Introduction of session 4

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International Workshop on Ageing Management Considerations in Mechanical codes and standards
28-29 June, 2023, Tokyo, Japan

Introduction of Session 4: Challenges of ageing phenomena in C&S applied to SMRs/AMRs

Suqiang Xu, Canadian Nuclear Safety Commission

Session 4: Challenges of ageing phenomena in C&S applied to SMRs/AMRs

- Objectives
  to share information and discuss the challenges related to ageing phenomena in small modular reactors (SMRs.)
- Scope
  Regulatory Requirements, Research Advances and Industrial Experience.
- Challenges
  Creep, fatigue, in-service deterioration as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material.
- Presenters
  - Dr. Suraj Persaud, Queens's University;
  - Dr. Suqiang Xu, CNSC;
  - Mr. Chris Wax, EPRI.

Degradation of Materials in High Temperature Small Modular Reactors

Dr Suraj PERSAUD

Queen’s university (Canada)

Degradation of Materials in High Temperature Small Modular Reactors

Suraj Persaud

International Workshop on Ageing Management Considerations in Mechanical Codes and Standards

Motivation for SMRs

- Scaled-down, flexible models of nuclear power plants.
  - Intrinsically safer design and improved performance
  - Reduced costs due to modular production in a factory setting
  - Technology readiness is established and at a high level for some designs
  - Application to remote and northern communities
  - Economic benefits
  - Key driver: very low carbon energy source, climate change

Key Technological Gap

- There are many proposed designs for SMRs over a range of power output (e.g., in Canada there are prototypes planned ranging from 5 MWe to 300 MWe, but up to 1500 MWe have been proposed)
- Many designs require the use of novel coolants to achieve intrinsic safety features and improve efficiency (i.e., operate at higher T)
  - Molten salts (chloride, fluoride, nitrate)
  - Liquid metal (sodium, lead)
  - High temperature gas reactor (He cooled)
  - Small modular water reactors

- Perhaps with the exception of water reactors, materials performance and selection is a key technological gap for SMR deployment

Sodium Cooled Fast Reactor

- Temperature range usually ~350 to 500 °C, atmospheric pressure, providing a level of intrinsic safety
- Very good neutron energy economy, also leads to less nuclear waste
- Probably the largest general concern is the reactivity of Na with water and air
- For materials – radiation fields in core and high temperature are most concerning

Some Key Materials Issues

- Higher radiation fields and higher temperature in the reactor core compared with conventional water reactors.
  - Leads to swelling of materials due to defect production from radiation damage – embrittlement, especially in austenitic stainless steels. Helium production is also an issue.

  - Creep (and radiation creep) are also an issue, which has required moving toward steels with fine precipitates (i.e., ODS steels).

  - The effect of the coolant on corrosion and/or phase transformations is less studied.

Novel Facilities for Research

- Several similar facilities in the world.
- 4 MV tandem producing 8MeV H or 12MeV He; for damage or implantation studies.
- Facility at Queen's led by Prof. Mark Daymond, UNENE Research Chair in Nuclear Materials.

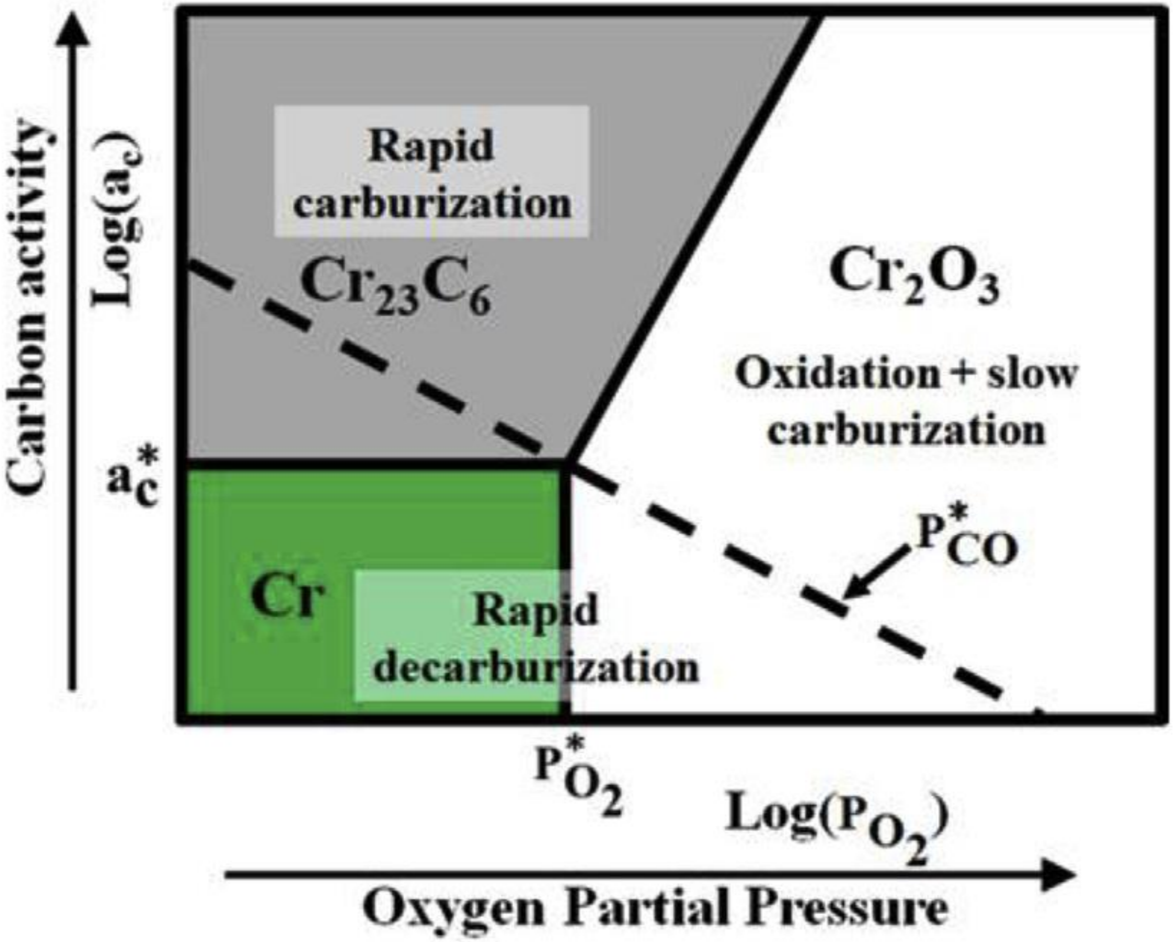
High Temperature Gas Reactors

- High temperature (750C to ~950 °C) He-cooled reactors, graphite moderated. Secondary side coolant could be something different (e.g., molten salts).

- At these temperatures, creep is an issue with materials proposed for use (e.g., high alloy stainless steels, Ni-based alloys).

- He gas is inert, and does not cause material degradation. The main driver of degradation is impurities in the coolant, which are present at ppm-levels.
    - CO, CO_2, H_2, H_2O, CH_4

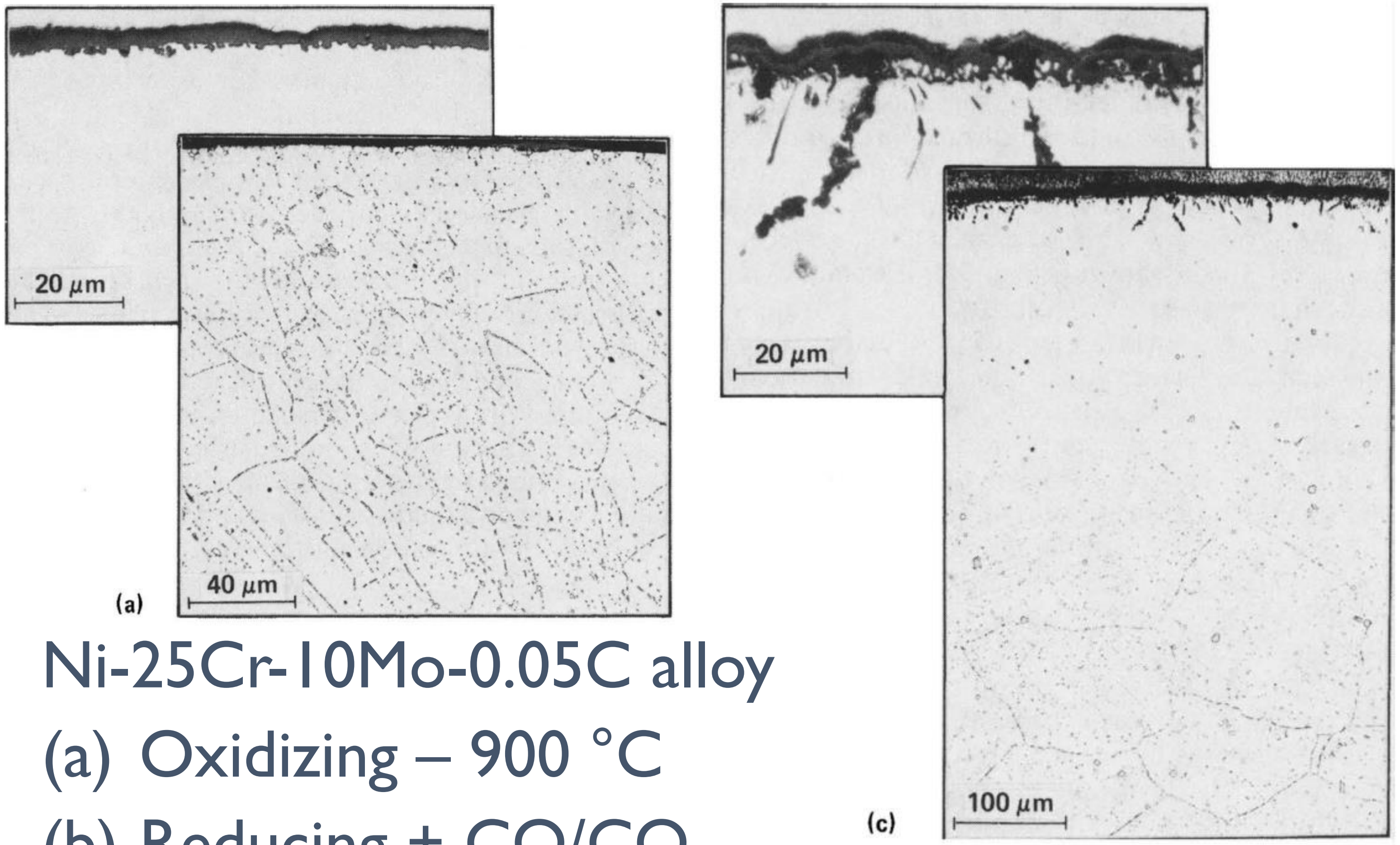
High Temperature Gas Reactors



G.S.Was et al., Journal of Nuclear Materials 527 (2019) 151837

- CO, CO_2, H_2, H_2O, CH_4 : These impurities can lead to carburization, decarburization, internal oxidation, or intergranular oxidation, leading to embrittlement.
- Possible mechanisms depend on gas ratios. However, can not apply simple thermodynamic principles (i.e., Ellingham diagram). The ppm-level concentration of the elements results in deviations due to kinetic limitations.

Nano-to-microscale Embrittlement

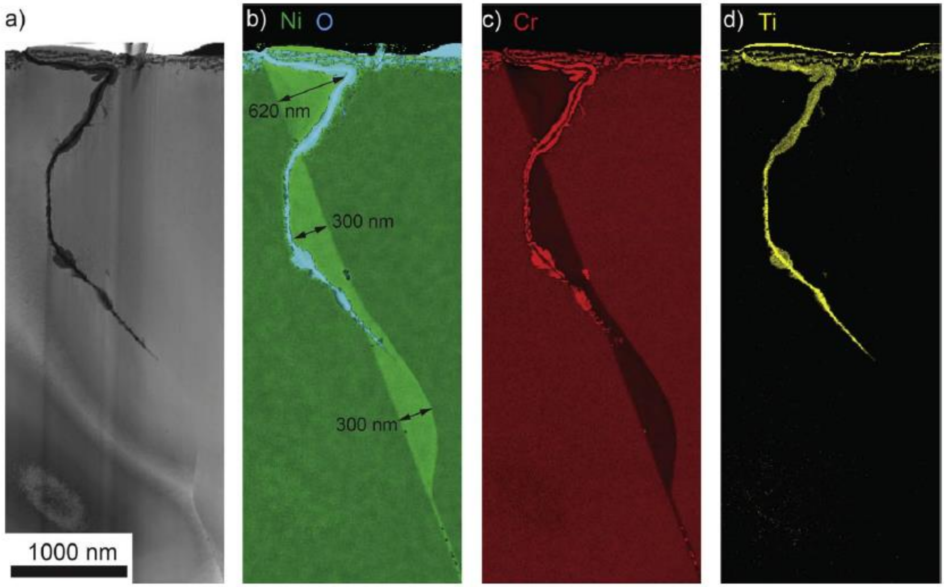


Ni-25Cr-10Mo-0.05C alloy

(a) Oxidizing – 900 °C

(b) Reducing + CO/CO_2

Quadakkers & Schuster, Nuclear Technology, 66:2, 383-391, 1984

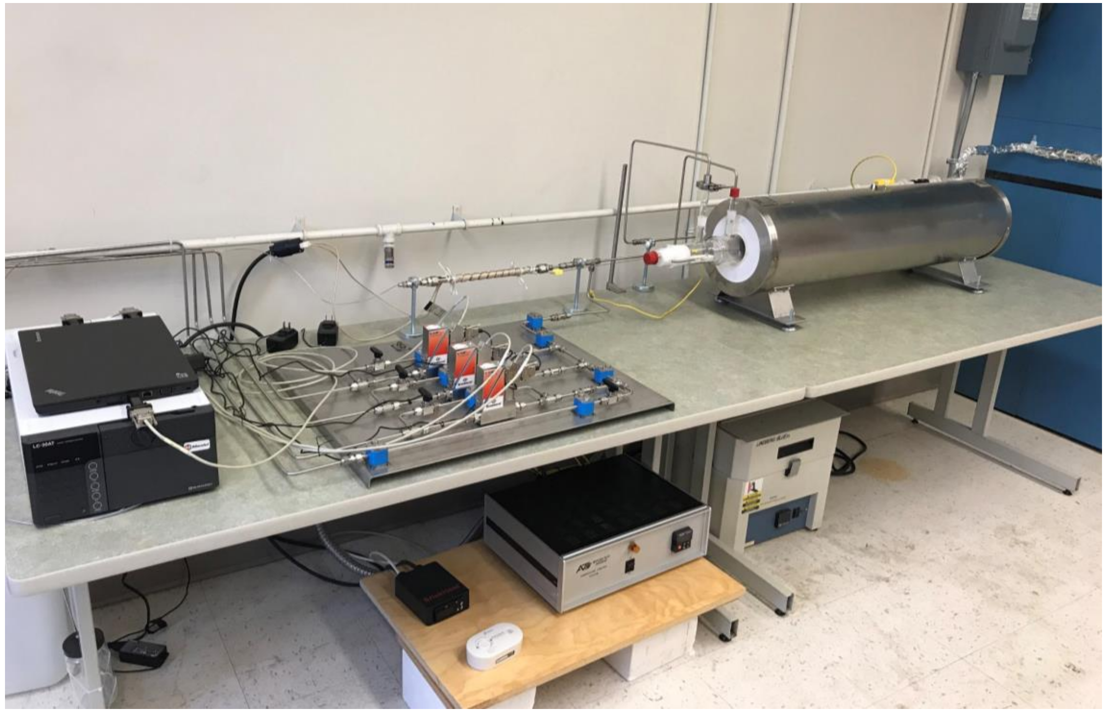


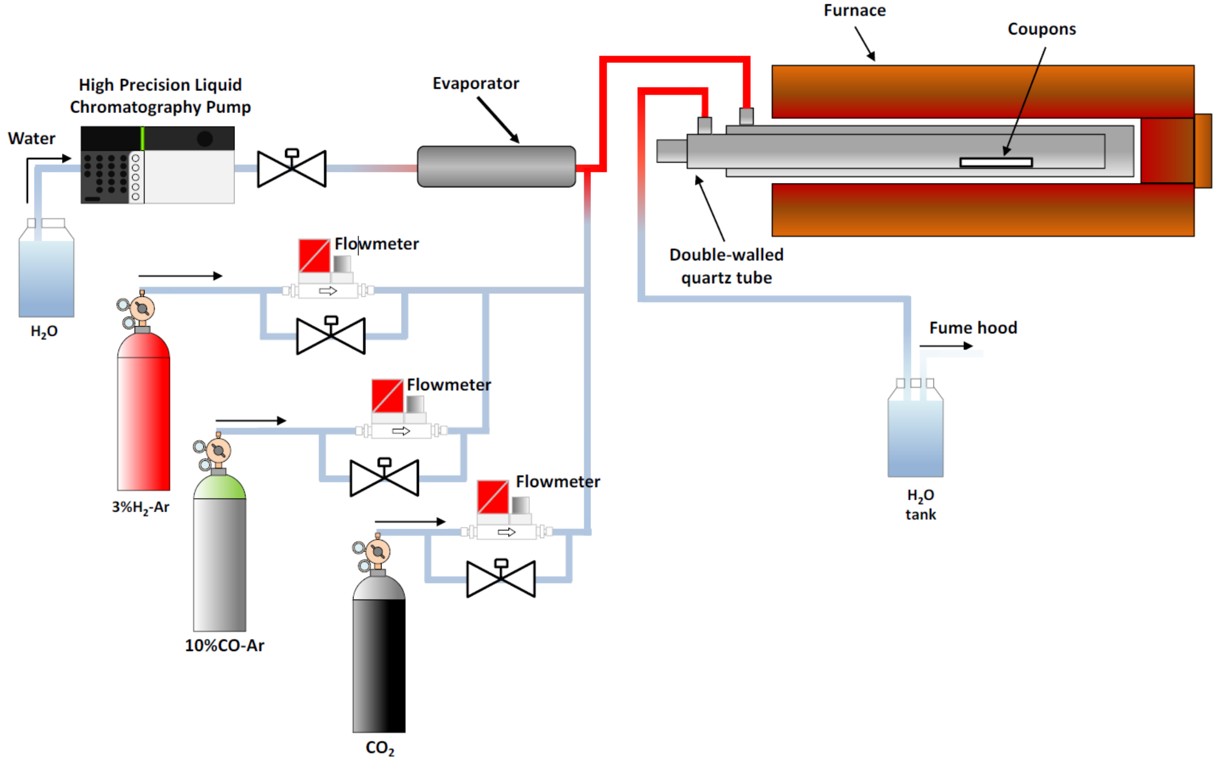
Nanoscale analysis of intergranular and internal oxidation in Alloy 600 in ~500 °C hydrogenated steam

S.Y. Persaud et al., Corrosion Science, 133, 36-47, 2018

Facilities for Studying HTGR Corrosion

- In addition to using the accelerator to generate radiation damage, exposure can be performed in high temperature mixed-gas reactors.





- In-situ experiments not possible yet, but are a part of future plans – more later.

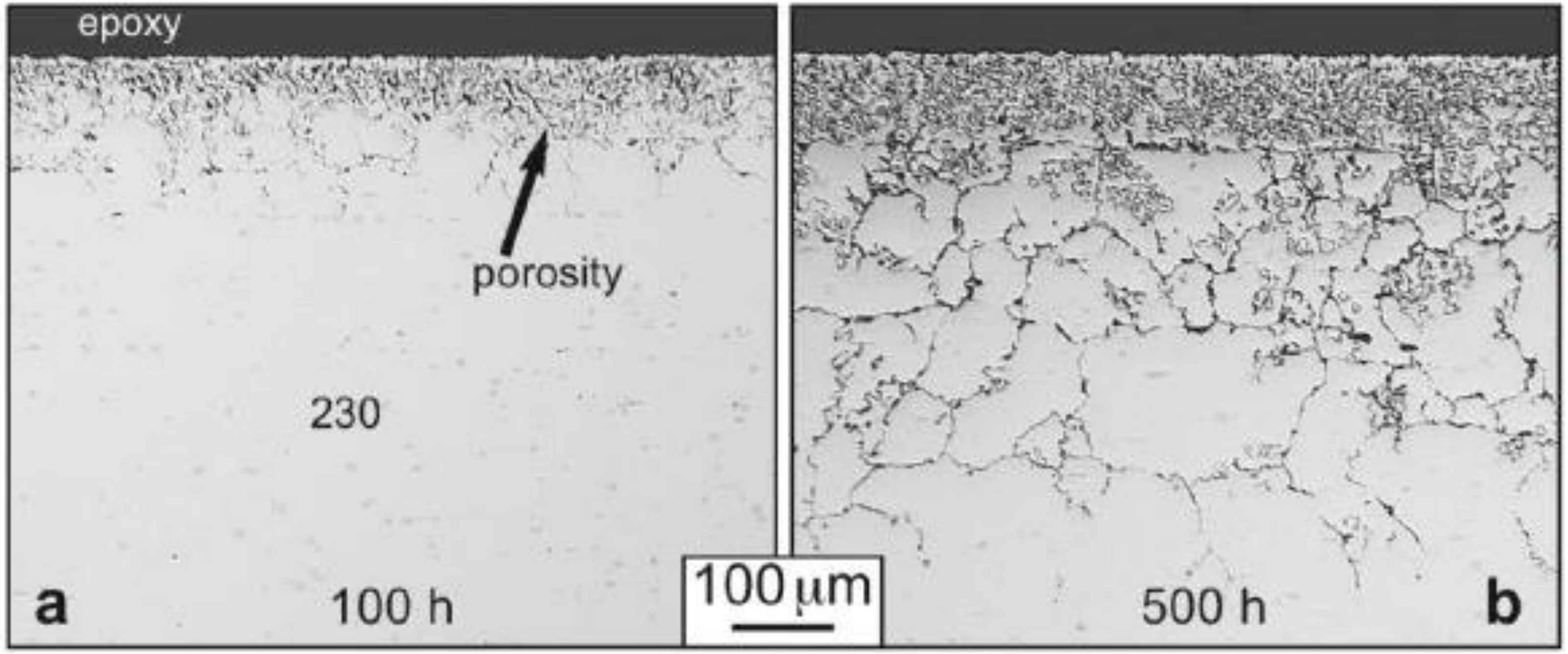
Molten Salt SMRs

- Molten salt SMRs typically use chloride or fluoride salts as coolant or mixed fuel-coolant.
- Negligible vapour pressure, radiolytic stability, low volatility, high thermal conductivity – very attractive as a coolant medium at up to 800 °C.
- However, salts are inherently hygroscopic and this leads to inevitable impurities in the salt, particularly moisture and oxygen.
- While there are other impurities, moisture and oxygen are considered the most detrimental to corrosion.

Corrosion in Molten Salts

- Cr (and Fe) are soluble in molten Cl and F salts with moisture contamination. Ni (and Mo) are relatively stable in comparison.
- This leads to selective dissolution of Cr in Ni-based and high alloy stainless steels proposed for MSRs. This phenomenon is known as dealloying, and has recently been discussed in this context.

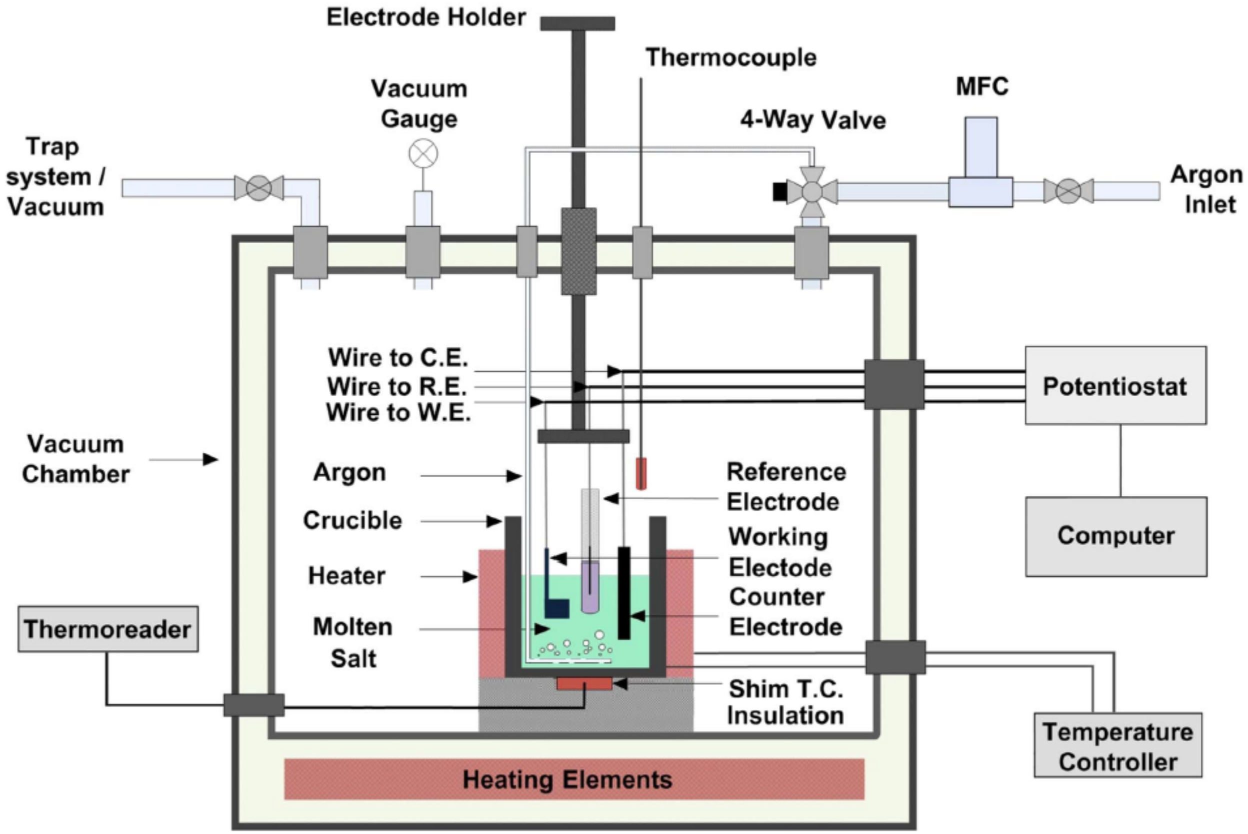
Haynes 230 in K-Mg-Na Cl salt mixture at 800 °C for 100 h



Pint et al., Materials and Corrosion, 70:8, 1439-1449, 2019

Need Careful Experiments

- Avoiding and/or measuring moisture content is very important (also mentioned crucible effects are scarcely studied).

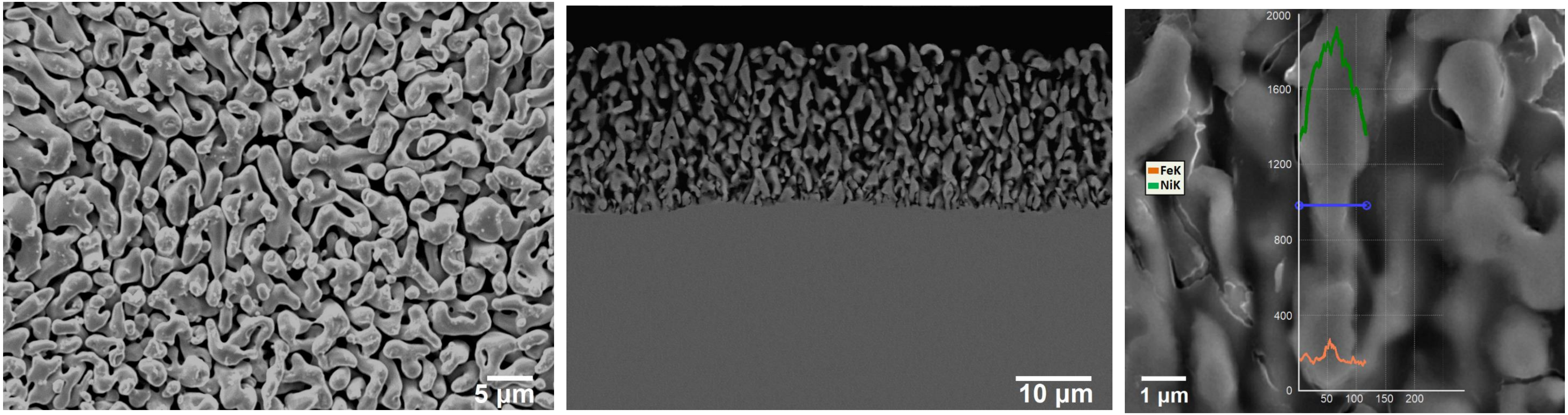


Ghaznavi, Persaud, Newman, Journal of the Electrochemical Society, 169, 061502, 2022

Dealloying in Molten Cl Salts at 350 °C

- LiCl-$MgCl_2$-KCl eutectic mixture at 350 °C

Fe52Ni48

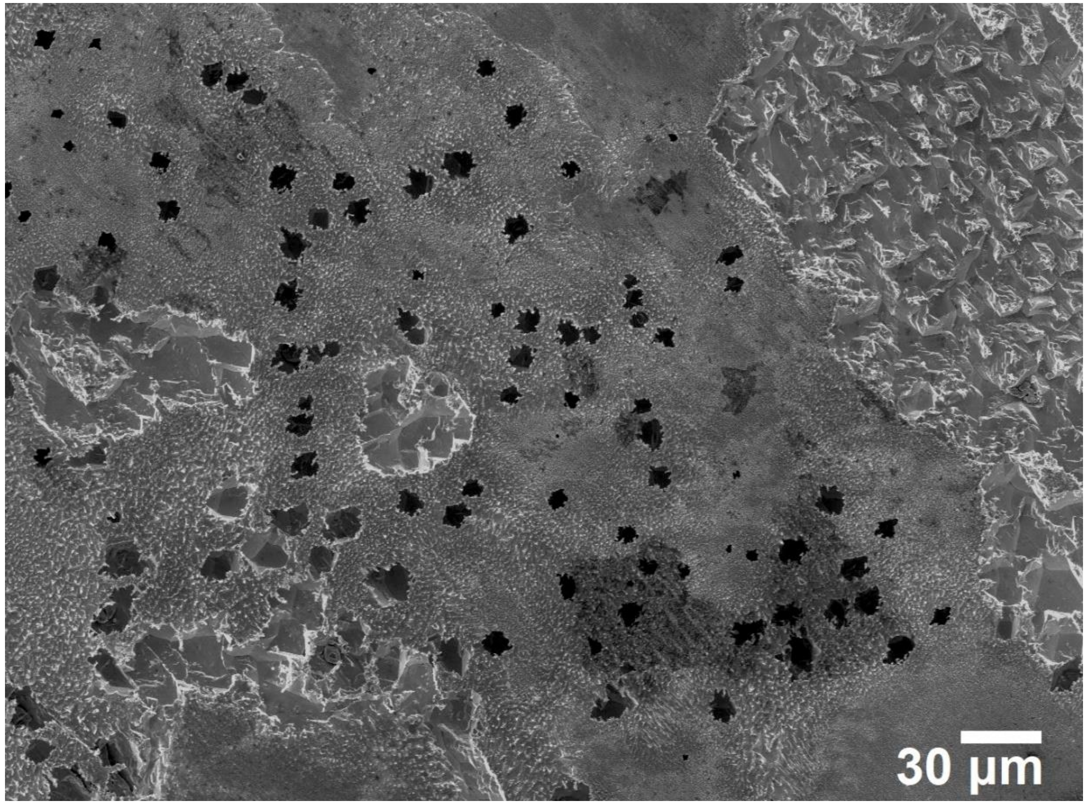


at +5 mA/cm^2 for 4000 s

Ghaznavi, Persaud, Newman, Corrosion Science, 197, 110003, 2022

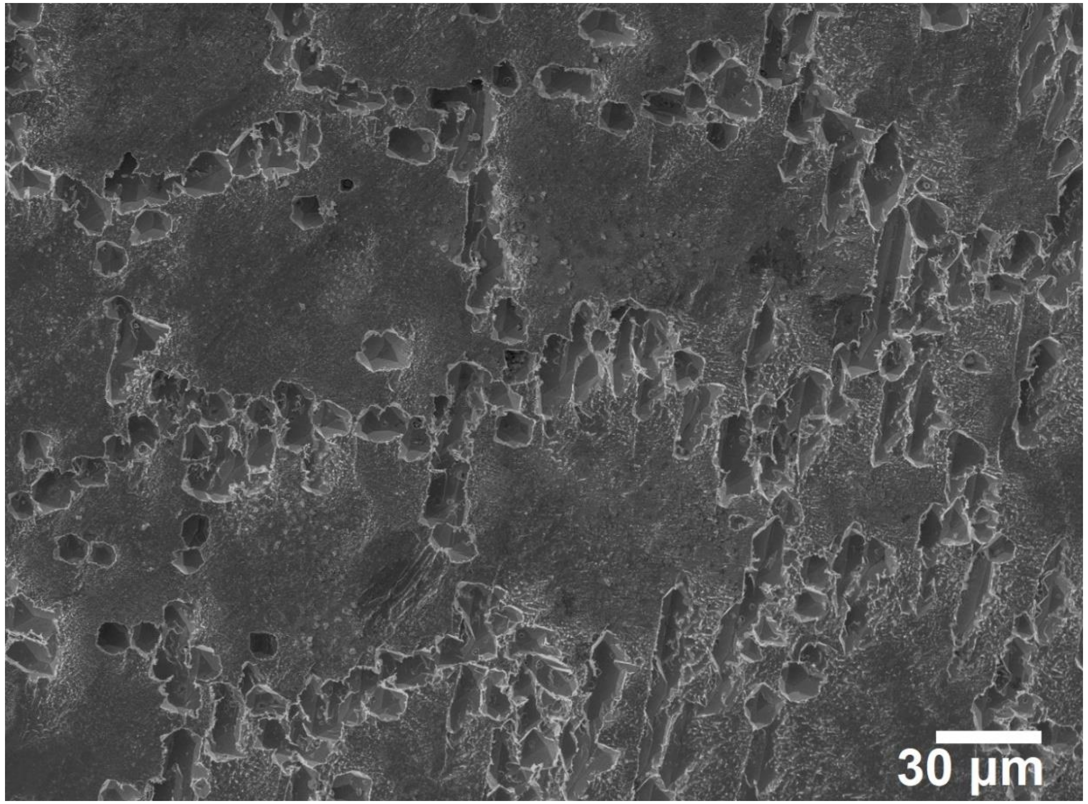
Dealloying in Molten Cl Salts at 350 °C

Fe38Ni62



at +5 mA/cm^2 for 4000 s

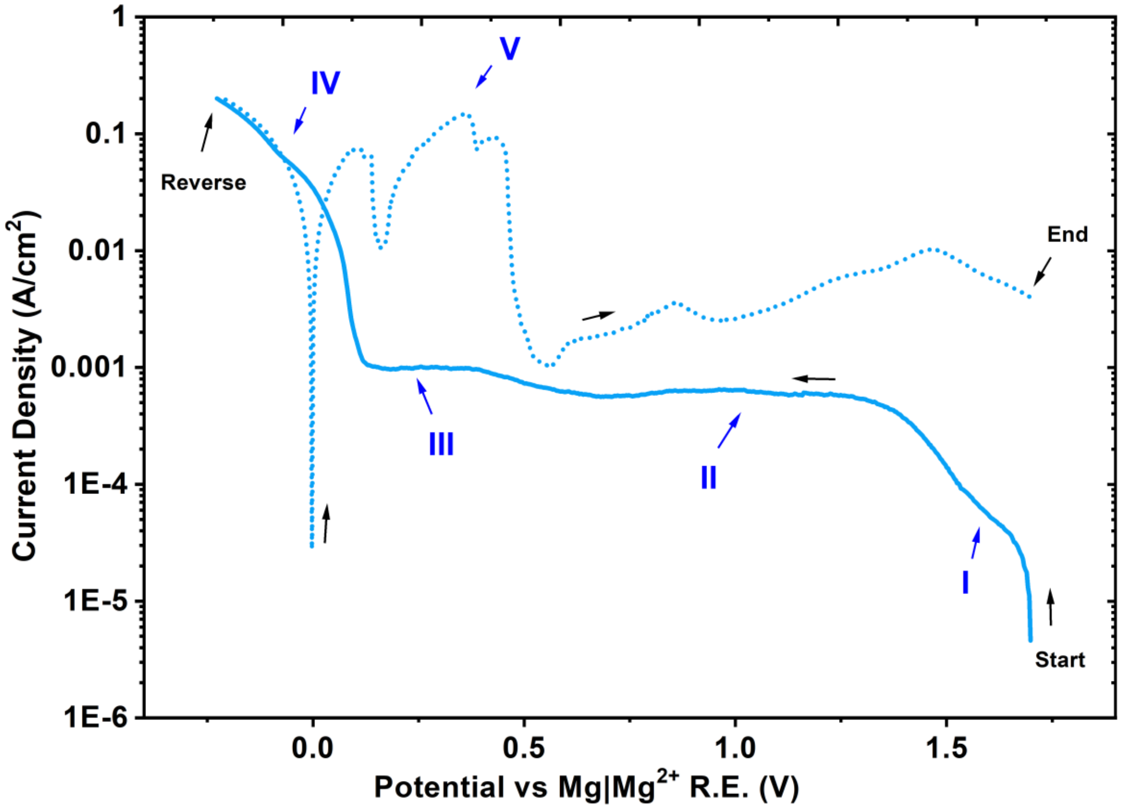
Fe32Ni68

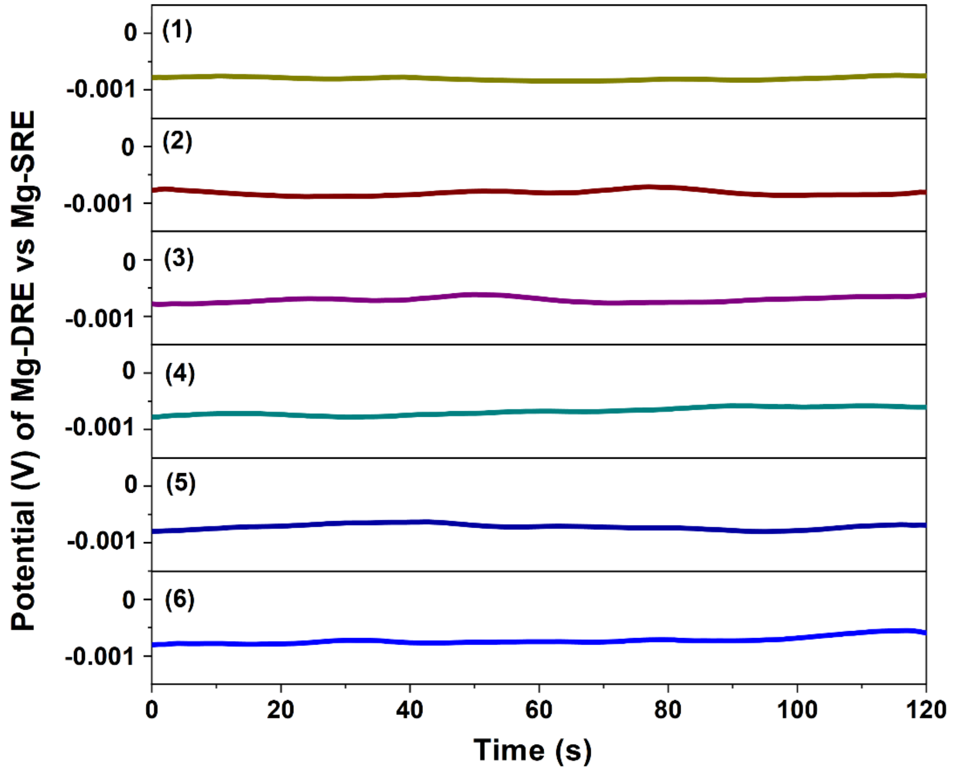


at +5 mA/cm^2 for 4000 s

Ghaznavi, Persaud, Newman, Corrosion Science, 197, 110003, 2022

Electrochemistry in Molten Salts





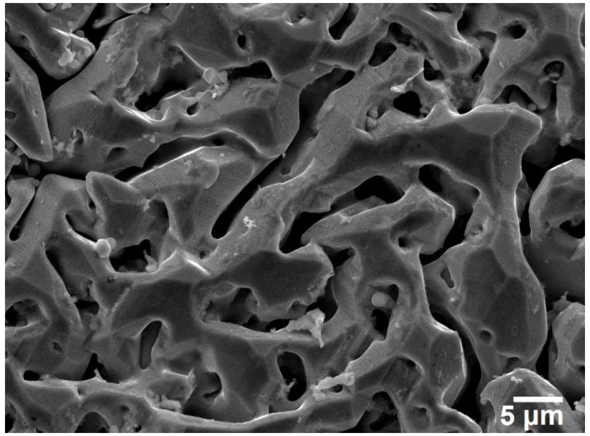
Mg behaves as a nearly reversible Mg|Mg^{2+} reference electrode,
as validated by in-situ-generated Mg dynamic reference electrodes.

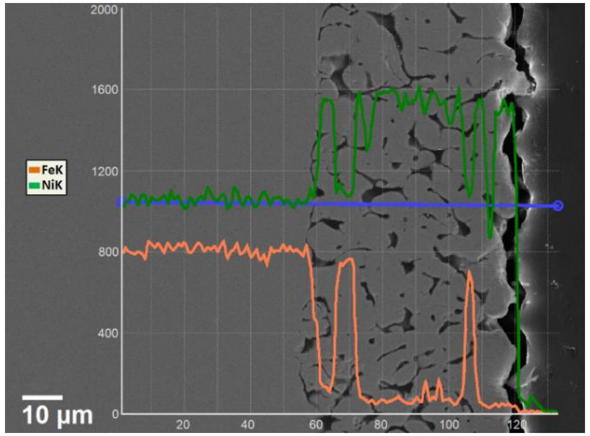
Ghaznavi, Persaud, Newman, Journal of the Electrochemical Society, 169, 061502, 2022

Molten Salt with Ni Cations

Fe32Ni68

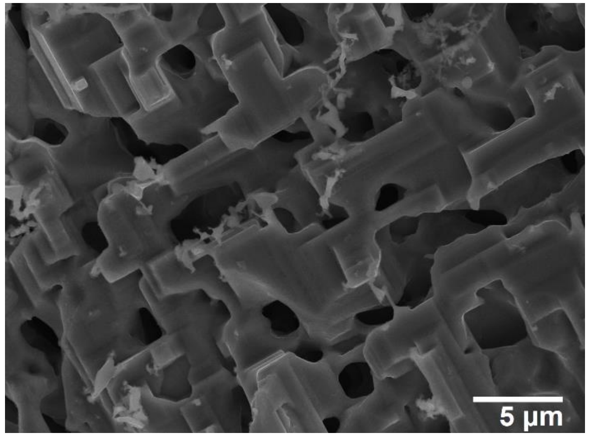
at 20 mV above OCP for 1 h,
1 wt.% $NiCl_2$

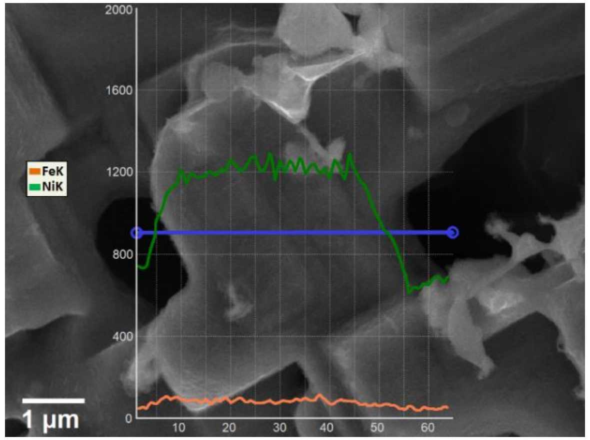




Fe32Ni68

at 20 mV above OCP for 1 h,
3 wt.% $NiCl_2$

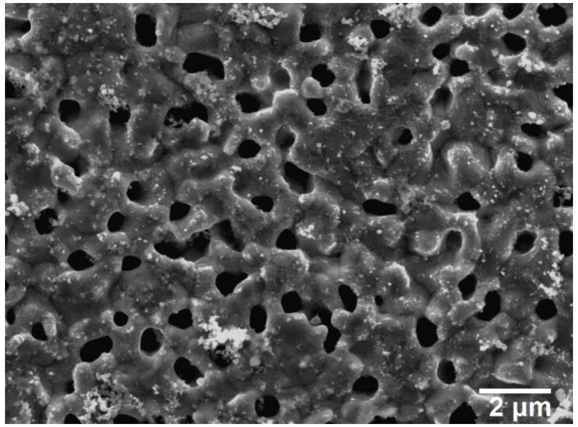


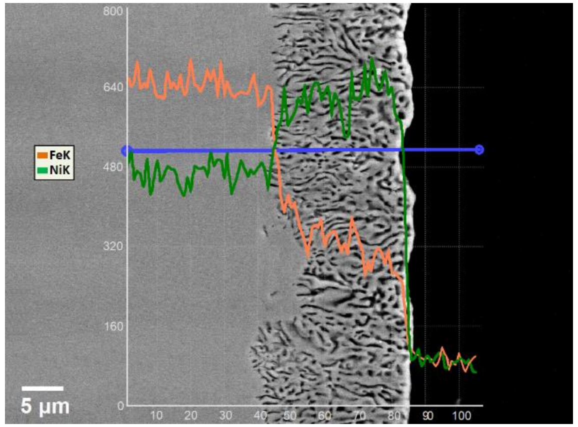


Ghaznavi, Persaud, Newman, Journal of the Electrochemical Society, 169, 061502, 2022

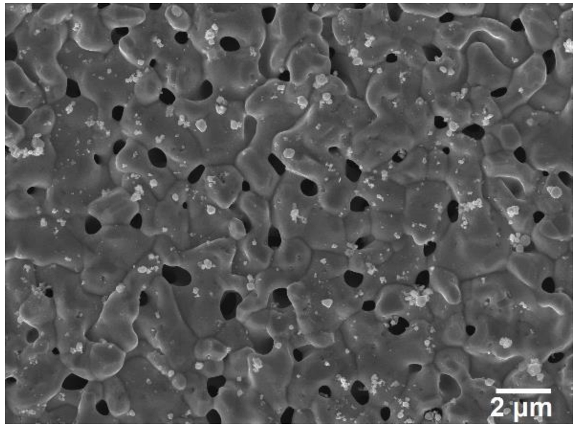
Molten Salt Corrosion at Higher T

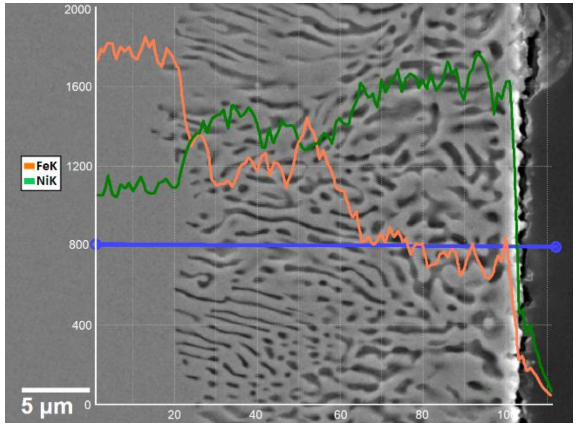
Fe52Ni48, 500 °C



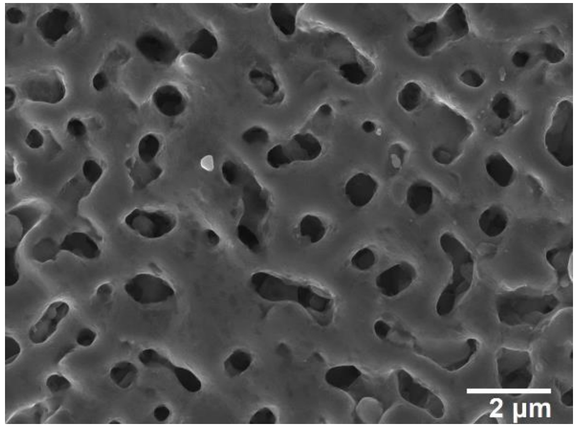


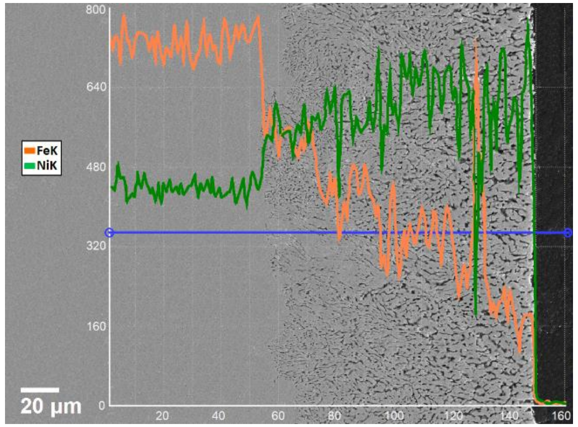
Fe52Ni48, 600 °C





Fe52Ni48, 700 °C

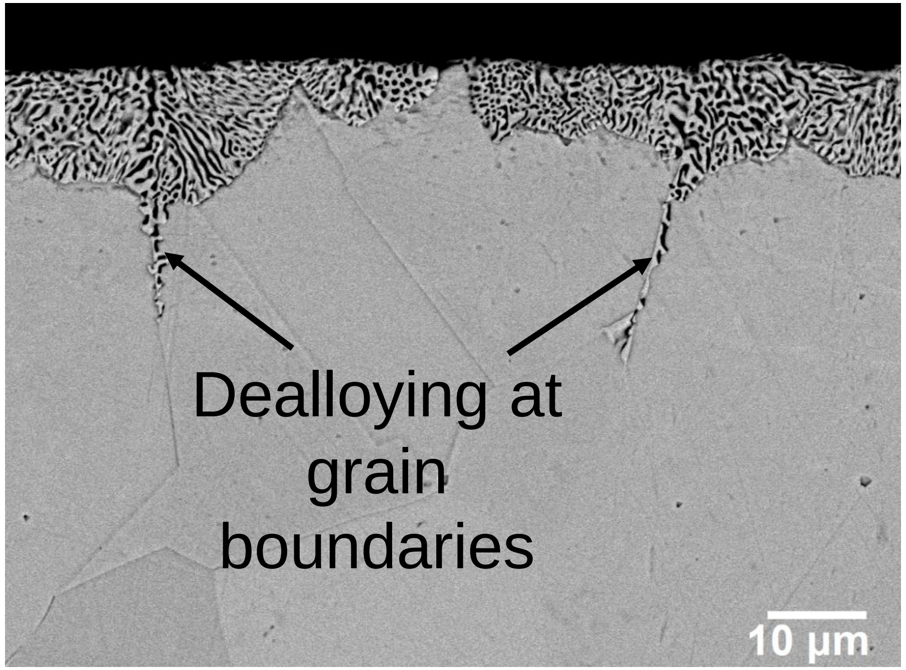


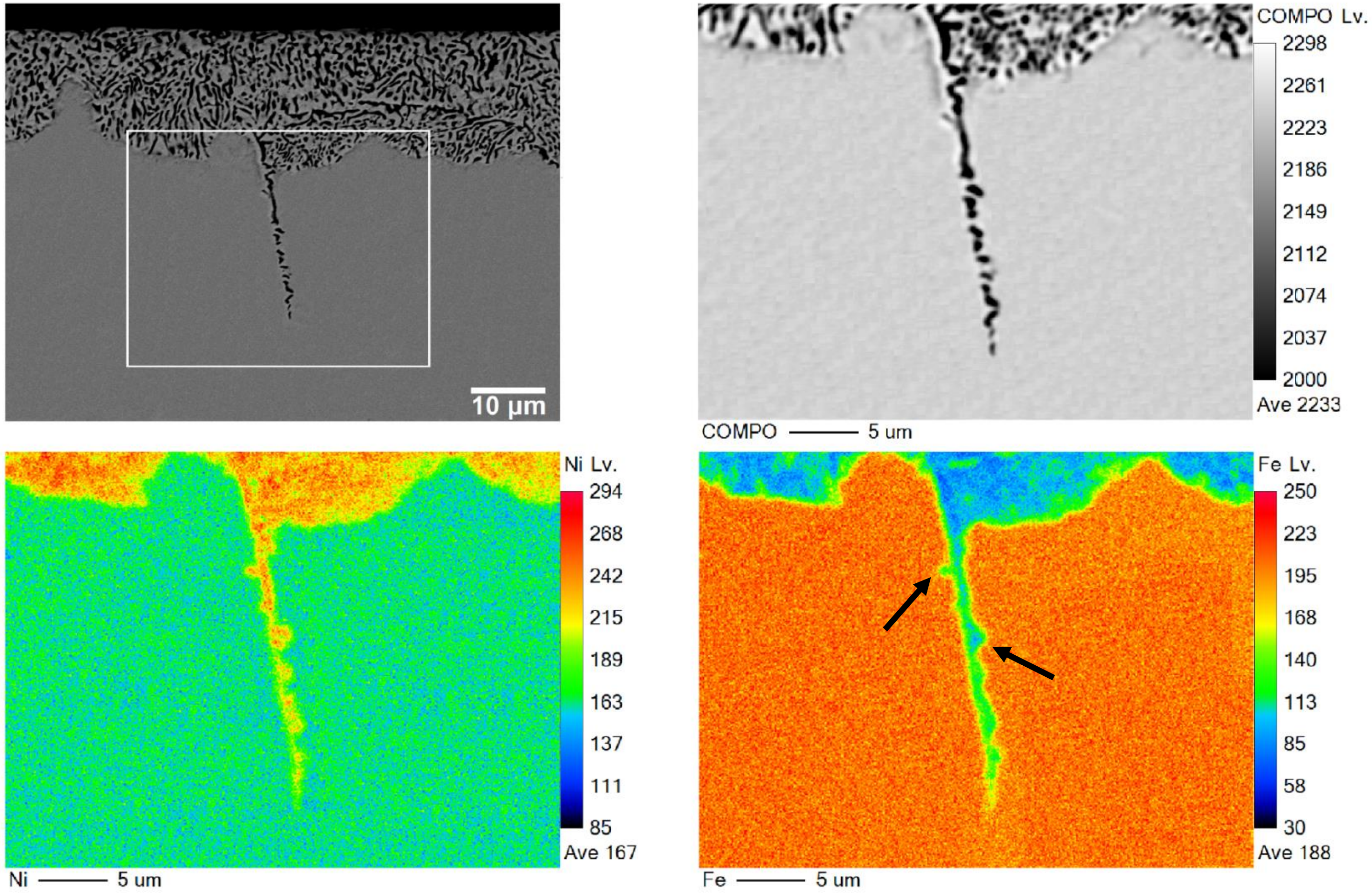


Ghaznavi, Persaud, Newman, Journal of the Electrochemical Society, 169, 111506, 2022

Grain Boundary Dealloying

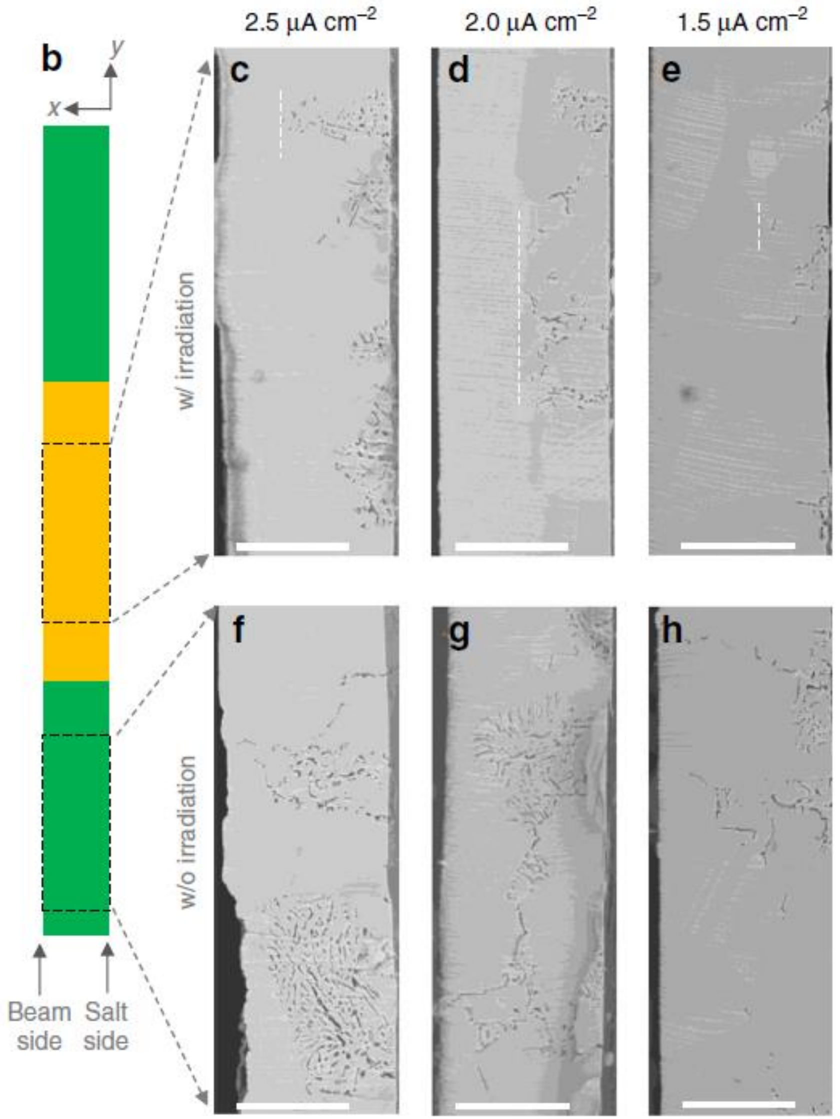
Fe52Ni48
at 500 °C





Ghaznavi, Persaud, Newman, Journal of the Electrochemical Society, 169, 111506, 2022

Effect of Radiation?



- This is still an area with little research, especially *in-situ*.
- The results on the left suggest that dealloying occurs with and without irradiation; however, it may be decelerated in a radiation field (in this one case, or not).
- Contextualizing the effect of radiation damage and dealloying simultaneously is still ongoing.

Zhou et al., Nature Communications, 11:3430, 2020.

Some Notes on Pb-cooled Reactors

- Not currently a research focus for my group at Queen's; however, there is recognition of structural materials issues

- Solubility of Ni is an issue, has led to proposal for ferritic steels in Pb reactors

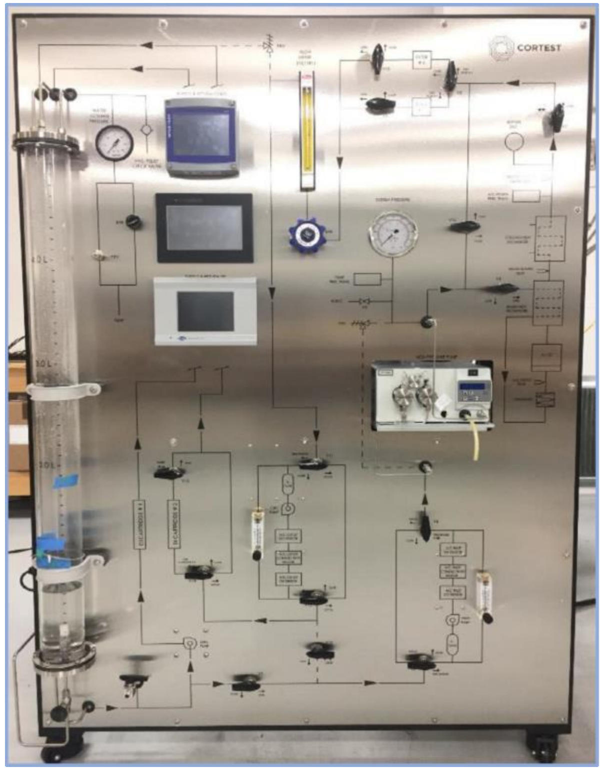
- Maintaining passivity is a significant concern and varies as a function of oxygen content and temperature

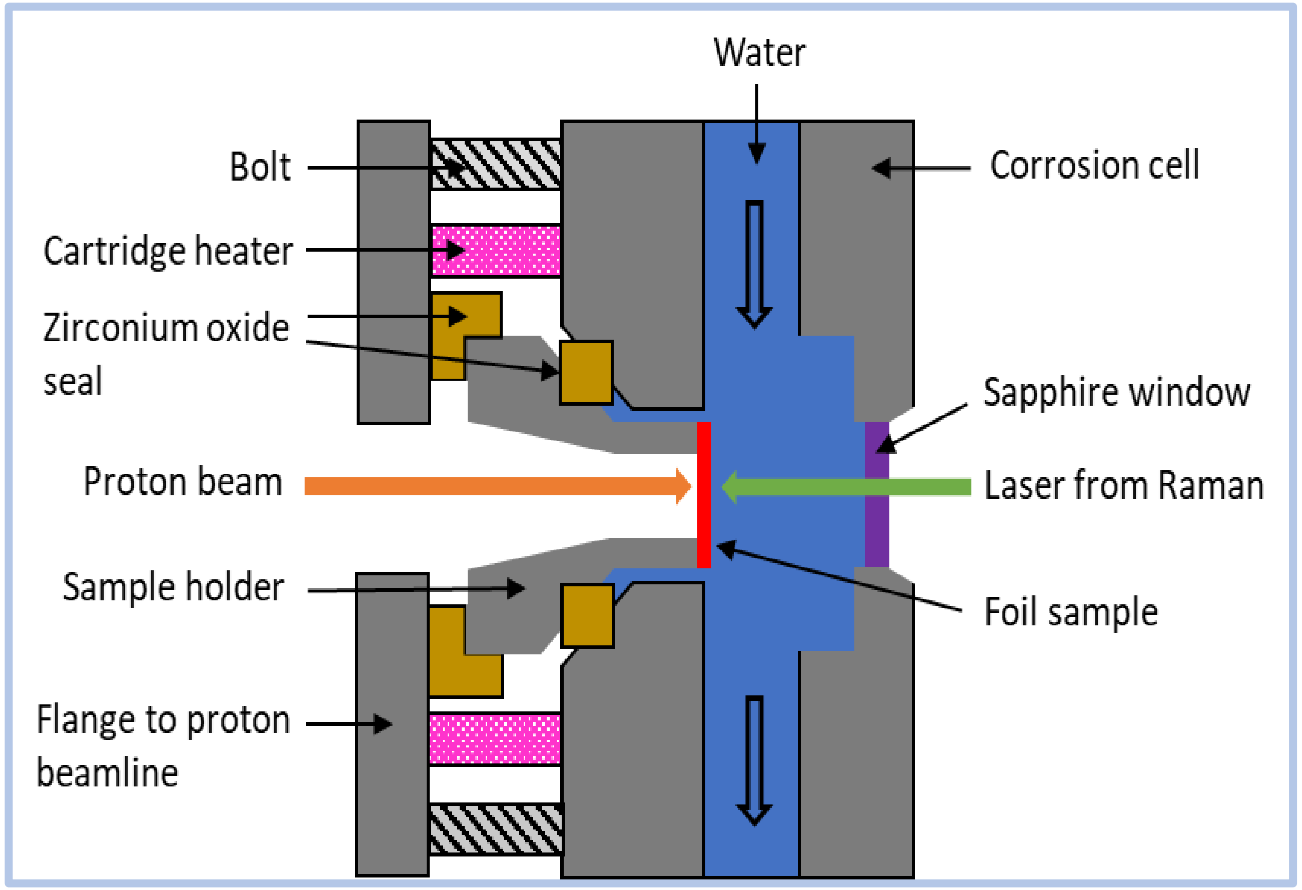
- Liquid metal embrittlement is a key concern – reduces the fracture stress of materials

Some Notes on Water Reactor SMRs

- The degradation mechanisms and materials selection is deemed to be similar for BWR and LWR-based SMR designs.
- However, the designs do have differences compared with conventional reactors – for example, steam generators sometimes have the secondary and primary side coolant on opposite sides.
- There also needs to be consideration for how design changes can impact stress and/or material conditions, and, from a chemistry perspective, lead to local chemistry variations. For example, some design consider AM materials for their components.

In-situ Irradiation-Corrosion Facility





Final Notes

- SMR are a viable option as a technology-ready, very low carbon energy source for combating climate change.

- Structural material degradation remains an issue for several of the proposed designs, and material selection is an uncertainty.

- However, proactive research and application of state-of-the-art facilities are making significant strides towards answering key questions about materials degradation, considering both radiation effects and the corrosive medium.

Acknowledgements

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CORROSION GROUP

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U N E N E
University Network of Excellence in Nuclear Engineering

Material Aging Challenges for Small Modular Reactors, a Regulatory Perspective

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Material Aging Challenges for Small Modular Reactors, a Regulatory Perspective

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Workshop on Aging Management Considerations in Mechanical Codes and Standards
Tokyo, Japan, June 28-29, 2023

CNSC CCSN

Introduction

- Canadian Nuclear Safety Commission (CNSC) is performing licensing or pre-licensing review of several SMR designs with various reactor coolants, e.g. high temperature (HT) gas, liquid metals, molten salts, and light water
- CNSC staff observes that international and domestic codes and standards, technical guidance, and research work provide solid basis for the design review
- Staff identifies/anticipates some potential challenges in material evaluations, based on the available information to date

SMR Vender Design Reviews in Canada

Vendor	Reactor	MWe
Terrestrial Energy Inc.	IMSR- Integral Molten Salt Reactor	200
Ultra Safe Nuclear Corp.	MMR High-Temperature Gas Reactor	5-10
ARC Nuclear Canada Inc.	ARC-100 Liquid Sodium	100
Moltex Energy	SSR-W300- Melt Salt Reactor	300
Holtec Int. Co.	SMR-160 Pressurized Light Water	160
GE-Hitachi Nuclear Energy	BWRX-300, Boiling Water Reactor	300
X Energy, LLC	Xe-100 , High-temperature Gas Reactor	80
Westinghouse Electric Co., LLC	eVinci Micro Heat Pipe Reactor	5

Typical Pressure Boundary Classification, Codes & Standards

Safety Classification	Important to Safety(ITS)				Non-ITS
	Safety Related			Non-Safety Related with special treatments	Non-safety Related
Low T PB Classification & Codes	Class 1 (QG-A)	Class 2 (QG-B)	Class 3 (QG-C)	Canada Class 6 (QG D)	Non-classified
	ASME, Section III, Division 1, Class 1	ASME, Section III, Division 1, Class 2	ASME, Section III, Division 1, Class 3	ASME, Section VIII, Division 1 or Division 2 ASME B31.1	Not Required
High T PB Classification & Codes	Class A (QG-A)	Class B (QG-B)		Canada Class 6 (QG C)	Non-Classified
	ASME, Section III, Division 5, Class A	ASME, Section III, Division 5, Class B		ASME, Section VIII, Division 1 or Division 2 ASME B31.1/B31.3	Not Required
Regulatory AM Requirements	Subjected to Aging Management(AM) Requirements				Not Required

A misconception is that ASME class B is listed non-safety related with special treatments.

ASME Section III D5, Limits

- With high quality assurance requirements, ASME Section III D5 addresses structural integrity subjected to many failure modes/load conditions, including mechanical and thermal fatigue and high temperature creep
- Clause HAA-1130 points out limitations of these rules. The rules do not cover deterioration that may occur in-service as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material
- Licensees/designers should take these effects into account

Potential Material Challenges

This presentation summarizes some potential challenges for licensing HT SMRs in the assessment of:

1. Stability of microstructure and material properties
2. Material compatibility with service environments
3. Environment effect on material design criteria
4. Properties of full-size components
5. Coating and cladding
6. Mechanical connection

Stability of Microstructure and Material Properties (1/2)

- Industry is actively developing new materials and qualifying additional high-performance materials for SMR construction. Various measures (precipitation, dispersion, solid-solution, grain-boundary strengthening) are used to increase material creep strength or degradation resistance

- In HT environment, mechanical properties and creep resistance may change slowly due to thermal aging:

  (a) Changes in precipitate & dispersed particle distribution

  (b) Formation of precipitates with solid solute elements

  (c) Recovery and recrystallization

Stability of Microstructure and Material Properties (2/2)

- The stability of microstructure and material properties will be assessed as part of the licensing review
- ASME code currently permits a time extrapolation factor of 5 for qualifying well-behaved, solid-solution alloys, while permitting a factor of 3 for other alloys (HBB-Y-2200)
- A 30-year design life may require 10 years of test data. This could be a challenge for material qualification, especially for qualifying new materials

Material Compatibility with Service Environments (1/3)

- SMRs have diverse designs & operational conditions: coolants, inlet/outlet temperatures, neutron spectrum & fluence, and thermal transients. The degradation mechanism and corrosion rates could be different among different SMRs

- More research is needed to understand and predict corrosion behavior for HT SMRs. For example, molten salt reactor:

  (a) Molten salt chemistry and impurities strongly affect degradation behavior. The salts are generally very hygroscopic; it is very difficult to control moisture-based impurities in salts or salt melt. Most of the literature does not provide sufficient information on salt impurity control

Material Compatibility with Service Environments (2/3)

(b) Some studies purposely added impurities into salts but may not actually represent SMR operation conditions. Accumulation of fission products will also affect salt corrosiveness, which has not been systematically studied

(c) Container/loop are not only a source of impurities but also form galvanic couples with test coupons and affect corrosion behavior. Graphite may also form galvanic couples with metal components, which has not been studied properly

(d) Loop temperature variation and mass transport behavior in flow can affect degradation process

(e) Maintaining mildly reducing conditions can mitigate corrosion and tellurium cracking. The envelope for SMR redox control has not been defined; therefore, it is not possible to generate representative data to assess the mitigation effectiveness

Material Compatibility with Service Environments (3/3)

- Understanding of HT cracking growth behavior is a key for flaw evaluation and component life prediction. Environment degradation could accelerate crack propagation; but the information in this area is very limited
- Environment degradation is a synergistic process, involving irradiation, corrosion, HT exposure, and stress. These factors could interact with each other and accelerate material degradation; more information on this is needed for assessing in-reactor material performance

Environment Effect on Material Design Criteria (1/3)

- For HT SMRs, service environment (such as neutron irradiation and corrosion) may significantly change material design limits

- Neutron irradiation affects code limits for load-controlled stresses. For example, the temperature & time-dependent stress intensity limit, S_t, is defined as the lesser of:

(a) 100% of the average stress required to obtain a total strain of 1%

(b) 80% of the minimum stress to cause initiation of tertiary creep

(c) 67% of the minimum stress to cause rupture

Thus, S_t value could be lower under neutron irradiation

Environment effect on material design criteria (2/3)

- Neutron irradiation also affects other material design parameters, such as stress-to-rupture curves, isochronous stress-strain curves, and deformation-controlled quantities
- In HT corrosion process, alloy elements can be preferentially dissolved or transported into coolants; impurities can diffuse into material (e.g., carburization and decarburization in HTGR and SFR). These element transfer processes can not only affect material time-dependent design parameters (as neutron irradiation does), but also affect time-independent allowable stress intensity Sm
- Synergistic effect of neutron irradiation and corrosion can be more complicated

Environment effect on material design criteria (3/3)

- Design parameters or the environment effect on design parameters are:
  - Essential (No way to use a material without reliable design limits/parameters)
  - Significant (Environment effect could be significant)
  - Urgent (A need in design assessment rather than in a late stage of fitness-for-service assessment)
  - Very difficult to obtain (very resource and time consuming)
- Little quality information is currently available to support such assessment

Properties of Full-size Component

- ❑ Information obtained from small sample tests may not represent full-size component behavior, for example:
  - Sample size and shape affects the graphite oxidation
  - Neutron irradiation may introduce high internal stress or cracks into full size graphite component (but not small samples)
  - Complicated weld designs of SMR components can make the code design parameters (stress rupture factor and deformation limits) inapplicable
- ❑ To date, very limited component tests are available for supporting the assessment of SMR component behavior

Coating and Cladding

- Vendors may use cladding or coating (such as Fe/Al, Ni, carbides, nitrides, borides, phosphides, and refractory coatings) to improve corrosion resistance
- However, coating affects heat transfer capability. Thermal cycling can cause coatings to crack or delaminate. Thin coatings reduce the tendency of cracking or delamination but are more vulnerable to imperfections; radiation-enhanced-intermixing could also be a significant issue for thin coatings
- The applicability of coating or cladding to SMR components needs to be assessed

Mechanical Connection

- ❑ Chemical compatibility of welds need to be investigated
- ❑ Gaskets may be challenging due to the tendency to develop leaks over time:
  - Gasket degradation (e.g. nitriding/carburizing in HT gas, corrosion in molten salt or liquid metals)
  - Bolt creep
  - Sealing-surface deformation
  - If gaskets are used for mechanical connection, systematic qualification needs to be performed

Conclusions

- International and domestic codes and standards, technical guidance, and research works provide a basis for CNSC review of SMR designs
- More data is still required for fully understanding material behavior in HT SMRs; this could be a challenge for deployment of some SMRs
- Material degradation research projects are being conducted by vendors and research Institutions to address these gaps

Aging Management Considerations for Advanced SMRs and Non-LWRs

Mr Chris WAX

EPRI

Aging Management Considerations for Advanced SMRs and Non-LWRs

Aging Management Considerations in Mechanical Codes and Standards

Chris Wax

Principal Technical Leader, EPRI ANT – AMM

June 29, 2023

Agenda

- EPRIs Advanced Manufacturing and Materials Technical Focus Area
  - The Challenge for Advanced Reactors
- High Temperature Reactor Considerations
- Materials Management Programs for Advanced Reactors
  - Reliability and Integrity Management Programs
  - Materials Degradation Matrices and Issue Management Tables
- Relevant Activities Ongoing at EPRI

EPRI's Advanced Manufacturing and Materials Technical Focus Area

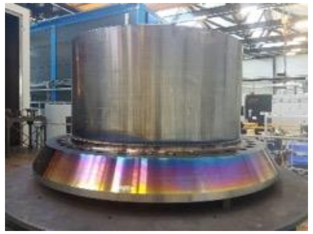
Advanced Manufacturing and Materials

GOAL & VALUE

Identify, develop, qualify, & implement more economical manufacturing, inspection & new materials that enable:

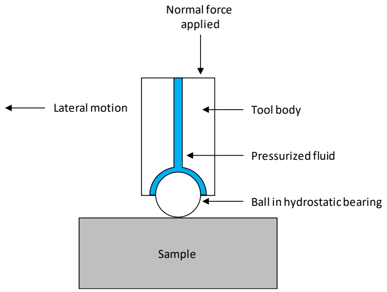
Higher Quality Components | Reduced Lead Times | Alternative Supply Chains | Cost Competitiveness | Enable Deployment

- Evaluate, Qualify, Demonstrate Advanced Manufacturing Methods
- Additive Manufacturing
- PM-HIP
- Advanced Welding
- Mechanical Connections
- Advanced Cladding



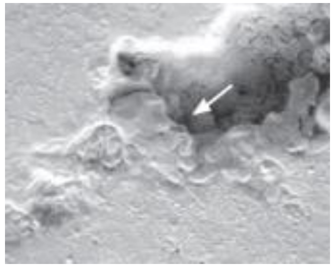
Advanced Manufacturing

- Mitigation Techniques
- Material Management
- Reliability & Integrity Management
    - ASME Section XI, Division II
- Enhanced Specifications



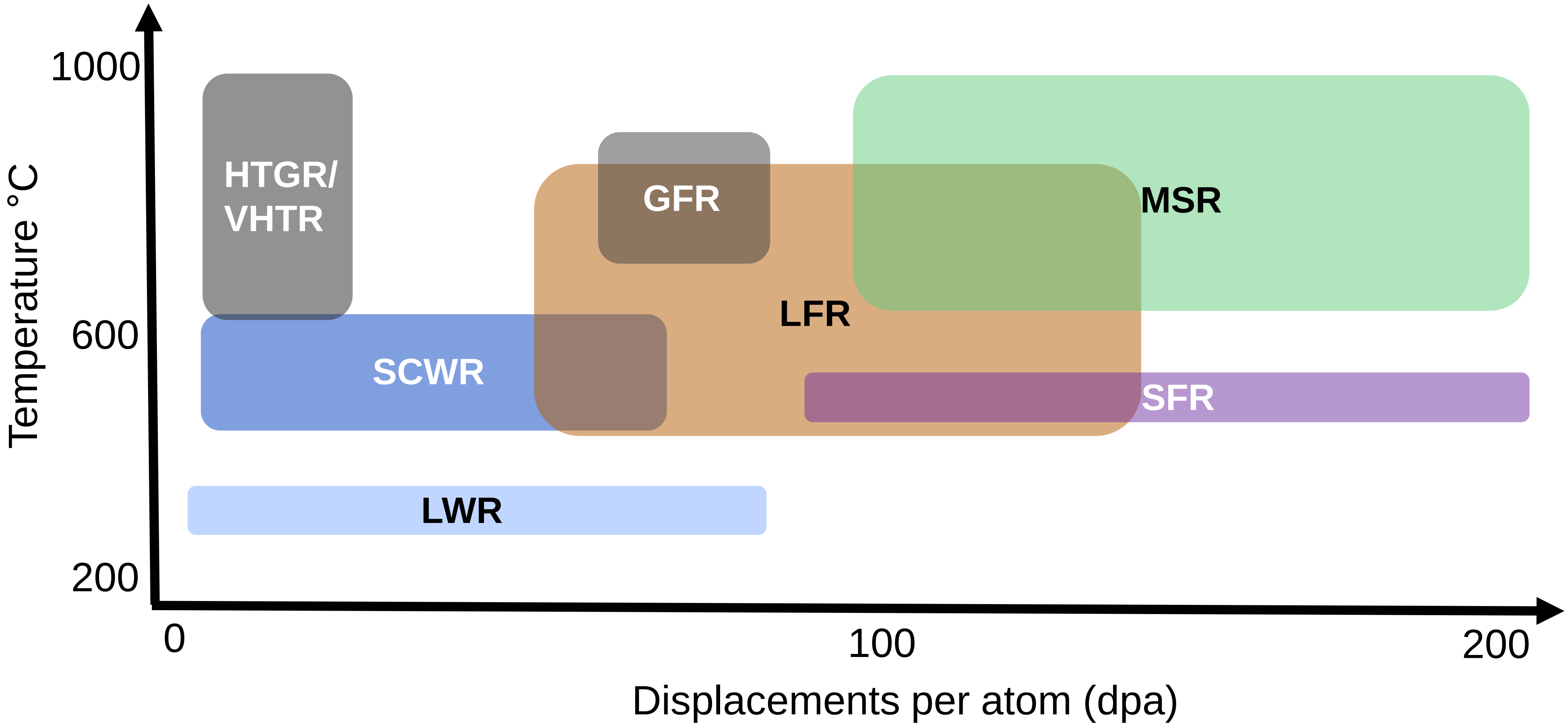
Material Performance and Inspection

- AR Materials Development
- Materials Qualification
- Degradation Mechanisms (MMM/MDM)



New Materials Development

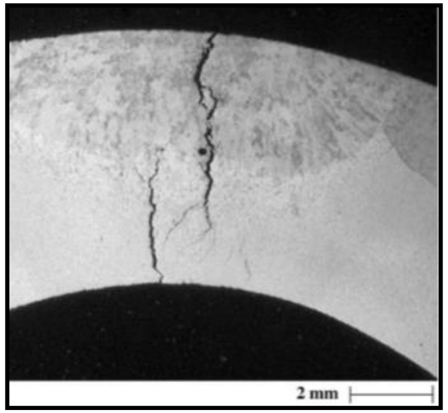
New Coolants, Conditions = New Material Challenges



Adapted from Y. Guerin, G. S. Was, and S. J. Zinkle. *Materials Challenges for Advanced Nuclear Energy Systems*. MRS Bulletin V34(1), (2009).

Unanticipated materials challenges in first-of-a-kind applications and demonstrations in power generation

New Environments



Materials selection & environmental effects

New Materials

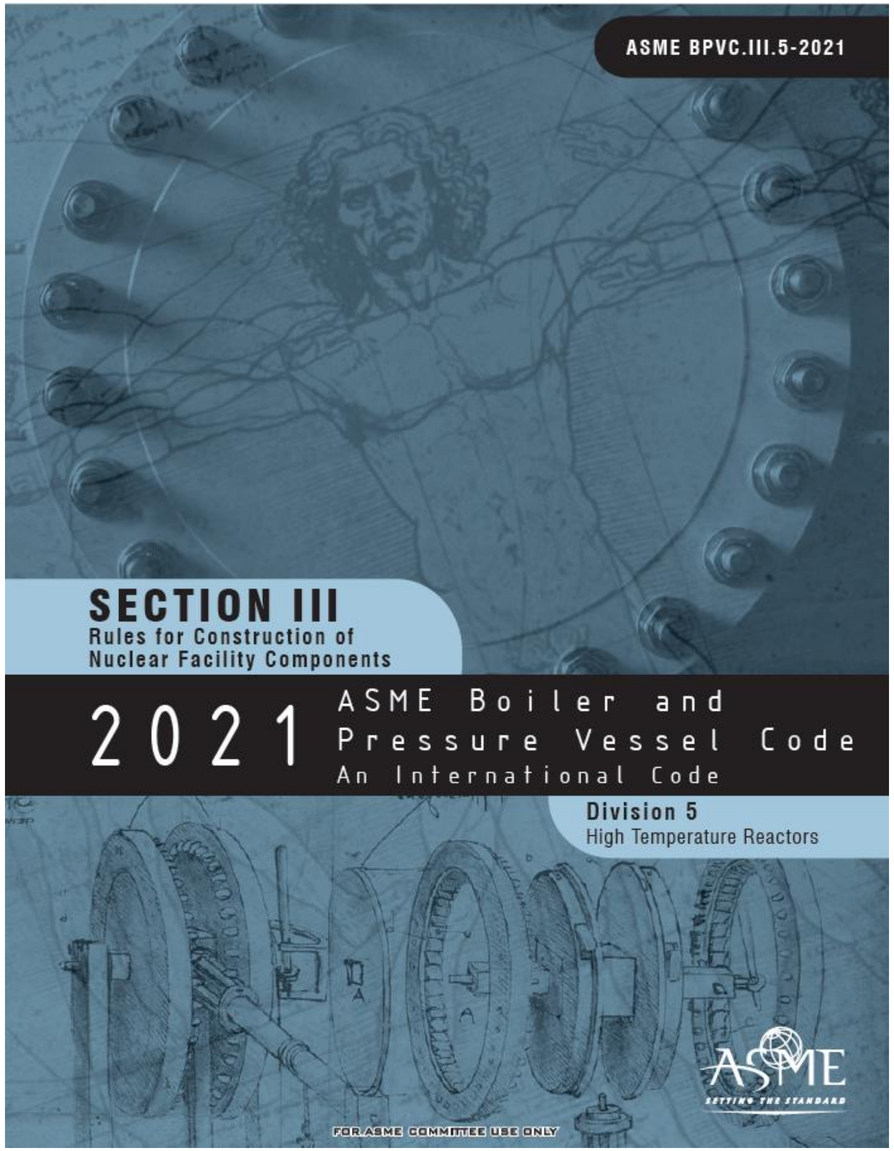
Manufacturing and fabrication challenges

Codes, Standards, and Specifications

Materials research during <u>Design</u> and continuing through <u>Demonstration</u> reduces overall project risk

High Temperature Reactor Considerations

ASME Boiler and Pressure Vessel Code



Section III

Rules for Construction of Nuclear Facility Components

Division 5

High Temperature Reactors

Subsection HB

Class A Metallic Pressure Boundary Components

- Subpart A — Low Temperature Service
- Subpart B — Elevated Temperature Service

Section III Division 5 Scope – Elevated Temp. Class A

- Design analysis shall consider **time-dependent** material properties and structural behavior by guarding against:
  - Ductile rupture (short term loadings)
  - Creep rupture (long term loadings)
  - Creep-fatigue
  - Gross distortion (incremental collapse and ratcheting)
- Brief guidelines are also provided for:
  - Loss of function due to excessive deformation
  - Buckling due to short-term loadings
  - Creep buckling due to long-term loadings

Materials for Class A Metallic Pressure Boundary Components - Elevated Temperature Service

- Only 6 base materials approved:

304 SS	316 SS	Alloy 800H
2.25Cr-1Mo	9Cr-1Mo-V	Alloy 617

- Limited service for low alloy steels (508, 533) via HBB-II
- For comparison, >100 alloys available for Sections I (Power Boilers) and VIII (Pressure Vessels)
- Why so few for Section III-5, Class A?

Table HBB-I-14.1(a)
Permissible Base Materials for Structures Other Than Bolting

Base Material	Spec. No.	Product Form	Types, Grades, or Classes
Types 304 SS and 316 SS [Note (1)], [Note (2)], [Note (3)]	SA-182	Fittings & Forgings	F 304, F 304H, F 316, F 316H
	SA-213	Smls. Tube	TP 304, TP 304H, TP 316, TP 316H
	SA-240	Plate	304, 316, 304H, 316H
	SA-249	Welded Tube	TP 304, TP 304H, TP 316, TP 316H
	SA-312	Welded & Smls. Pipe	TP 304, TP 304H, TP 316, TP 316H
	SA-358	Welded Pipe	304, 316, 304H, 316H
	SA-376	Smls. Pipe	TP 304, TP 304H, TP 316, TP 316H
	SA-403	Fittings	WP 304, WP 304H, WP 316, WP 316H, WP 304W, WP 304HW, WP 316W, WP 316HW
	SA-479	Bar	304, 304H, 316, 316H
	SA-965	Forgings	F 304, F 304H, F 316, F 316H
	SA-430	Forged & Bo	
Ni-Fe-Cr (Alloy 800H) [Note (4)]	SB-163	Smls. Tubes	
	SB-407	Smls. Pipe &	
	SB-408	Rod & Bar	
	SB-409	Plate, Sheet,	
	SB-564	Forgings	
$2^1/_4$Cr-1Mo [Note (5)]	SA-182	Forgings	
	SA-213	Smls. Tube	
	SA-234	Piping Fittin	
	SA-335	Forg. Pipe	
	SA-336	Fittings, For	
	SA-369	Forg. Pipe	
	SA-387	Plate	
	SA-691	Welded Pipe	
9Cr-1Mo-V	SA-182	Forgings	
	SA-213	Smls. Tube	
	SA-335	Smls. Pipe	
	SA-387	Plate	

NOTES:
(1) These materials shall have a minimum specified room temperature … cified carbon content of 0.04%.
(2) For use at temperatures above 1,000°F (540°C), these materials m… minimum temperature of 1,900°F (1 040°C) and quenching in wat…
(3) Nonmandatory Appendix HBB-U provides nonmandatory guidelin… mance in certain service applications.
(4) These materials shall have a total aluminum-plus-titanium content… perature of 2,050°F (1 120°C) or higher.
(5) This material shall have a minimum specified room temperature … room temperature ultimate strength of 60,000 psi (414 MPa), a … 85,000 psi (586 MPa), and a minimum specified carbon content o…
(6) The material allowed under SA-234 shall correspond to one of:
(a) SA-335, Grade P 22
(b) SA-387, Grade 22, Class 1
(c) SA-182, Grade F 22, Class 1 in compliance with Note (4).

ASME BPVC.CC.NC-2021

CASE
N-898

Approval Date: October 6, 2019

Code Cases will remain available for use until annulled by the applicable Standards Committee.

Case N-898
Use of Alloy 617 (UNS N06617) for Class A Elevated Temperature Service Construction
Section III, Division 5

Inquiry: May 52Ni–22Cr–13Co–9Mo, Alloy 617 (UNS N06617) be used at elevated temperatures in the construction of components conforming to the requirements of Section III, Division 5, Subsection HB, Subpart B "Elevated Temperature Service"?

Reply: It is the opinion of the Committee that 52Ni–22Cr–13Co–9Mo, Alloy 617 (UNS N06617) may be used in the construction of components conforming to the requirements of Section III, Division 5, Subsection HB, Subpart B "Elevated Temperature Service," provided the following requirements are met:

(a) The modifications and additions to the rules provided in Subsection HB, Subpart B defined in this Code Case shall be met.

(b) The service temperature shall be limited to 1,750°F (954°C)[1] and below.

(c) Service time shall be limited to 100,000 hr.

(d) All other applicable requirements of Section III, Division 5, Subsection HB, Subpart B shall be met.

(e) This Case number shall be listed on the Data Report Form for the component.

This Code Case was written to be used in conjunction

References within Section III, Division 5 to figures and tables in Mandatory Appendix HBB-I-14, design fatigue curves or isochronous stress–strain curves should be extended to include corresponding figures and tables for Alloy 617 within this Code Case.

Thermal expansion, thermal diffusivity, and thermal conductivity are not currently contained in Section II for Alloy 617 (UNS N06617). Values for these properties are shown in Tables TE-4 and TCD of Nonmandatory Appendices A and B, respectively, of this Code Case. Elastic modulus values for Alloy 617 are currently included in Section II, Part D (Table TM-4) in U.S. Customary units for temperatures up to 1,500°F and in SI units for temperatures up to 850°C, but the temperature range must be increased to 1,750°F (954°C) to cover the maximum service temperature of Nonmandatory Appendix C of this Code Case. Elastic modulus values are shown in Table TM-4 of this Code Case.

ARTICLE HBB-2000 MATERIAL

HBB-2100

HBB-2160 Deterioration of Material in Service

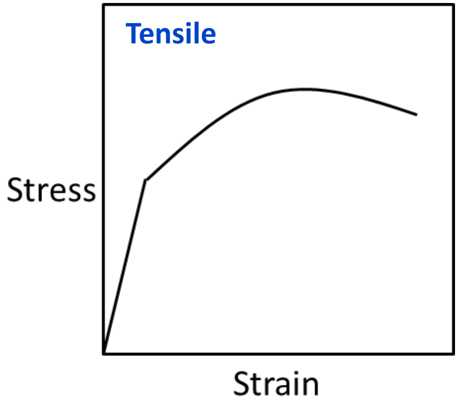
(d) Long-time, elevated temperature service may result

Answer: material property data!

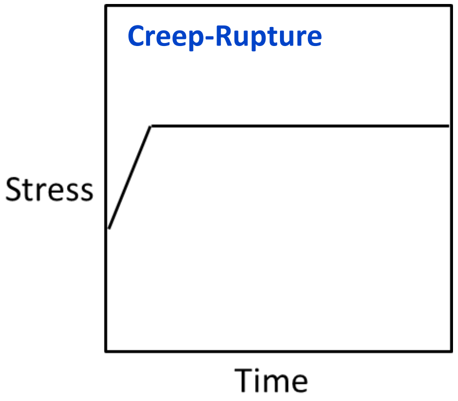
Design Parameters and Associated Mechanical Tests for Section III Div. 5 (High Temp. Class A)

Design Parameters	Required Testing			
	Tensile	Creep-Rupture	Fatigue	Creep-Fatigue
Strength Values (Su, Sy)	X			
Allowable Stresses (S0, Sm, St, Smt, Sr)	X	X		
Stress Rupture Factor for Welds (R)		X		
Thermal Aging Factors	X			
Cold-Forming Limits	X	X		
Isochronous Stress-Strain Curves	X	X		
Fatigue Design Curves			X	
Creep Fatigue Interaction Diagram				X
…	…	…	…	…

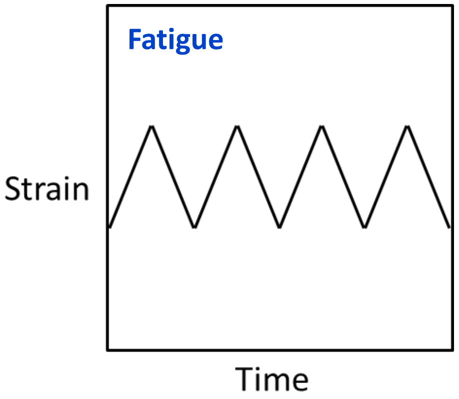
Time-Independent



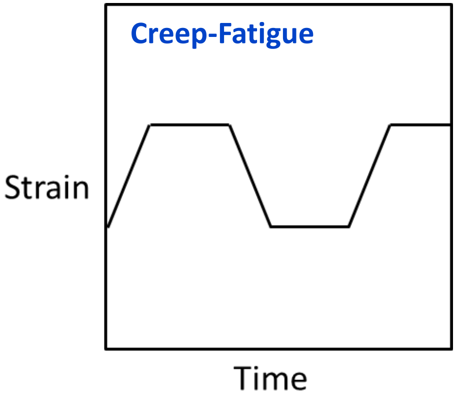
Time-Dependent



Cyclic



Time-Dependent + Cyclic



Materials Challenges for Advanced Reactors

Code Qualification

Limited number of Code Qualified alloys for nuclear service at high temperature

Can take a DECADE to fully qualify a new alloy

$

Just because the material is **Code Qualified** doesn't mean it is **approved** for Nuclear Service

US NRC issued Interim Staff Guidance (DANU-ISG-2023-01) discussing Materials Compatibility for Non-Light Water Reactors

Environmental Effects

Owner/Operator has the responsibility to demonstrate to regional regulator that the effects on structural failure modes are accounted for in their specific reactor design

Irradiation Data | Corrosion Data | Thermal Performance | Environmental Testing

Qualification ≠ Licensing ≠ Deployment

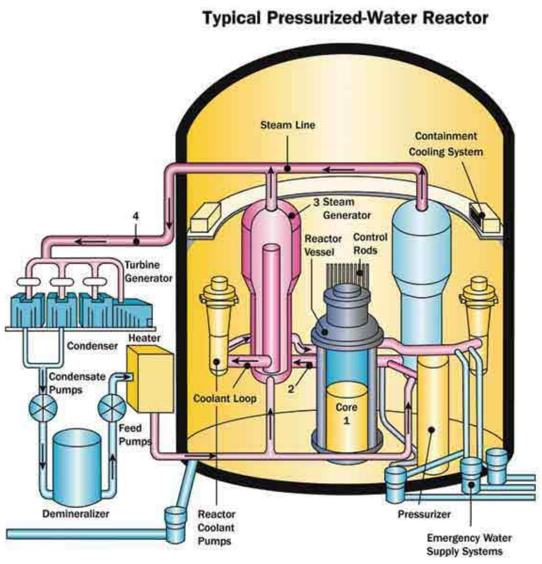
Materials Management

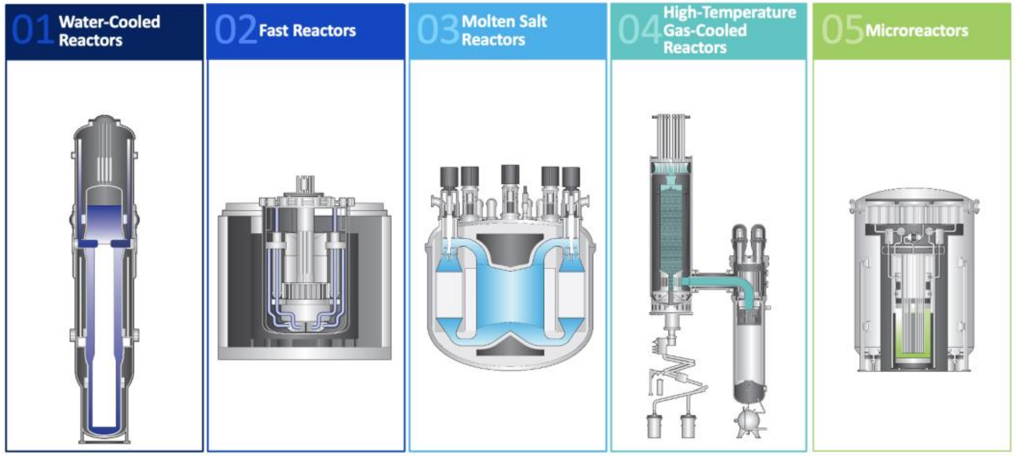
Critical data and understanding required for materials management to ensure safe and efficient long-term operation of the new fleet

Materials Management Programs for Advanced Reactors

Materials Management Program Background

- For the operating, light-water reactor fleet, license holders use a deterministic approach to assure as-designed safety margins through a selection of mandated examinations and tests
  - ASME Section XI, Division I
    - Developed and evolved with over 40 years of operating experience guiding the requirements
  - NEI 03-08 Materials Initiative
    - Industry requirement, endorsed by NRC, to proactively manage aging and degradation of materials
- To support a broader range of reactor specifications/designs, a performance-based alternative approach to define an appropriate program of examinations and tests is now available
  - ASME Section XI, Division II

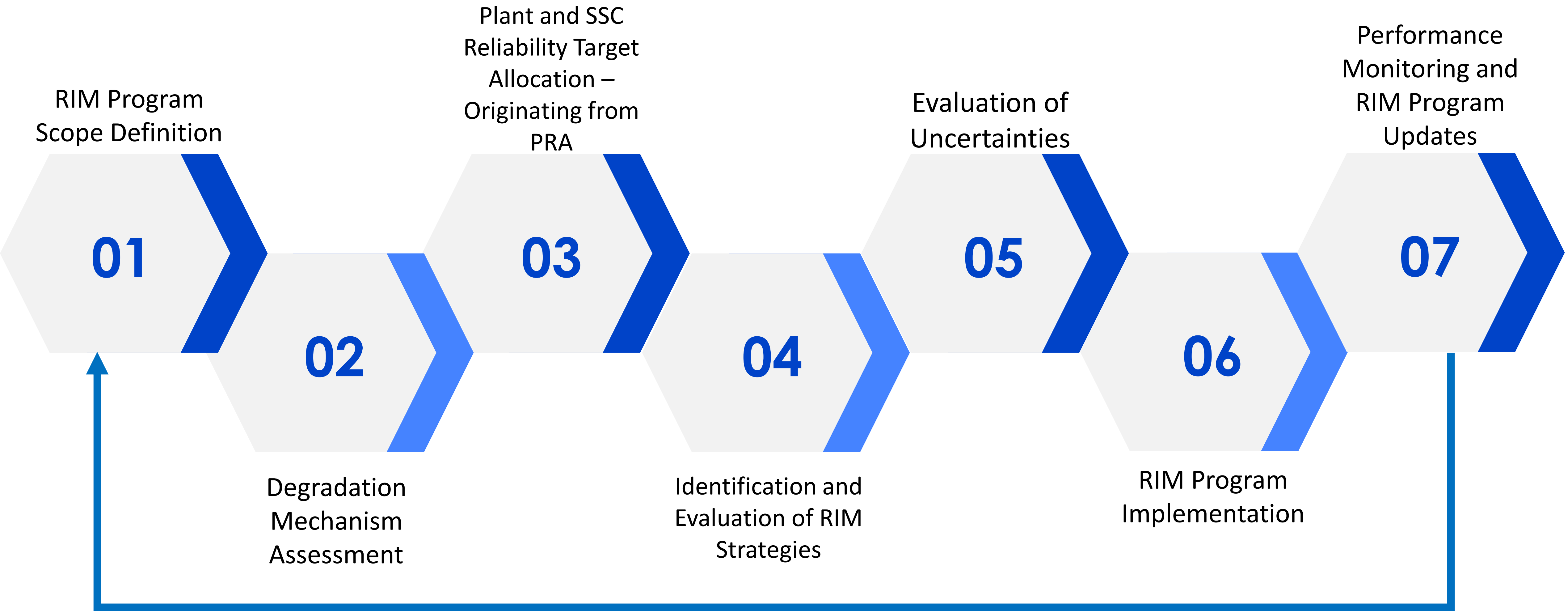




The legacy Section XI, Division I ISI program requirements may be a poor fit for many new designs

Advanced Reactor Materials Management - The RIM Process

ASME Section XI, Division II – Article RIM-1, RIM-1.1 Scope



Materials Management Program (MMP) Summary

Why? During the Licensing process, the creation of an MMP to ensure reliability, integrity, and longevity of critical systems, structures, and components (SSCs) will be required

Reliability and Integrity Management Program

Documented program and process to ensure SSC reliability and integrity by reduction of impacts due to materials degradation through periodic monitoring or examinations

What is it?

Materials Degradation Matrix

Documents the state of knowledge (world-wide) for degradation mechanisms as applied to the subject materials and environments

- Fundamental mechanistic understanding
- Inspection capabilities
- Mitigation opportunities

Issue Management Tables

- Assesses the consequences of failure
- Identifies gaps in inspection, mitigation, repair, and replacement
- Gap prioritization guides tactical and strategic R&D plans
- Supports leveraged research collaboration to benefit all participants

What is the underlying technical basis?

In order to know where to inspect, what to inspect for, and how often to inspect, an understanding of the degradation mechanisms and how they manifest is pivotal

Periodic Assessment and Update

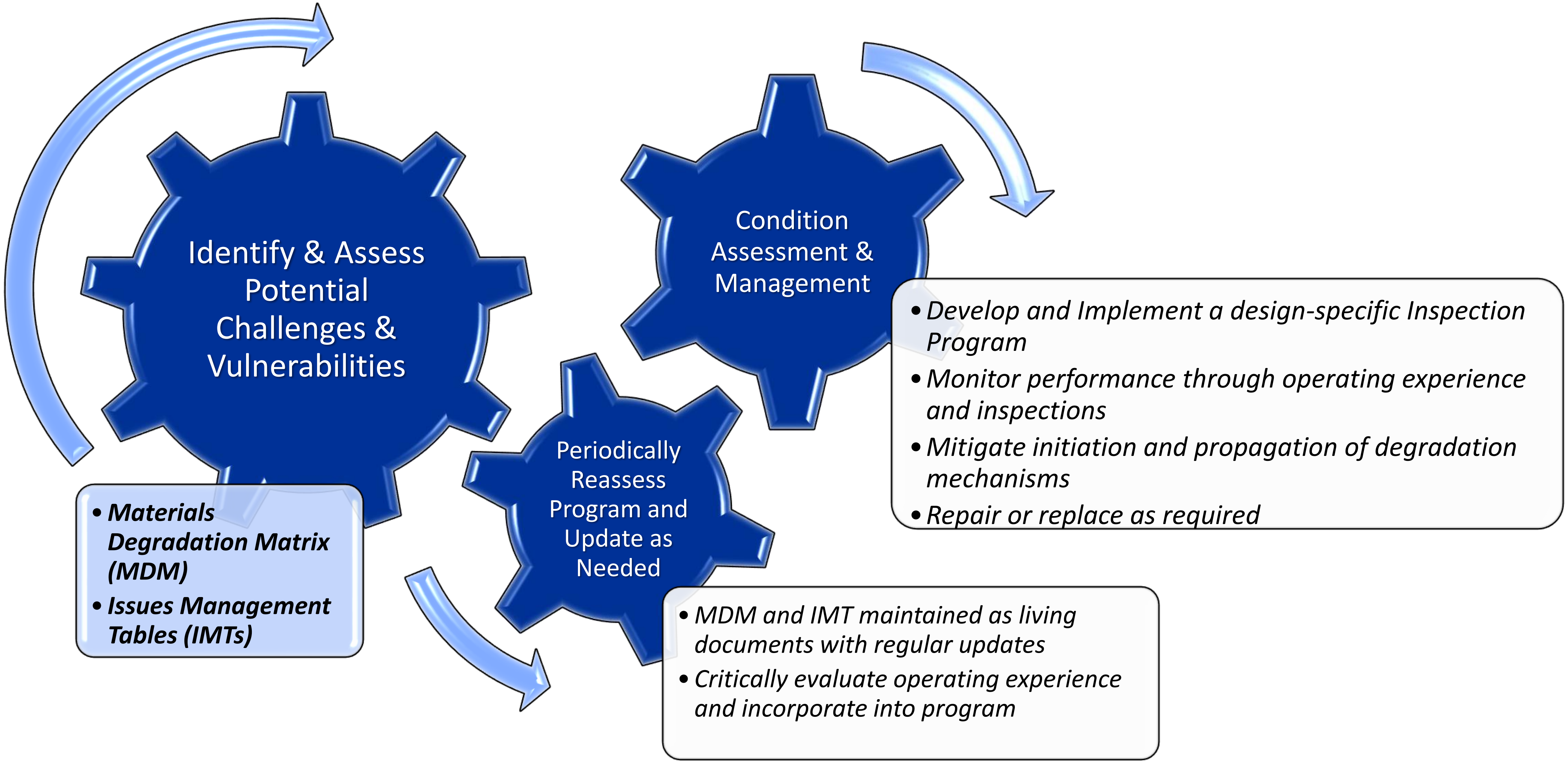
Periodic program revisions will be required as information is gathered through

- New research and development
- Operating experience
- Advanced monitoring or NDE techniques

A Living Program

Materials Management - *Integrated and Strategic*

Building off successes from the light water fleet, EPRI is applying a similar approach to materials degradation assessments for the advanced reactor fleet



Systematic Approach Summarized

Materials Degradation Matrix
Rev. 4
3002013781

Component Material Degradation Tables

Degradation Mechanism Explanatory Notes

(With R&D Status Notes)

Component Management Tables

R&D Gaps

(with Management Issue, Description/R&D Status and Needed Knowledge)

Every material, every potential degradation mechanism and status of knowledge

- Mapped to 80 years of operation
- Covering BWR, PWR, CANDU and VVER
- In Progress: LW-SMRs & ANLWRs

Issue Management Tables
PWR: 3002018255
BWR: 3002018319
VVER: 3002021033

Every component/material, failure modes, mitigation, repair/ replacement, I&E Guidance → Knowledge Gaps identified and prioritized

- Covers BWRs, PWRs and VVERs
- In Progress: CANDU
- Planned: LW-SMRs & ANLWRs

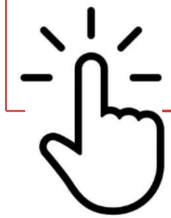
Materials Degradation Matrix (MDM)

- The EPRI Materials Degradation Matrix summarizes the state of industry knowledge regarding the degradation mechanisms and related research activities
- Degradation is defined for normal operating conditions
- Color codes:
  - Green → well understood, no R&D necessary.
  - Yellow → sufficient R&D in progress to address gaps in reasonable timeframe.
  - Orange → insufficient R&D ongoing/planned to address gaps in reasonable timeframe.
  - Blue → insufficient data exists to establish degradation mode applicability.

Table 4-1
PWR Primary Pressure Boundary (1)

PWR Primary Pressure Boundary

MATERIAL	DEGRADATION MODE													
	Corrosion				Wear	SCC		Fatigue		Reduction in Fract Properties		Irradiation Effects		
	Wstg.	Pitting	FAC	Foul	Wear	IG / TG	IA	HCF	EAF	Th	Env	Emb	VS	IC / SR
C&LAS: Base Metal & HAZ	Y p1-1a	N	N	N	N	Y p1-6a	Y p1-7a	N	Y p1-9a	? p1-10a	Y p1-11a	Y p1-12a	N	N
C&LAS: Welds	Y p1-1b	N	N	N	N	Y p1-6b	Y p1-7b	N	Y p1-9b	? p1-10b	Y p1-11b	Y p1-12b	N	N
SS: 300 Series SS Base Metal & HAZ	N	Y p1-2c	N	N	N	Y p1-6c	N	Y p1-8c	Y p1-9c	N	Y p1-11c	N	N	N
SS: 300 Series SS Welds & Clad	N	Y p1-2d	N	N	N	Y p1-6d	N	Y p1-8d	Y p1-9d	Y p1-10d	Y p1-11d	Y p1-12d	N	N
Cast Austenitic Stainless Steel	N	N	N	N	N	Y p1-6e	N	Y p1-8e	Y p1-9e	Y p1-10e	Y p1-11e	N	N	N



NOTE ID	EXPLANATORY NOTE
	PWR PRIMARY PRESSURE BOUNDARY AND STEAM GENERATOR DIVIDER PLATE
p1-1a p1-1b	Gaps in the channel head cladding can permit low low-alloy steel to come into contact with primary water. Significant corrosion could potentially occur as a result of exposure to oxidizing conditions, occurring most likely after shutdown and draining activities due to concentration of a puddle of primary water by evaporation. There is a general understanding of how corrosion can progress and the overall timeframe. There are no fundamental R&D gaps. Any gaps related to management of this issue should be addressed within the PWR IMT report. [R&D Status = GREEN] **References:** Other: [4-2]

2004 – Initial Issue: Information available on a limited basis; not in Technical Report format

2008 – Rev 1: Technical Report format, MDM adopts multi-level structure

2010 – Rev 2: LTO considerations addressed

2013 – Rev 3: Addition of CANDU

2018 – Rev 4: Addition of VVER

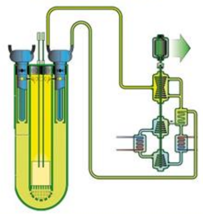
2023 – Rev 5: Foundation for SMR MDM

Relevant Activities Ongoing at EPRI

Coordination

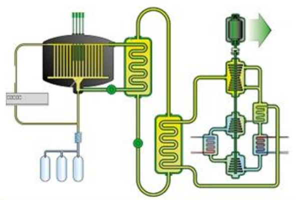
Sodium Fast Reactors
3002016949

Lead Fast Reactors
3002016950



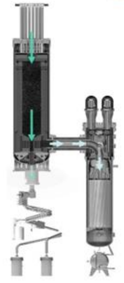
EPRI Material Gap Analyses

Molten Salt Reactors
3002010726



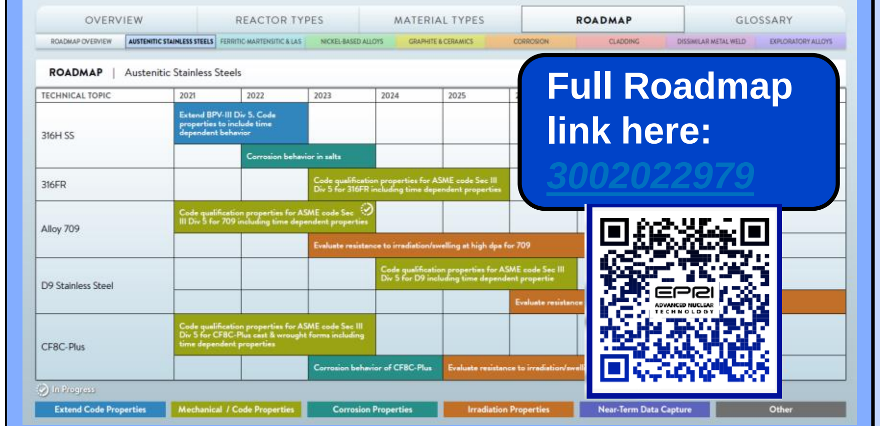
High-temperature Gas Reactors

3002015815



Prioritization

Advanced Reactor Materials Development Roadmap



Full Roadmap link here: [3002022979]

Material Degradation Matrices

Collaboration

<u>Industry Advisory Groups & Expert Panels</u>
(all stakeholders)

Willingness to share needs, challenges, & shared interests is critical

Advanced Reactor Materials Initiative (ARMI)

Details and Contact

- Project is open to EPRI and non-EPRI members

Marc Albert

- malbert@epri.com

Overview

- A global collaborative multi-year project focused on developing and deploying new and existing materials to support the licensing and long-term operation of advanced non-light water reactors (ARs).

Problem Statement

- Some material property data and environmental compatibility are available to support conceptual design, but there are many gaps in the data to evaluate and qualify long-term service suitability.
- Research to address these gaps can be expensive and time-consuming – too resource intensive for one org

Objectives and Scope

- Targeted material testing, demonstration, qualification, and supply chain development to close technology and knowledge gaps.
  - Extend operating envelope (temperature and lifetime) for materials already qualified
  - Develop required data to add new materials to codes & standards (ASME, RCC-M, etc.)
  - Generate, compile, and curate environmental data (irradiation, corrosion, etc.) to support reactor design, licensing, reduce project risk and long-term operation.
  - Develop accelerated qualification approaches & technical basis to de-risk reactor licensing & deployment
  - Conduct manufacturing and fabrication demonstrations to establish an available supply chain

ARMI brings together AR developers, national labs, and the material & component suppliers with a coordinated approach to ensure the data, material understanding, and supply chain are well established for advanced reactor deployment and operation.


Together...Shaping the Future of Energy®

Questions ? Answers !

Coffee break

Panel discussion

Conclusions

Dr Sangmin LEE

Korean Institute for Nuclear Safety

End of day 2

Thank you　ありがとう

www.oecd-nea.org