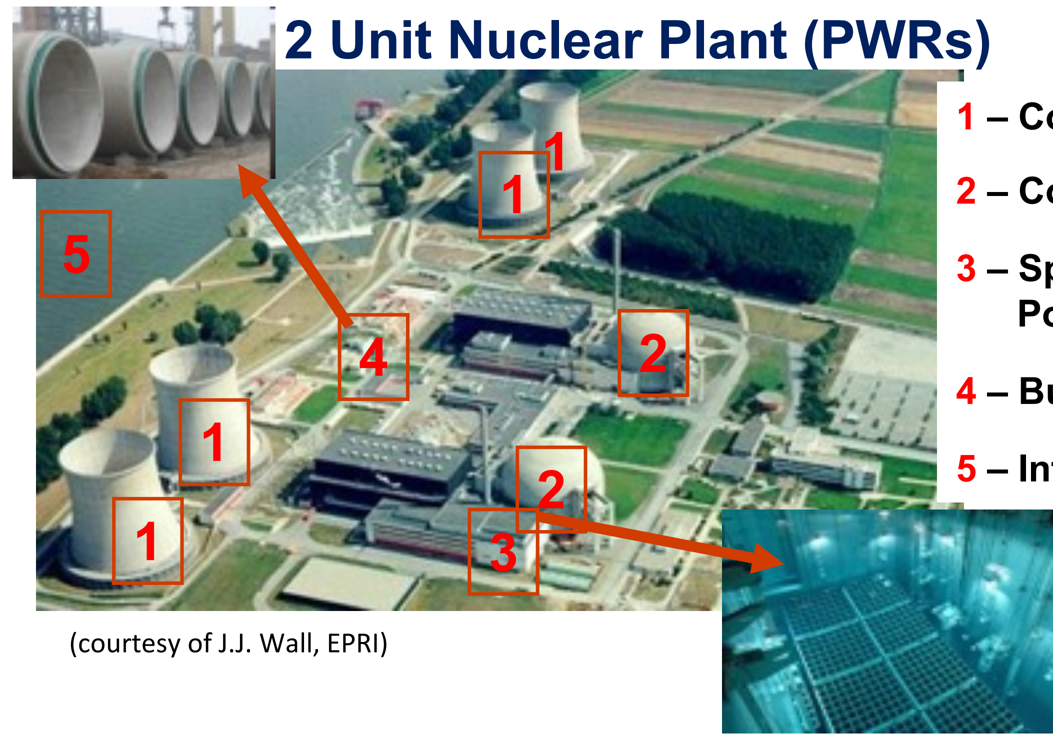


[Esselman et al., 2013]



Examples of mineral phase maps for varied concretes tested at ORNL

2 Unit Nuclear Plant (PWRs)



1 – Cooling Towers

2 – Containments

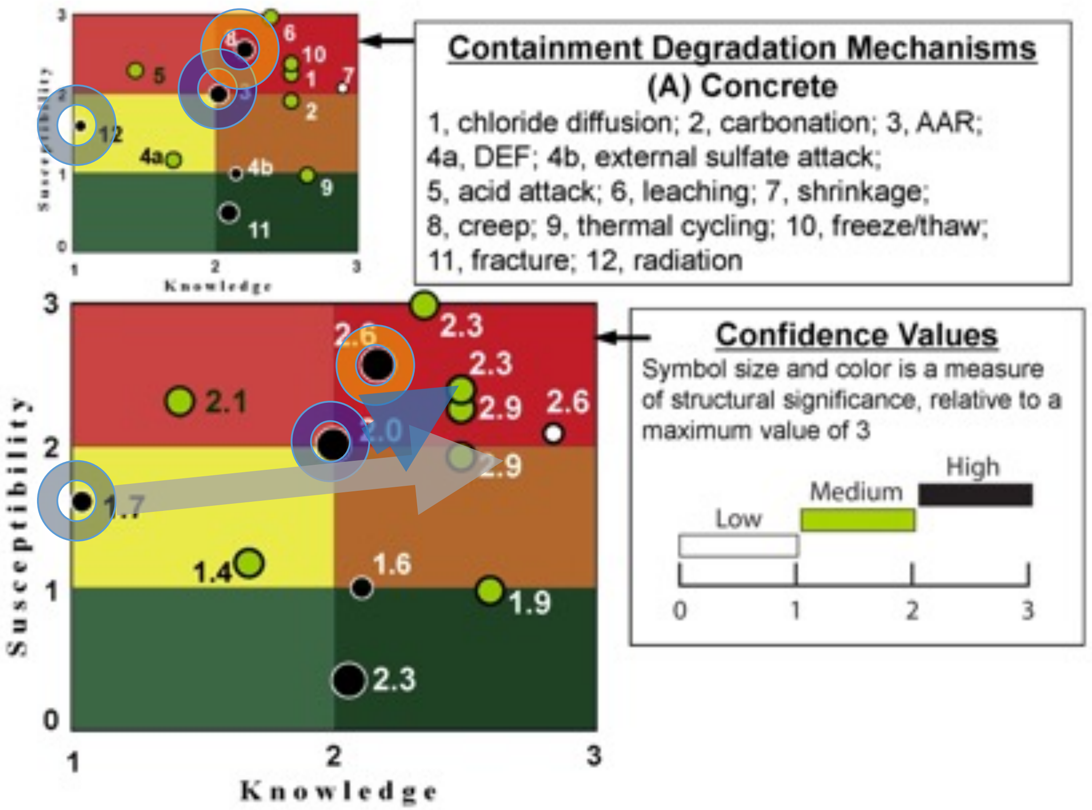
3 – Spent Fuel Pools/Transfer Canal

4 – Buried Pipe

5 – Intake Structure

(courtesy of J.J. Wall, EPRI)

Motivations Determined 8 Years Ago



Irradiation

Alkali-silica reaction (ASR)

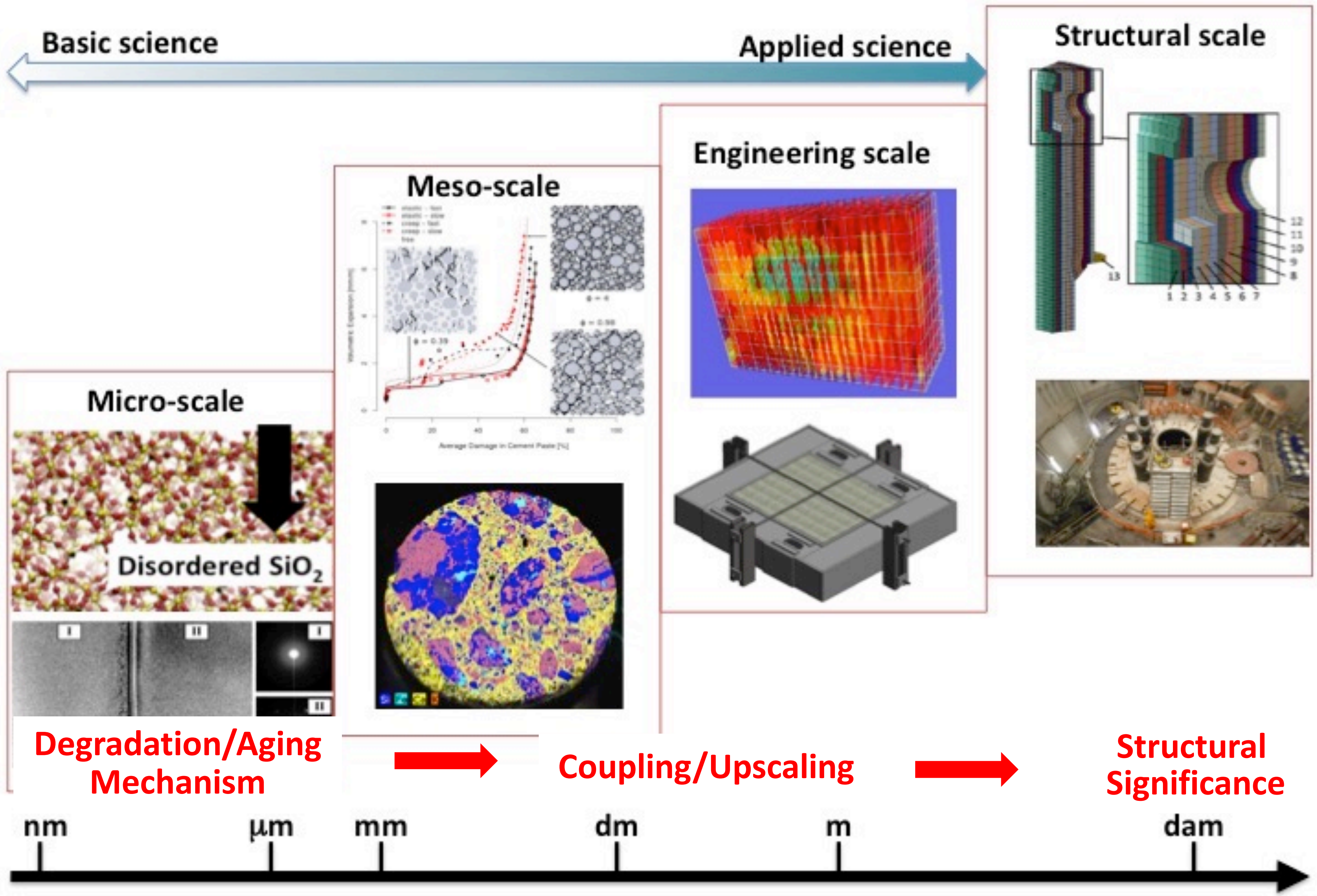
Creep / creep-fracture

Excerpt from Expended Materials Degradation Analysis report (2014):

'Irradiation for "Containments-Concrete Component" emerged as the most important degradation mechanism, mainly driven by the fact that insufficient data is available to improve the level of knowledge about the effects of irradiation on concrete mechanical properties.'

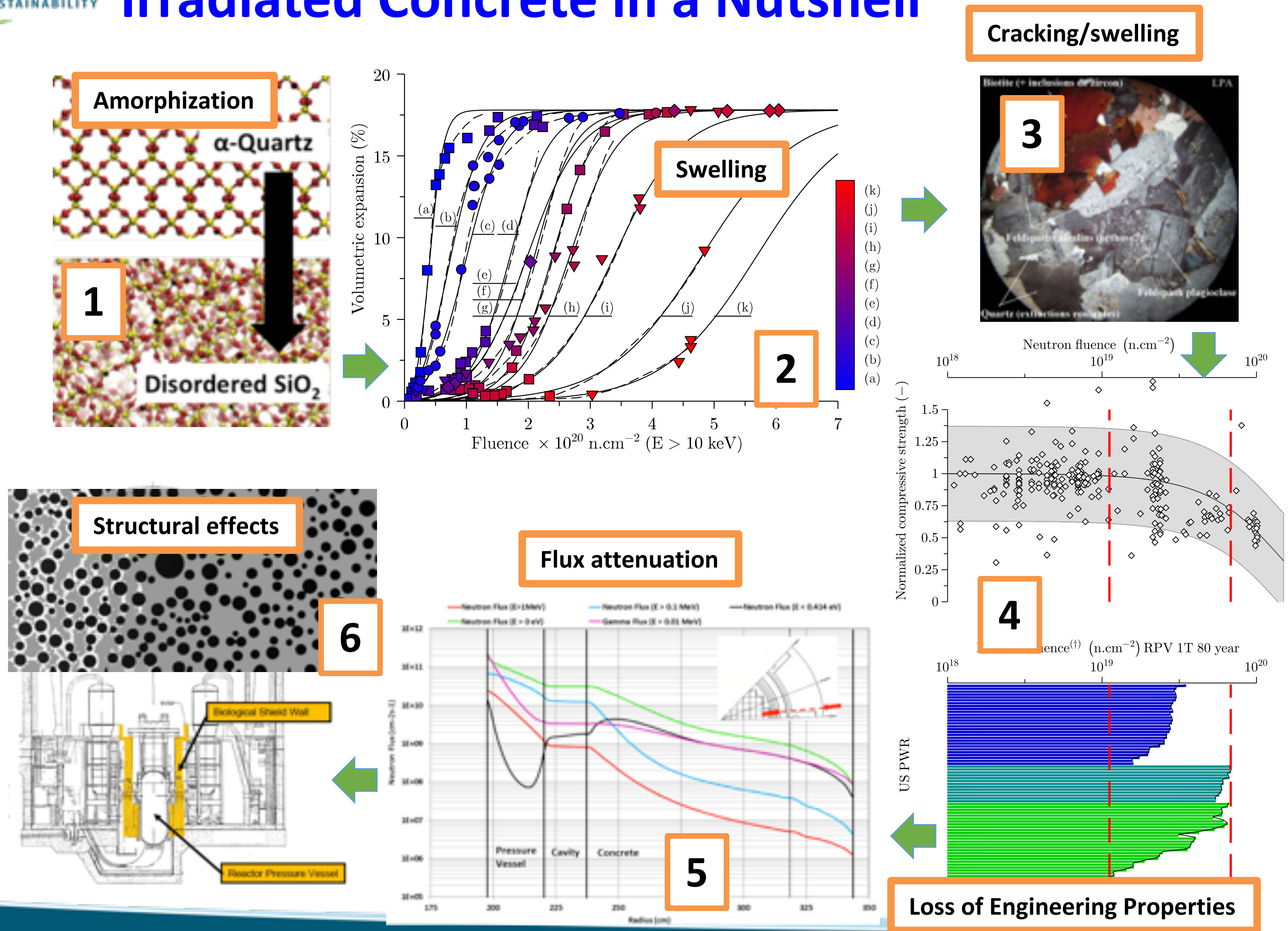
'Though ASR is well documented by the operating experience (for bridges and dams in particular) and scientific literature, its high ranking in the EMDA analysis describes the need to assess its potential consequences on the structural integrity of the containment.'

Research significance: **Second license renewal** (60+ years) of U.S. nuclear fleet



LWRS
Basic science
Applied science
Structural scale
Engineering scale
Meso-scale
Micro-scale
Disordered SiO_2
Degradation/Aging Mechanism
Coupling/Upscaling
Structural Significance
nm
µm
mm
dm
m
dam

Irradiated Concrete in a Nutshell

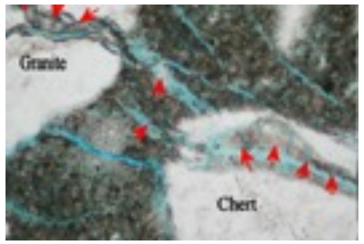


Alkali-Silica Reaction in a Nutshell

www.chaneyenterprises.com

Cement **alkali**

Reacts with



www.cmc-concrete.com

Aggregate reactive **silica**

And absorbs

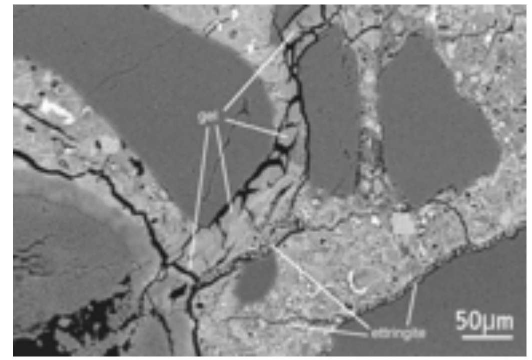
www.borealwater.com

Water
>~70% moisture content

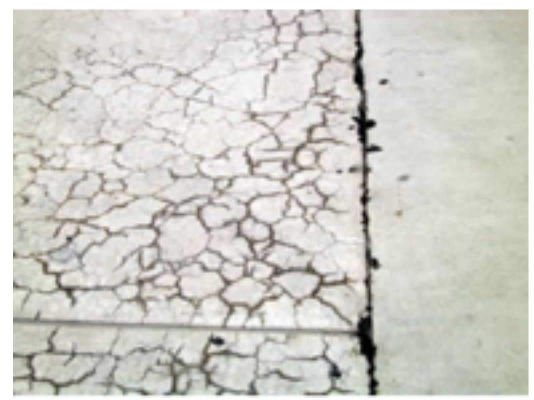
www.fhwa.dot.gov

Thermal activation
(accelerated test 100ºF)

Expansive gel resulting from the **alkali-silica reaction** and **micro-cracking**



Macro-cracking / Swelling



LWRS LIGHT WATER REACTOR SUSTAINABILITY

Corrosion of Embedded Steel in Concrete in a nutshell

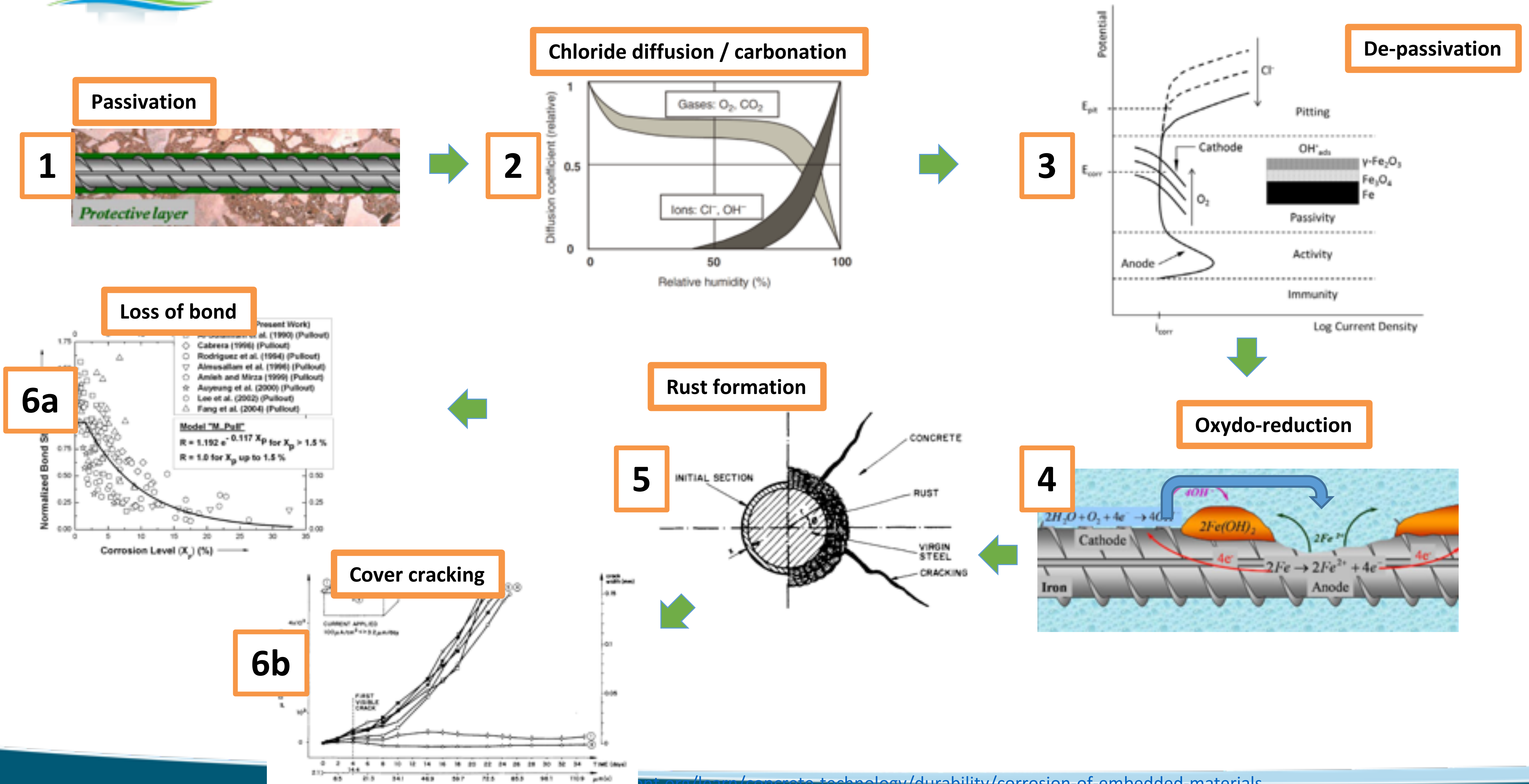
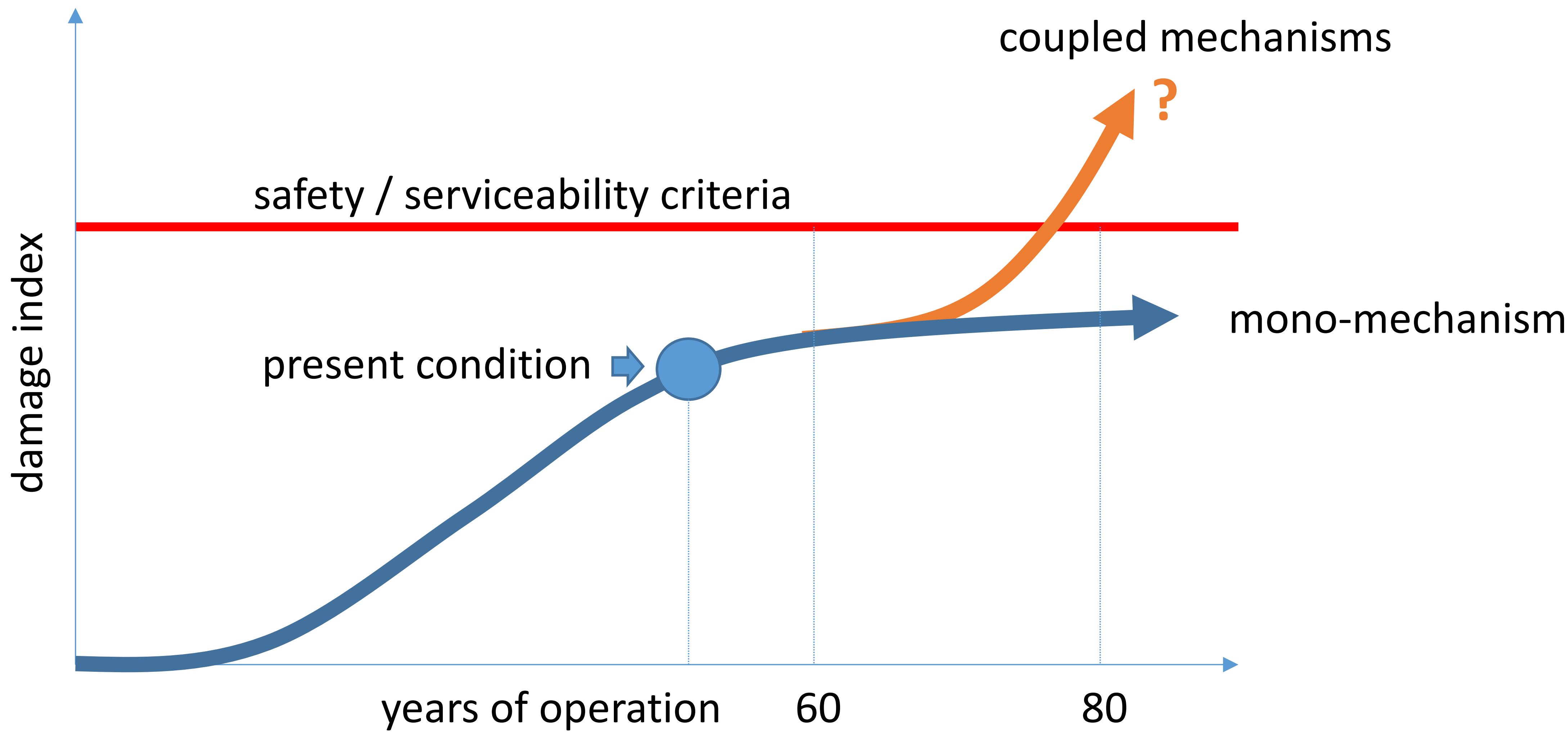


Figure from: https://www.cement.org/learn/concrete-technology/durability/corrosion-of-embedded-materials

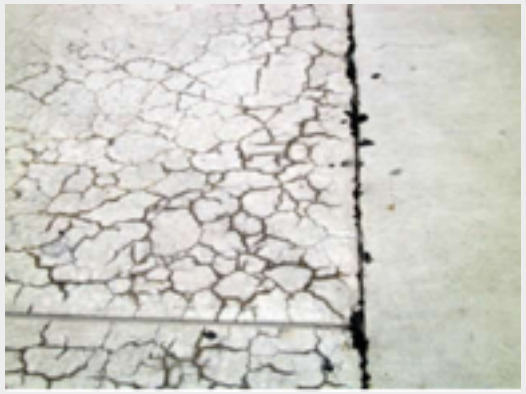
Kinetics and Synergistic Effects



LWRS LIGHT WATER REACTOR SUSTAINABILITY

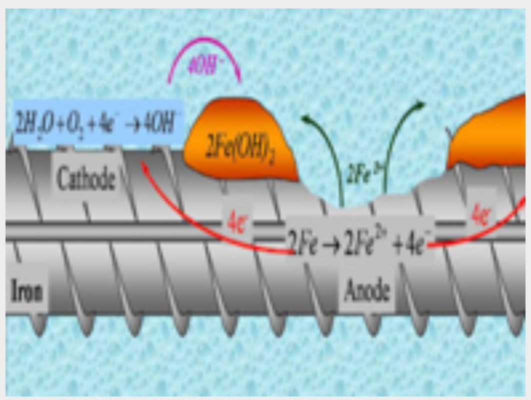
Examples of Possible Synergies

Alkali-silica reaction



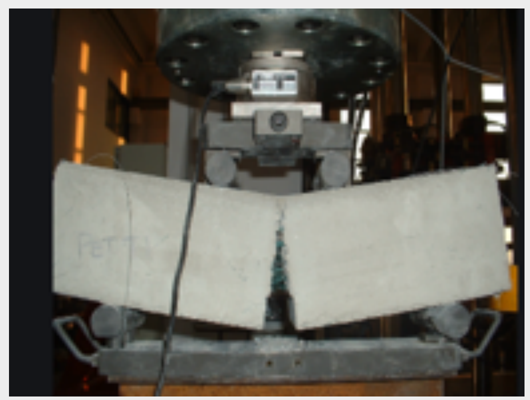
?

Corrosion



Creep

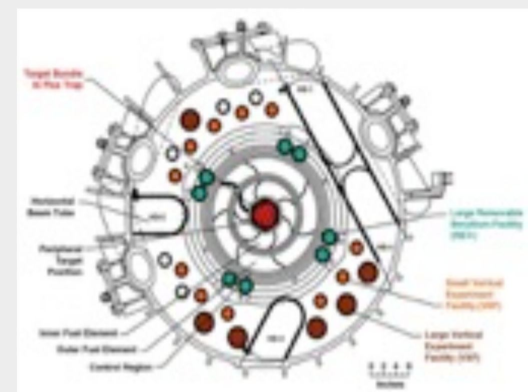
Fracture



Alkali-silica reaction

?

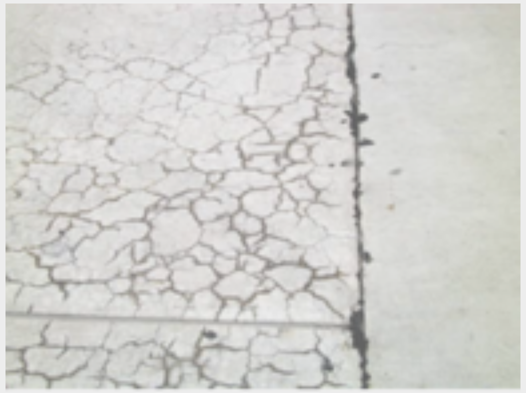
Irradiation



Fukushima

Evidences of Synergies

Alkali-silica reaction



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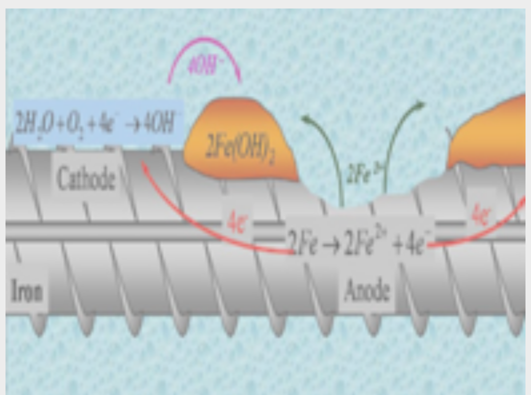
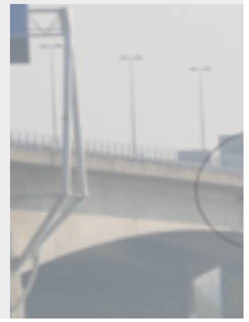
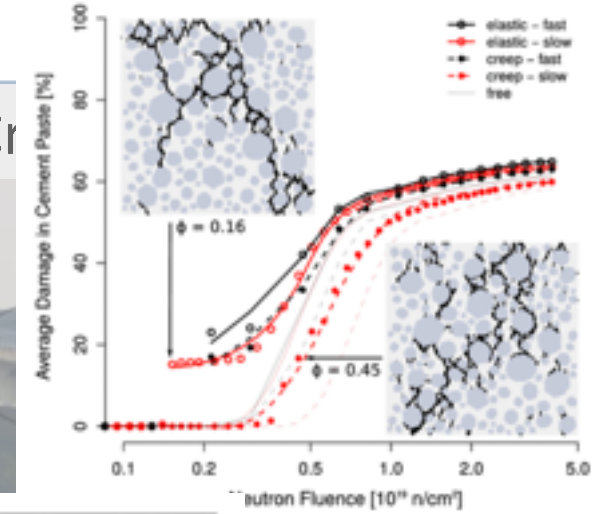
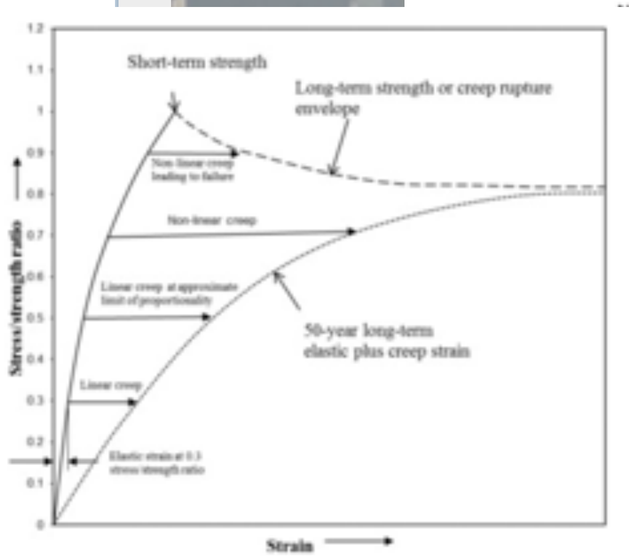


Corrosion

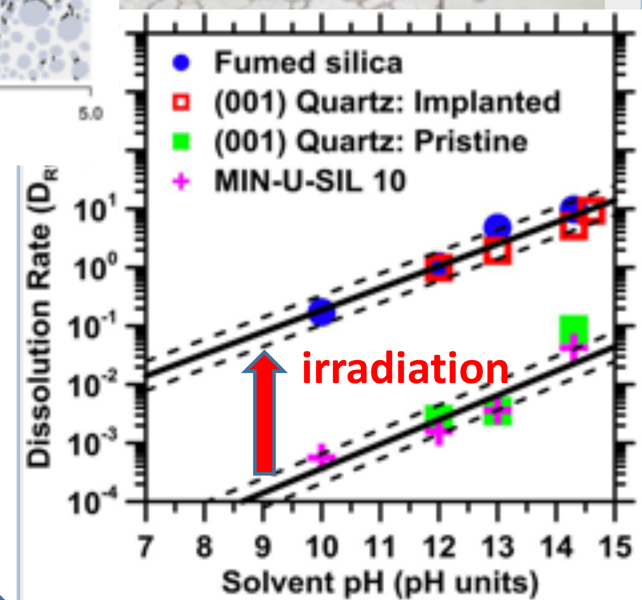
Cr







lkali-silica reaction

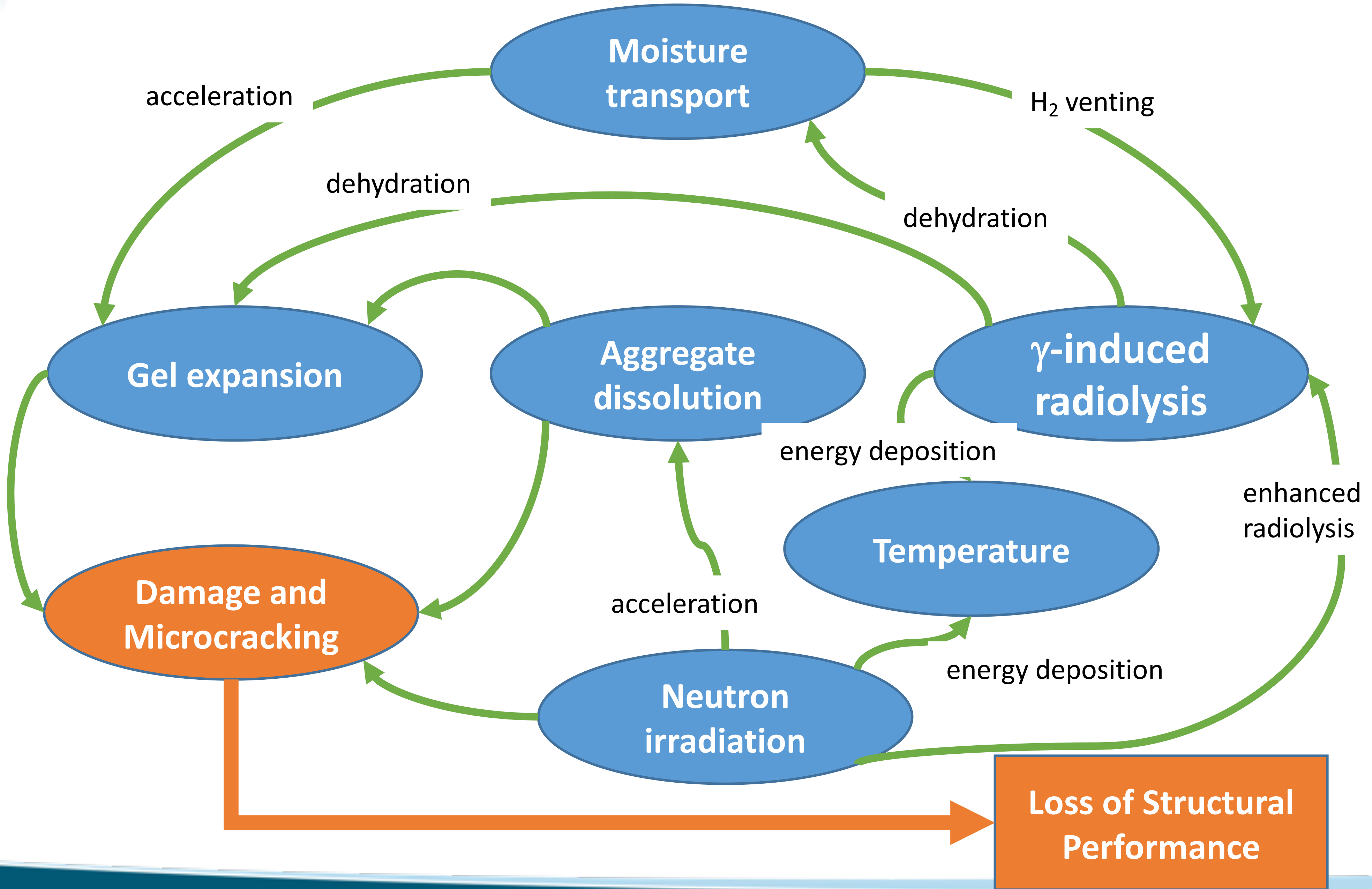


Fracture

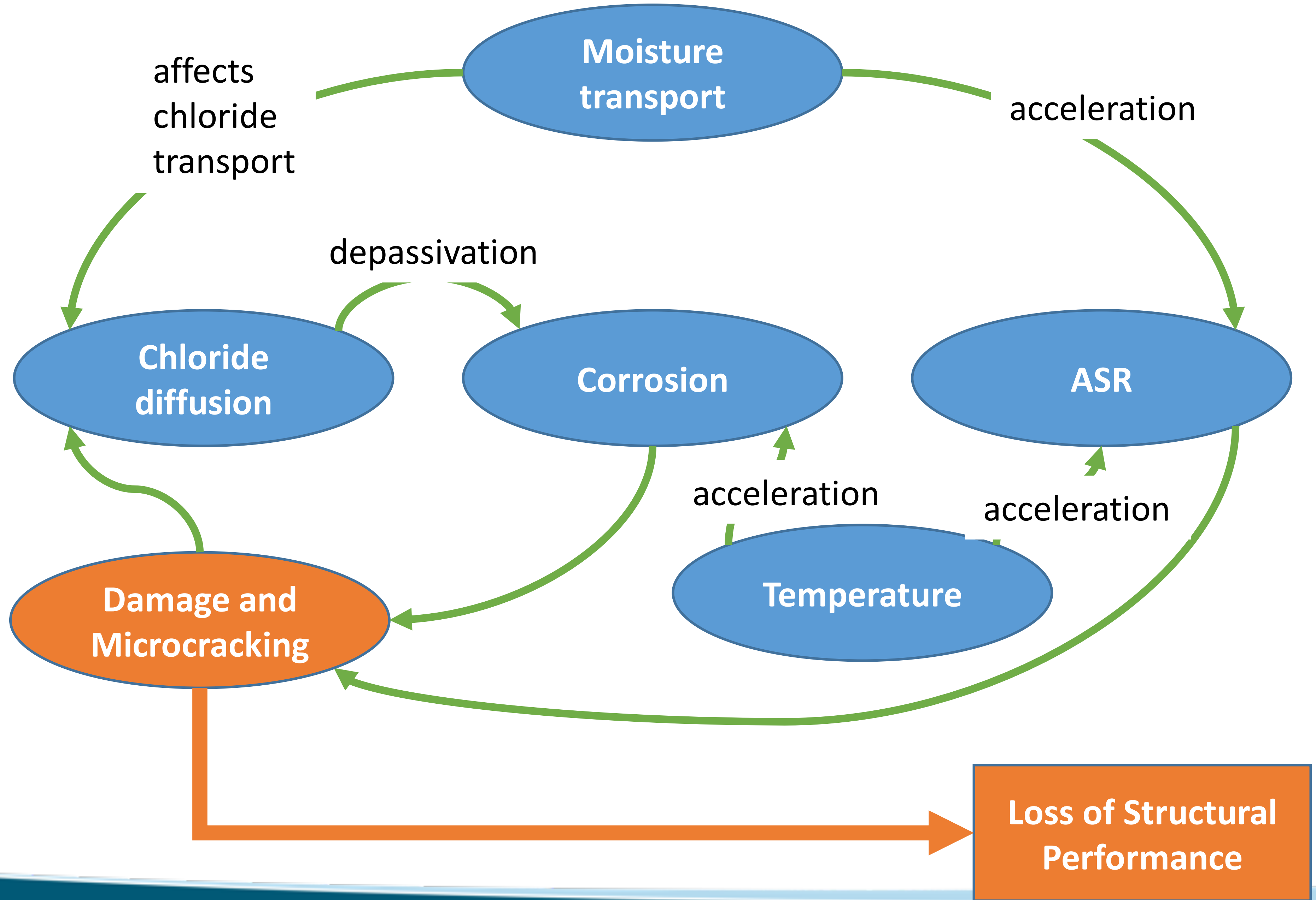
Irradiation

Fukushima

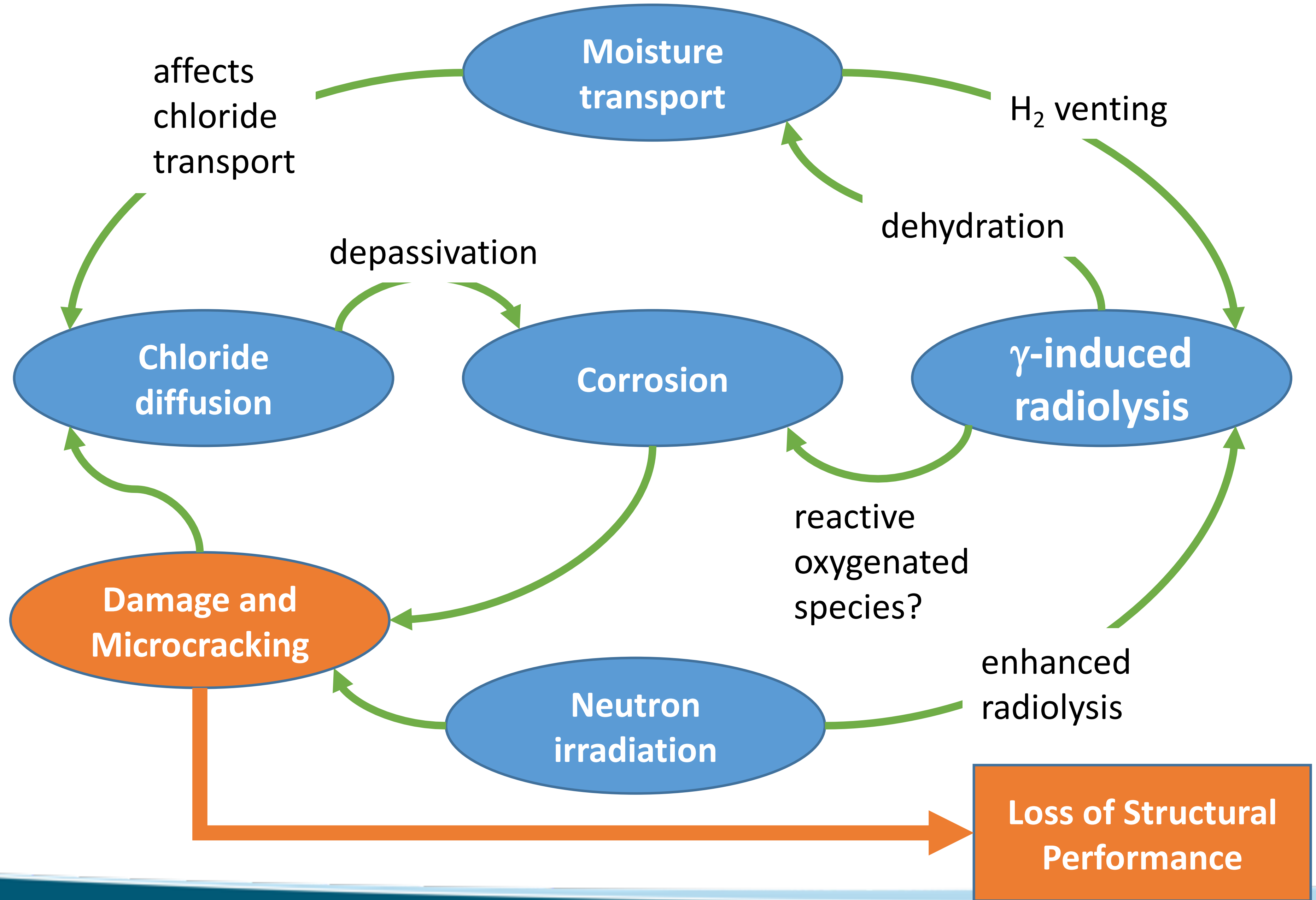
Possible Interactions between Alkali-Silica Reaction and Irradiation



Interactions between Corrosion and Alkali-Silica Reaction



Possible Interactions between Corrosion and Irradiation (Hypothesis)



Conclusions and Perspectives

The LWRS program advanced considerably the understanding, characterization, modeling of concrete subjected to irradiation and alkali-silica reaction

Some knowledge gaps still exists:
Irradiation: rate effects, neutronic effects on creep, bond strength properties between concrete and embedded steel
ASR: role of the aggregates' mineralogy on ASR kinetics and damage development

Synergies between irradiation, ASR, corrosion, creep and damage still largely unknown for an assessment of operation beyond 80 years

Backup slides

Mechanisms of Corrosion of Embedded Steel in Concrete (Cont'd)

Change of alkalinity-induced de-passivation:

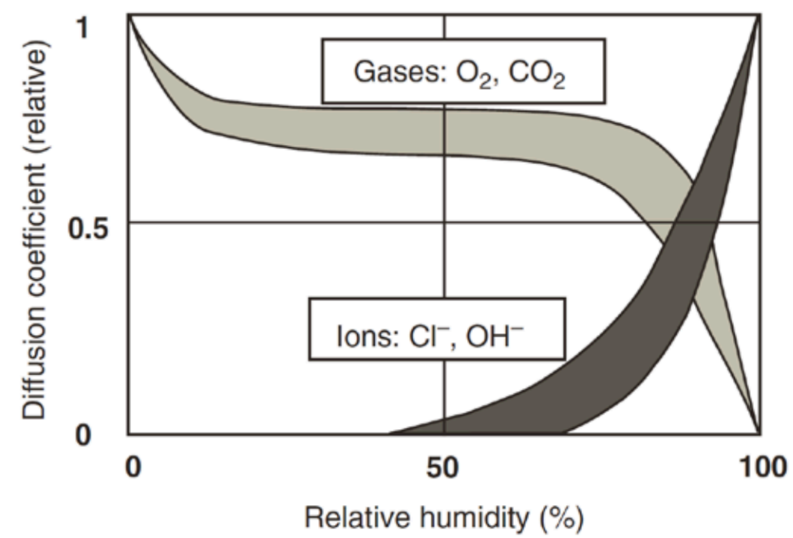
1. **Carbonation**: slow CO_2 diffusion in atmospheric conditions
2. **Chloride ingress** (marine environment, de-icing salt or use of brackish/retreated water for cooling): usually higher rate than carbonation

Chloride penetration is mainly governed by Fick's diffusion:

$$\frac{\partial C}{\partial t} = D\frac{\partial^2 C}{\partial x^2}$$ with C the chloride concentration

And D the chloride diffusion coefficient function of:

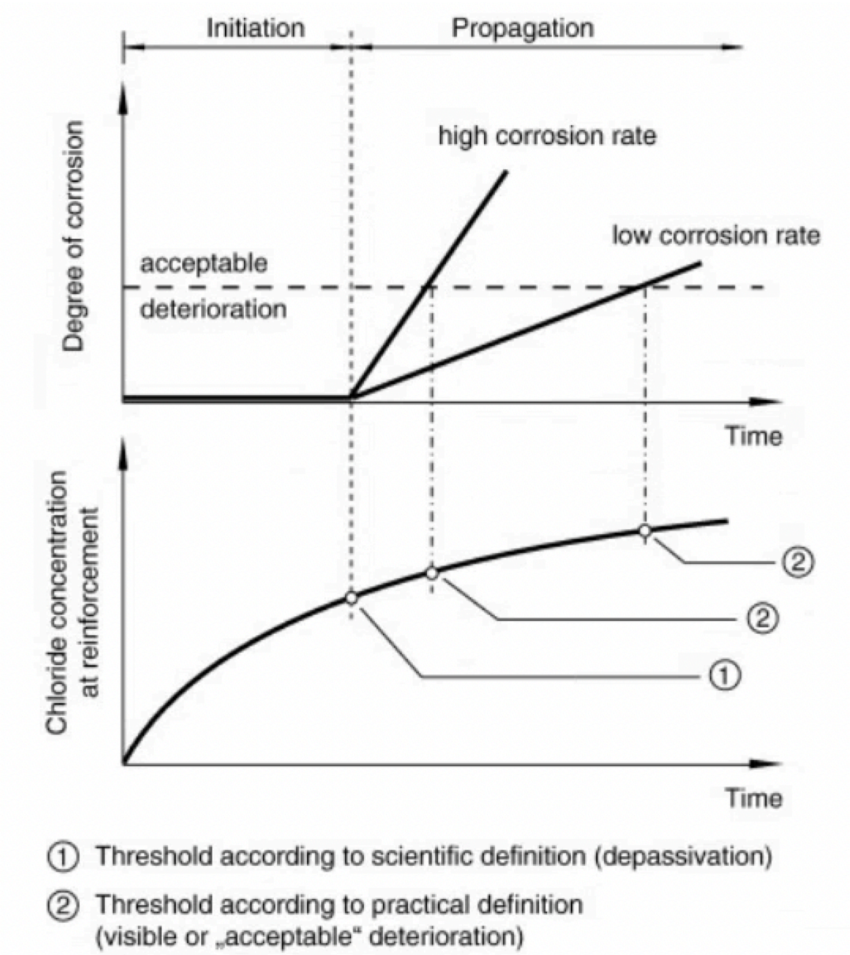
(i) Time
(ii) Space
(iii) Moisture content
(iv) Chloride binding during diffusion (Friedel's salt)
(v) Damage in concrete cover



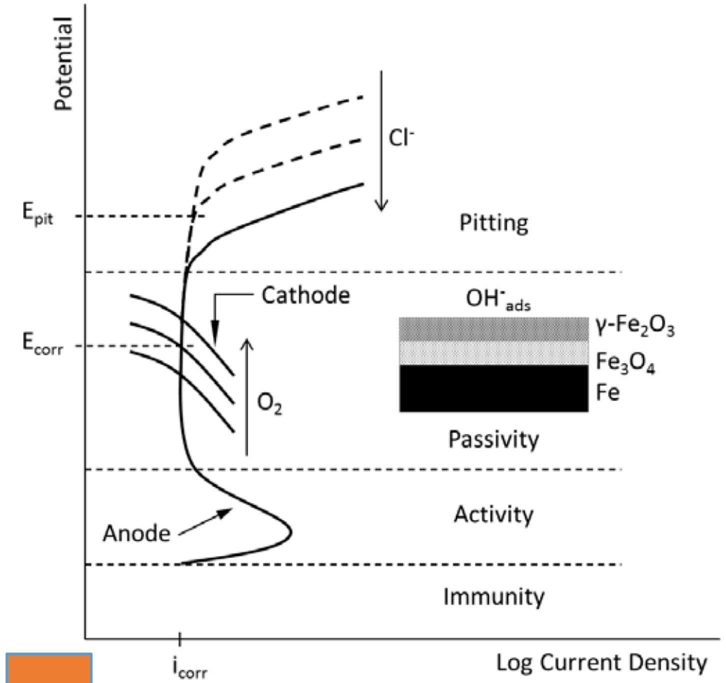
Hunkeler F (2005) Corrosion in reinforced concrete structures, edited by Hans Böhni, Woodhead Publishing, Cambridge, England.

Mechanisms of Corrosion of Embedded Steel in Concrete (Cont'd)

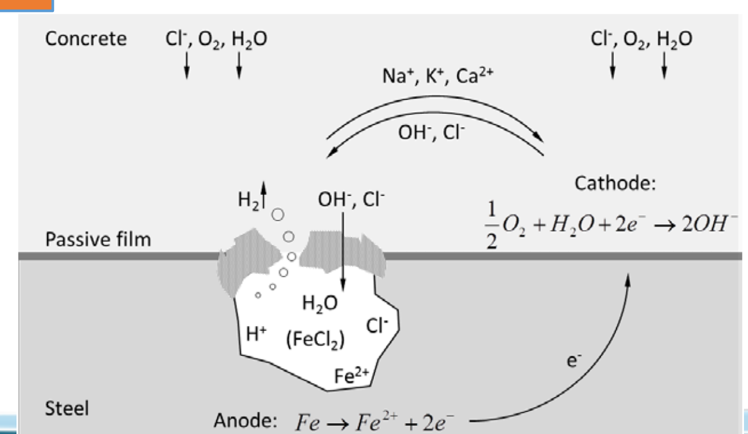
1 Chloride diffusion



2 Local potential modification
(break through passivation layer)



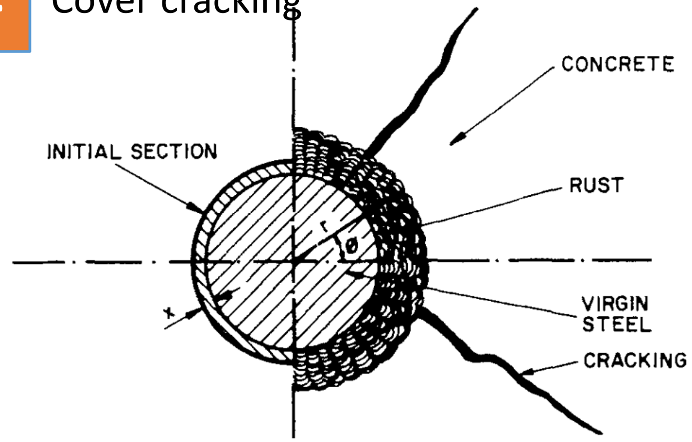
3 Corrosion pitting

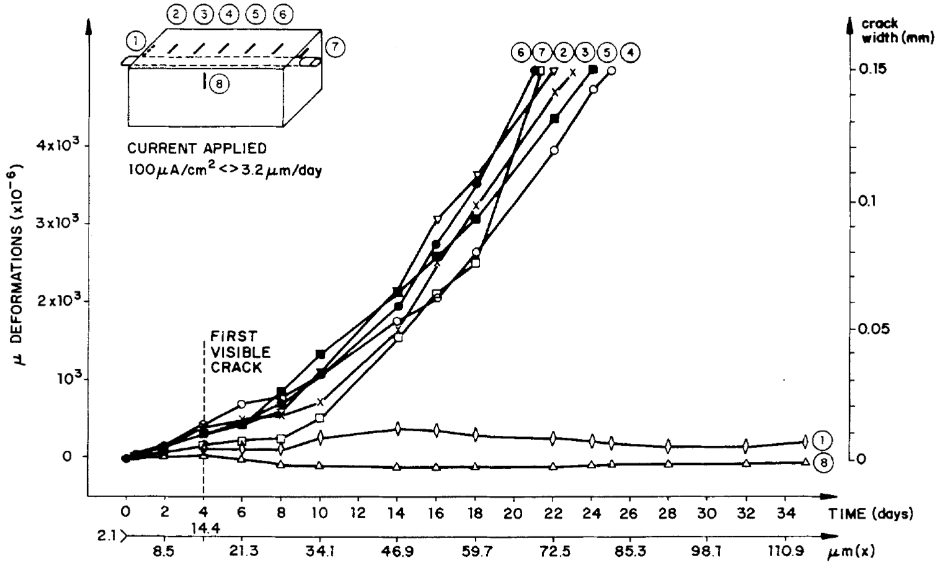


1. Angsta, U., et al. *CCR*, 39(12):1122-1138, 2009
2. MacDougall B and Graham MJ (2002) Growth and stability of Passive films, in Corrosion Mechanisms in Theory and Practice, Marcel Dekker Inc., NY, USA.
3. N. Silva, Ph.D. Dissertation Chalmers University of Technology, 2013.

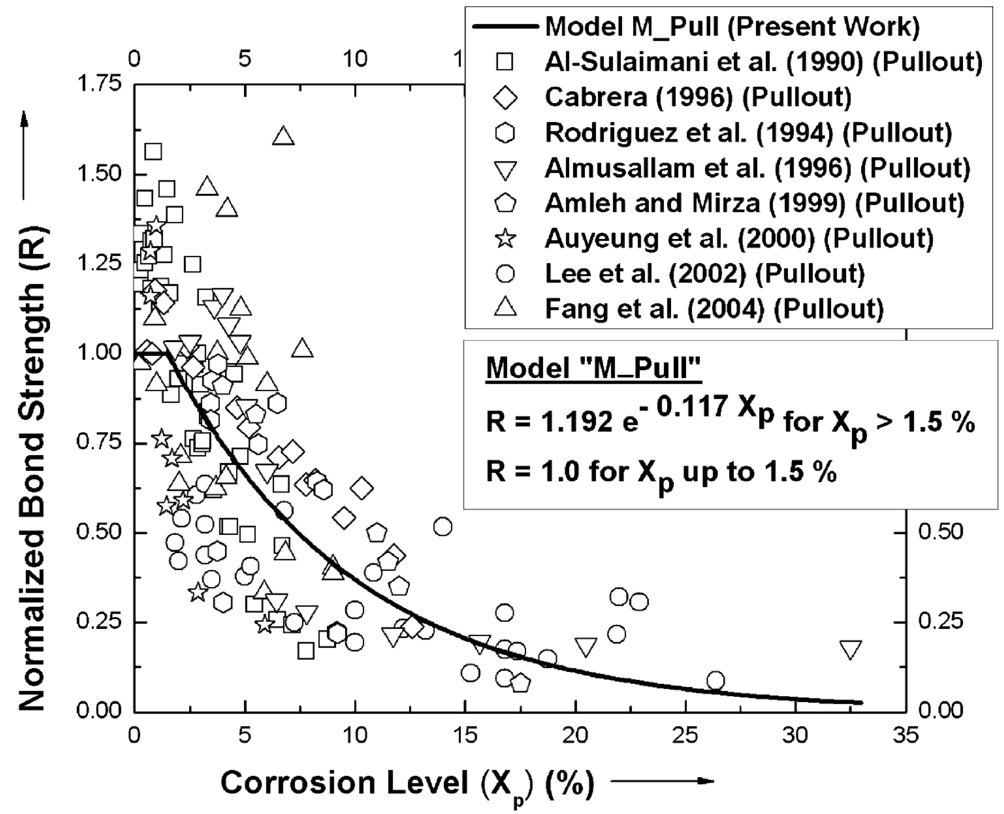
Effects of Corrosion of Embedded Steel in Concrete

4 Cover cracking





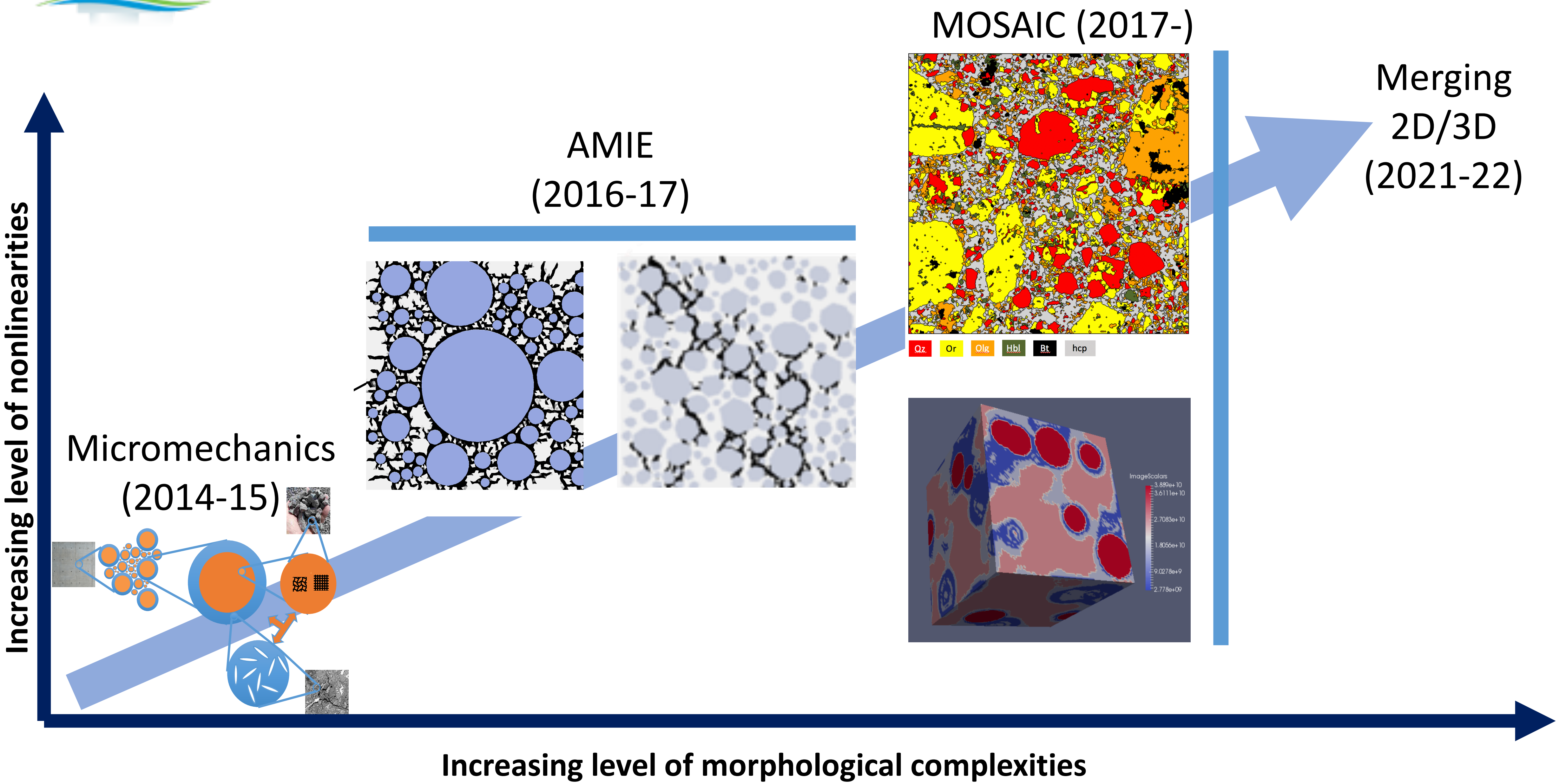
5 Loss of bond strength



Bhargava, K. et al. Corrosion-induced bond strength degradation of reinforced concrete - Analytical and empirical models *Nuclear Engineering and Design*, **2007**, *237*, 1140-57

Andrade, C.; Alonso, C. and Molina, F. Cover cracking as a function of bar corrosion: Part I -- Experimental test *Materials and Structures*, **1993**, *26*, 453-464

Advanced Modeling Strategies



Reliable Use of Old Cables in Extended Operation

Leo Fifield

Materials Research Pathway Life Beyond Eighty
Stakeholder Engagement Meeting
05 November 2020

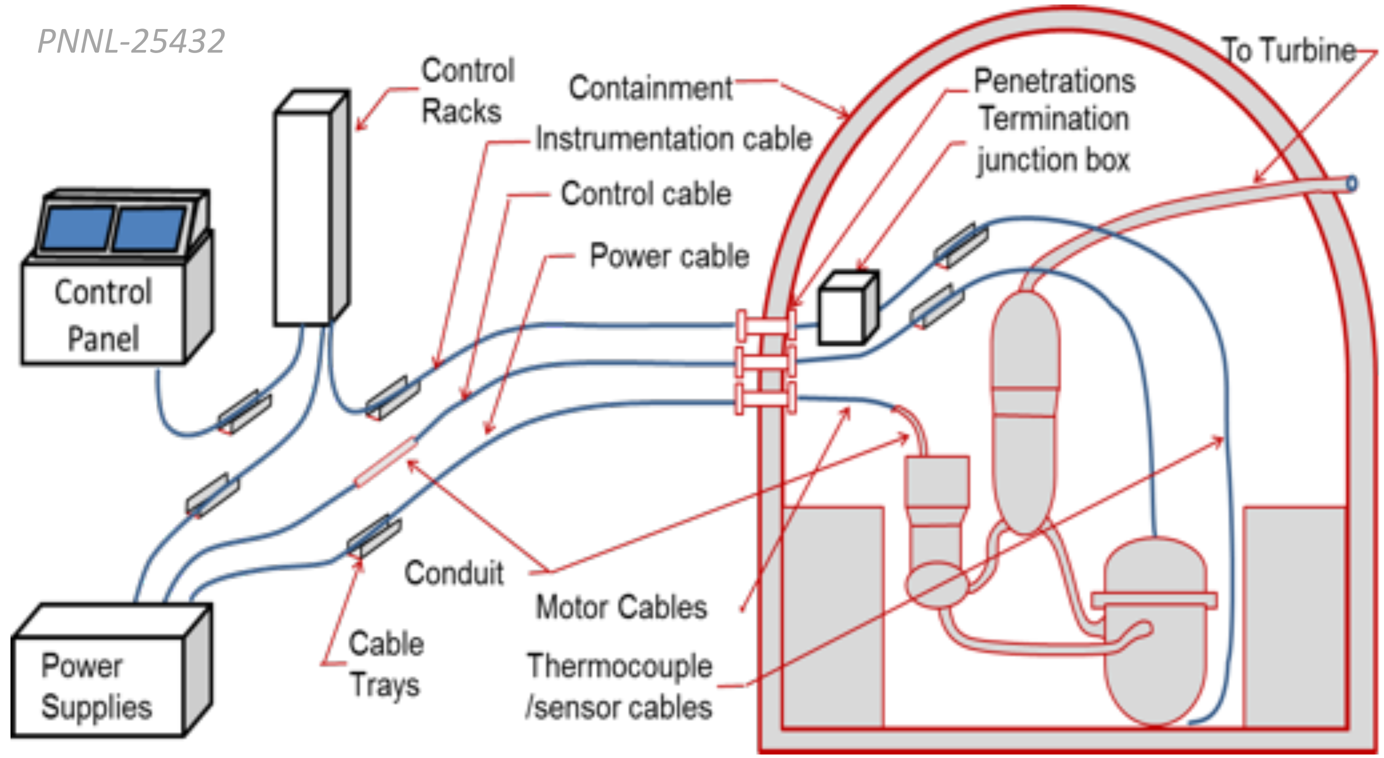
Cables in Subsequent Subsequent License Renewal

"Provided plant operators continually maintain, replace or repair equipment and components and make necessary upgrades, there is no operational reason that the [subsequent] license renewal process should be different than the first license renewals."

"Effective programs require a technical understanding of aging effects, inspection and assessment techniques, mitigation measures, and guidance on repairs or replacements."

NEI Second License Renewal Roadmap 2015

PNNL-25432



Continued Use of Aging Cables

CAMP	➢ Cable Aging Management Plans
Mitigation	➢ Arrest/attenuation of active aging
Rejuvenation	➢ Treatment to extend useful life
Monitoring	➢ Effective testing and Online monitoring
Simulation	➢ Digital twin assessment of physical asset
Validation	➢ Qualification in place
Test Bed	➢ Integrated Test Bed: NDE validation

Managing Inventory

CAMP

allthingsnuclear.org

- ➢ Cable Aging Management Plans
- Operating Experience: Local, Industry
- Generic Aging Lessons Learned
- Licensee Event Reports
- Inventory: Cables, Environments
- Testing: Historical Record, Established Criteria

Halting/Reversing Aging

Mitigation

- Arrest/attenuation of active aging
  - Alleviate adverse environments
  - Provide location intervention

Rejuvenation

- Treatment to extend useful life
  - Renew mechanical properties
  - Renew electrical properties

Leo Fifield (leo.fifield@pnnl.gov)

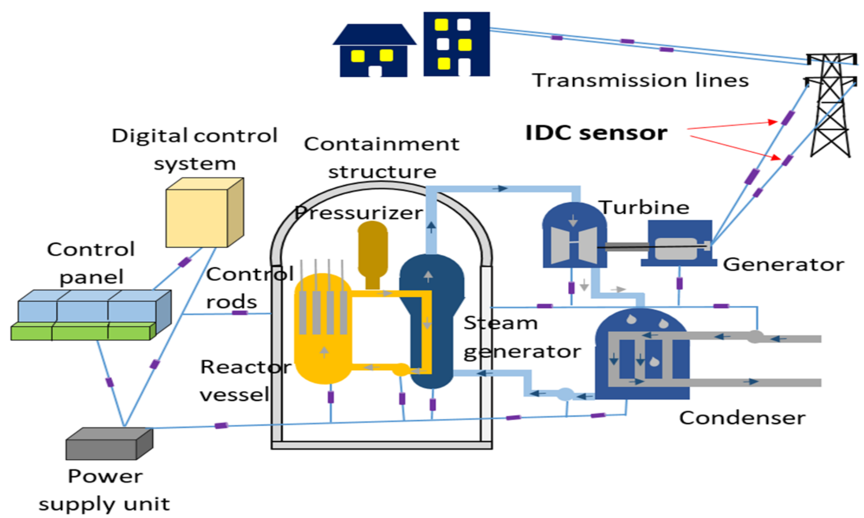
Condition-based Management

Monitoring

➢ Effective testing and Online monitoring
- Combination of test methods
- Online or in-situ assessment

Validation

➢ Qualification in place
- Risk based
- Risk Insights

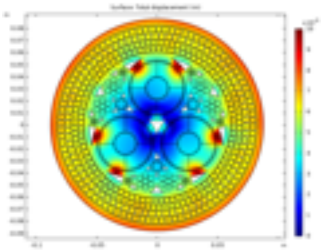


Aging Management Tools

Simulation

➢ Digital twin assessment of physical asset

- Analytical and data-based models
- Tighter following of cable health



Test Bed

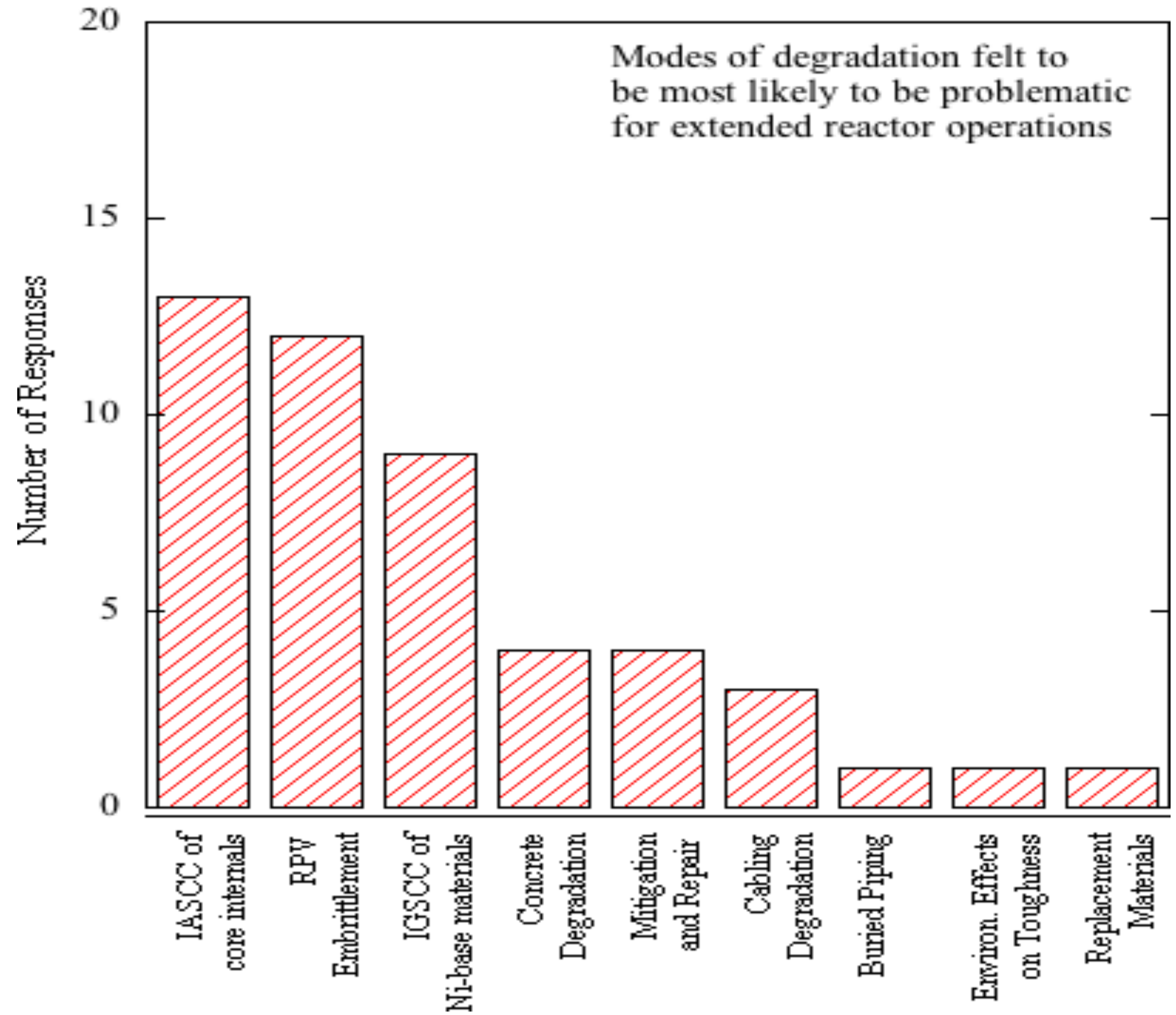
➢ Integrated Test Bed: NDE validation

- Offline analysis of NDE tools
- Offline type testing

Life Beyond Eighty Path Forward

Identification and Prioritization of Research Activities

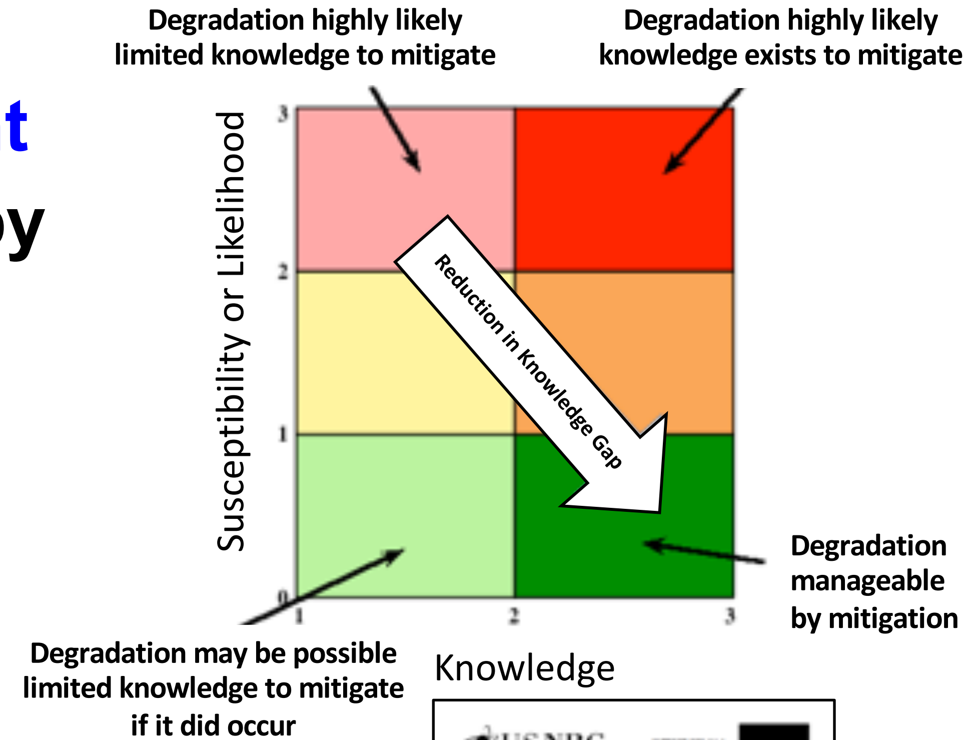
- In **2008**, three national laboratories, two universities, two nuclear reactor vendors, a nuclear power plant utility, and nine key experts from EPRI participated in research discussions **to address extended NPP operations** at EPRI Charlotte.
- Goal was to **identify, formulate, and prioritize the competing needs** in a collaborative manner with, DOE NLs, industrial and regulatory partners.
- Teams identified an **initial list of the most pressing research tasks.**

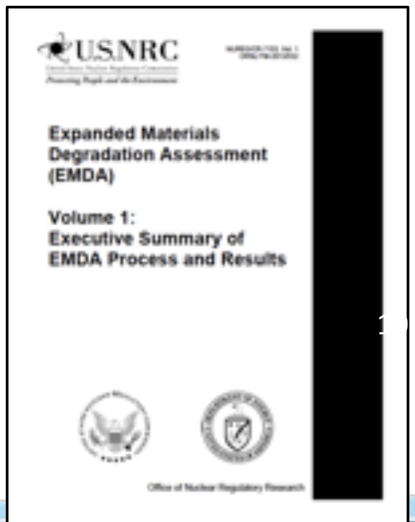


Summary of modes of degradation that are the most likely to be problematic for long- term operation of NPPs.

Establishing LBE Research Needs Based on an Expert Panel Consensus

- In FY 22, initiate a Subsequent or Second Expanded Materials Degradation Assessment (SEMDA) and publish a gap analysis report by 2024:
- Expert panel from the nuclear community (industry, EPRI, national labs, universities, international, & NRC)
- Address gaps but will not rank in terms of priority.
- SEMDA: RPV, Internals, Concrete, Cables, Mitigation, NDE / On-line Monitoring, Harvesting




U.S.NRC

Expanded Materials Degradation Assessment (EMDA)

Volume 1: Executive Summary of EMDA Process and Results

SEMDA: Path Forward

- **When should it begin**? (FY 22….)
- **Who should support the effort**? (DOE / NRC / PWROG / BWROG)
- **Who should participate**? (DOE NL, NRC, EPRI, Industry, International, Universities)
- **What should be included**?
  - *Metals (RPV, Core Internals and Piping)*
  - *Concrete (Irradiation, Creep, ASR, debonding)*
  - *Cables (rejuvenation, simulations,*
  - *Mitigation (welding, advanced materials, annealing)*
  - *Monitoring (On-line, embedded sensors)*
  - *Validation (Harvesting ex-service materials)*

Thank you for your participation!
Adjourn

Sustaining National Nuclear Assets

http://lwrs.inl.gov

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Xiang (Frank) Chen, LWRS Materials Research Deputy Pathway Lead (chenx2@ornl.gov)

Bruce Hallbert, LWRS Technical Integration Office Director (bruce.hallbert@inl.gov)