Department of Architecture
School of Engineering
The University of Tokyo
Building Material Laboratory

Summary of recent progress of JCAMP and Hamaoka project

Ippei Maruyama[*1]

Toshiaki Kondo [*2], Shohei Sawada[*2], Osamu Kontani[*2], Kiyoteru Suzuki[*3], Takafumi Igari[*4], Takahiro Ohkubo[*5], Kenta Murakami[*1]
with the aid of CVR team

Abudushalamu Aili[*6], Kazuhiro Yokokura[*7], Yoshito Umeki[*7]

*[1]: The University of Tokyo, *2: Kajima Corporation, *3: Mitsubishi Research Institute, Inc.,*4: MRI Research Associates, Inc., *5: Chiba University, *6: Nagoya University, *7: Chubu electric company

Background

1.NRA Project　Previous Study（until JFY2016）

Purpose

- Confirmation of the scientific validity of the current reference values for irradiation doses.
- Securing the scientific basis for the operator's application for more than 40 years, etc.

Outcome

- Development of Scientific Basis Data for Concrete Irradiation Deterioration
- Revision of reference values for concrete degradation due to neutron fluence
  $1\times10^{20} \rightarrow 1\times10^{19}\mathrm{n/cm^2}$

Remaining Issue

- Rational evaluation methods and scientific basis data for irradiation degradation according to the actual compositional conditions of concrete

2.METI Project　（until JFY2022）

Purpose

- Improvement of regulatory accountability in special inspections for operations exceeding 40yrs
- Expansion of knowledge and international contribution through the collaboration based on the joint research

Outcome

- Data set of irradiated concrete aggregates in Japan
- Develop the procedures of integrity evaluation for irradiated concrete and RC members.

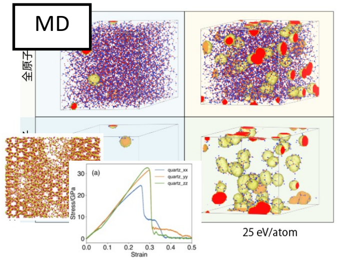
Remaining Issues

- Development of a method for evaluating the degradation of concrete that takes into account the irradiation rate effects and the scientific basis for this method.
- Achievement of international consensus on irradiation degradation of concrete

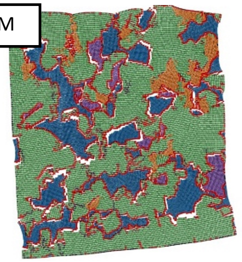
Safety / integrity evaluation process

Calculation

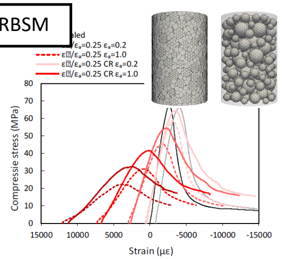
MD



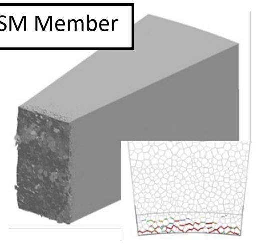
FEM

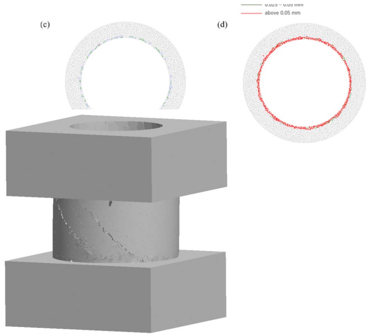


RBSM



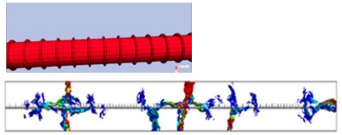
RBSM Member





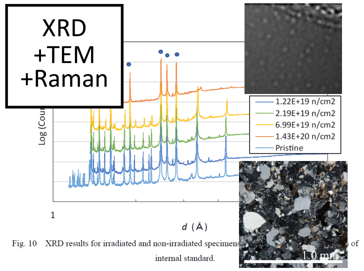
Mineral / atomic level → Aggregate level → Concrete level

RBSM Member

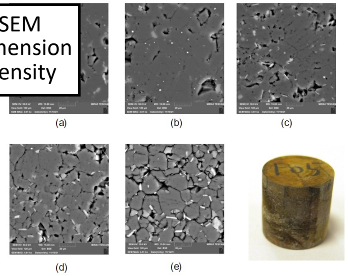


Experiment /Validation

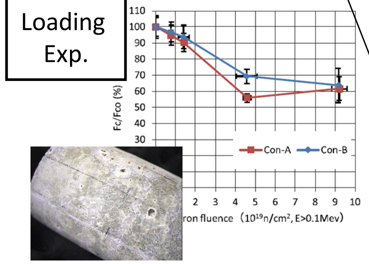
XRD +TEM +Raman

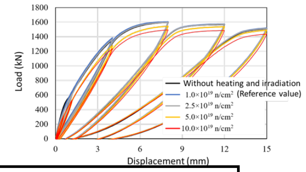


SEM Dimension Density



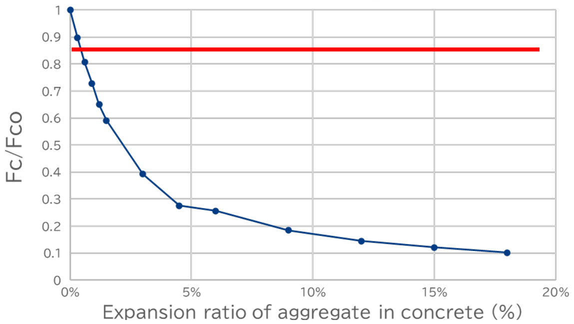
Loading Exp.



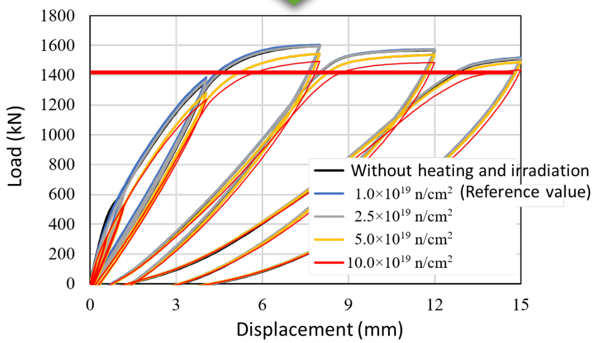


Component → Structure

Neutron fluence threshold value



Concrete Strength Evaluation



Structural performance evaluation

Irradiation experiments

Information of concrete aggregates used in Japanese PWR were collected and representative aggregates + pure phases were selected for irradiation.

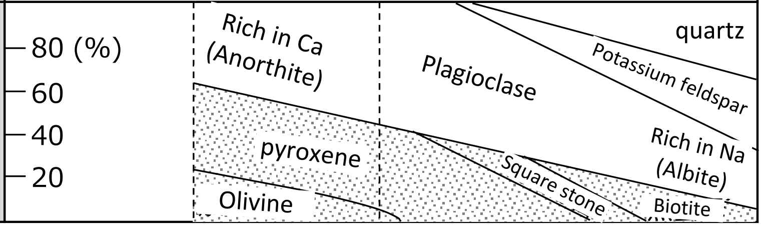
Specimen ID, type and origin

ID	Rock type / Mineral	Origin / Manufacturer
A	Synthetic quartz	Tokiwa Tech Co., Ltd.
B	Quartz glass	Shinetsu Quartz Co., Ltd.
C	Plagioclase	Itoigawa, Niigata
D	Alkaline feldspar	India
F	Granite	Takamatsu, Kagawa
G	Altered tuff	Kasugai, Aichi
H	Andesite	Satsumasendai, Kagoshima
J	Basalt	Karatsu, Saga
K	Peridotite	Samani, Hokkaido
L	Sandstone	Tsuruga, Fukui

Classification of igneous rocks

Blue: coarse aggregate used for PWR in Japan
Red: additionally selected rocks

		Ultramafic rock	Basic rocks	Intermediate rock	Acidic rock
Mottled tissue	Volcanic rock		Basalt	Andesite	Rhyolite
Equal grain texture	Plutonic rock	Peridotite	Gabbro	Diorite	Granite
SiO_2(weight%)		45% ◄► 52% ◄► 66%			
color		dark ◄► Whitish			
specific gravity		About 3.2 ◄► About 2.7			
Mineral composition					




Synthetic quartz


Quartz glass


Plagioclase


Alkaline feldspar


Granite


Altered tuff


Andesite

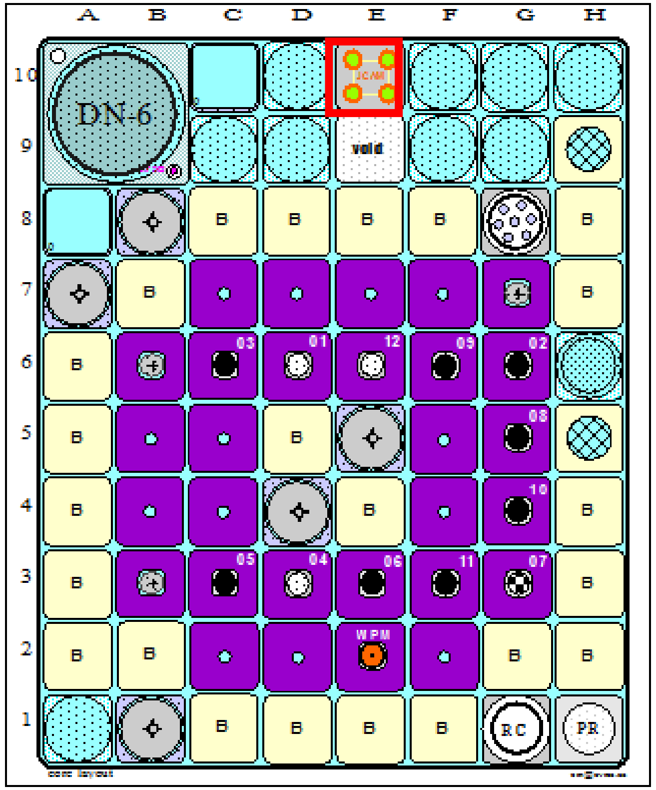

Basalt


Peridotite

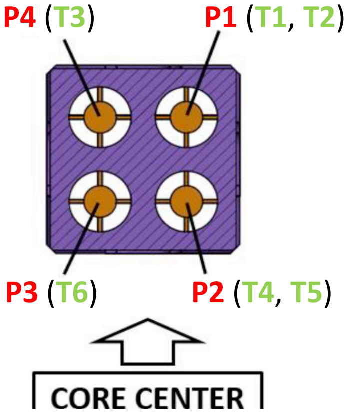

Sandstone

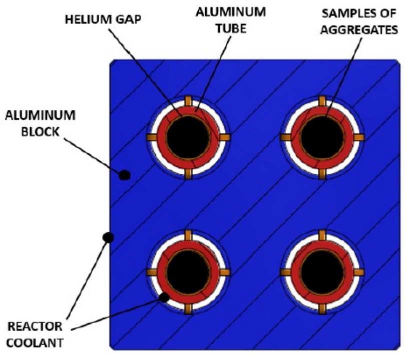
Reference: Harutaka Sakai, "Introduction to Earth Sciences, 2nd Edition, Planetary Earth and Atmosphere / Ocean Systems", Tokai University Press

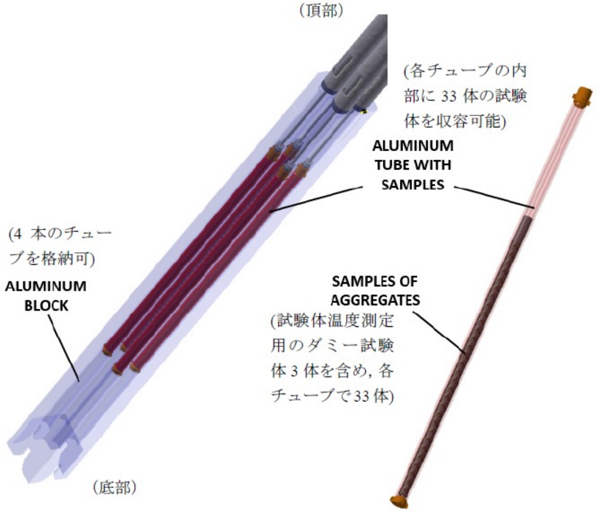
Irradiation experiment
Low flux (4.77 n/cm^2/s) + Medium flux(8.70 n/cm^2/s)

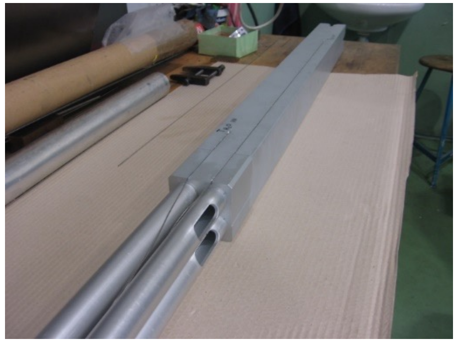


LVR-15

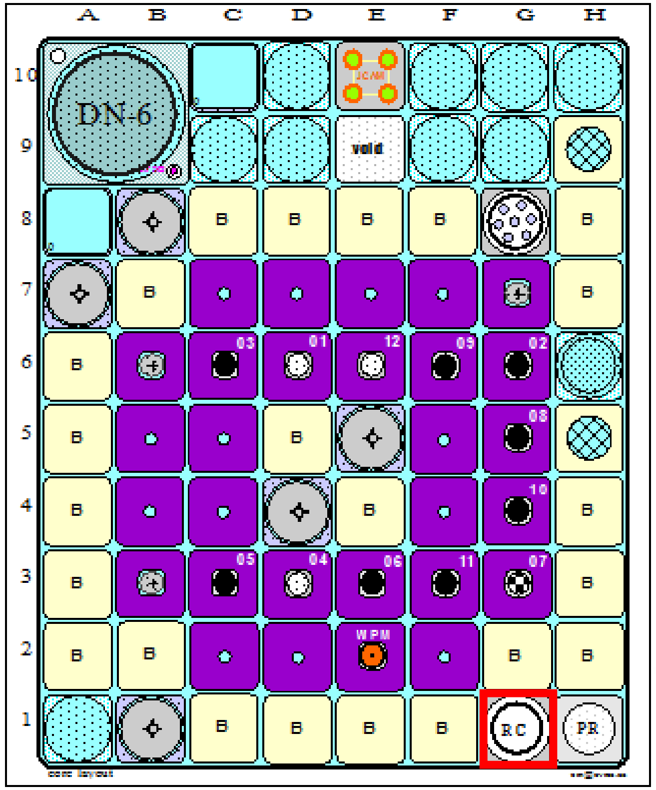


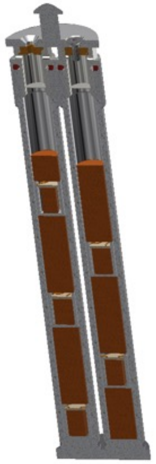


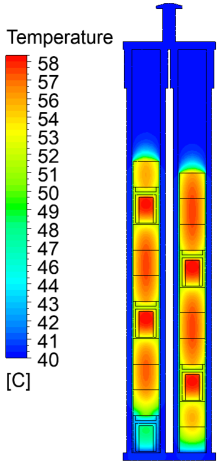




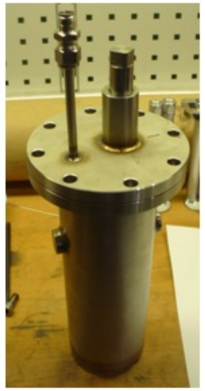
Irradiation experiment: High flux (18.7 n/cm²/s)







Reactor power (MW)	Cooling water temperature	Temperature – high (°C)	Temperature – low (°C)
1.1	12.2	13.3	13.0
2.0	14.8	17.3	16.9
3.5	20.4	25.3	23.5
6.9	33.1	42.2	39.7
8.8	45.1	56.7	52.8
9.0 (estimate)	44.1	56.1	52.3
9.7 (estimate)	47.1	60.0	55.9
Change of cooling circuit heat removal to full			
9.0	37.00	48.9	45.0
9.7 (estimate)	39.46	52.3	48.1



Sample capsule	Estimated temperature maximum (°C)
T8	57.22
T7	55.01
T9	53.25

Background data: IFE-irradiated sample PIE results

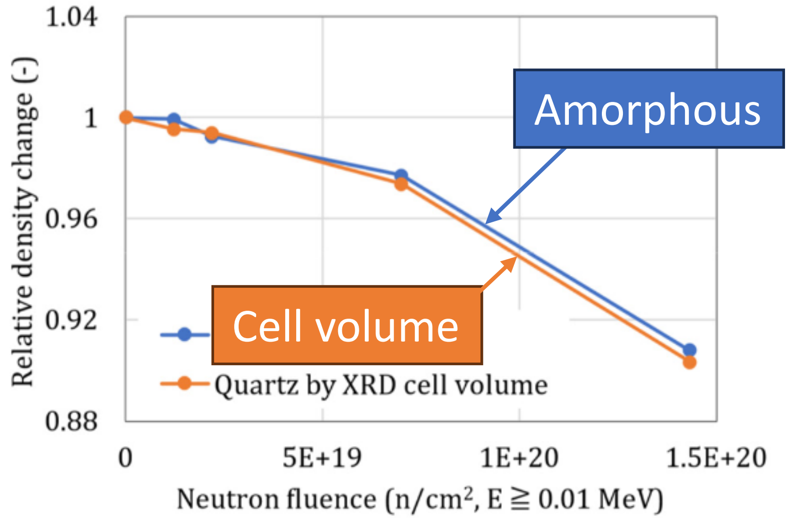


Fig. 12 Relative density change of α-quartz and amorphous phase calculated by assuming that other phases do not contribute to the volume change of aggregate.

Maruyama, I., Kondo, T., Sawada, S., Halodova, P., Fedorikova, A., Ohkubo, T., Murakami, K., Igari, T., Rodriguez, E. T., & Suzuki, K. (2022). Radiation-induced alteration of meta-chert. *Journal of Advanced Concrete Technology*, *20*(12), 760–776. https://doi.org/10.3151/jact.20.760

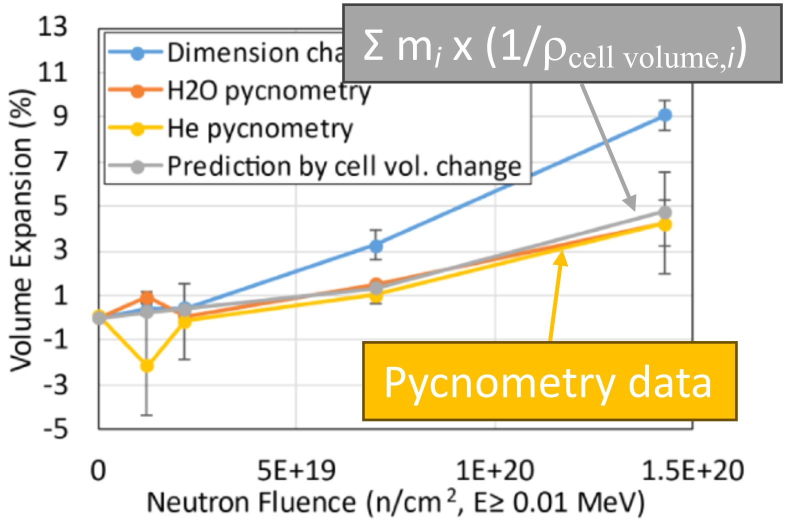
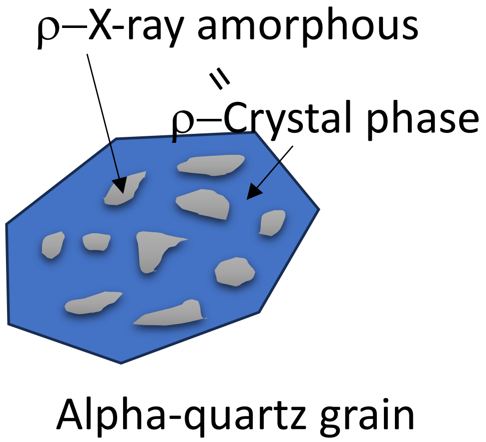


Fig. 7. Volume expansion of sandstone based on dimensional changes, water pycnometry, and He pycnometry, and the predicted expansion based on the cell volumes of the minerals as a function of neutron fluence.

Ref: Maruyama, I., Meawad, A., Kondo, T., Sawada, S., Halodova, P., Fedorikova, A., Ohkubo, T., Murakami, K., Igari, T., Rodriguez, E. T., Maekawa, K., & Suzuki, K. (2023). Radiation-induced alteration of sandstone concrete aggregate. *Journal of Nuclear Materials*, *583*, 154547. https://doi.org/10.1016/j.jnucmat.2023.154547

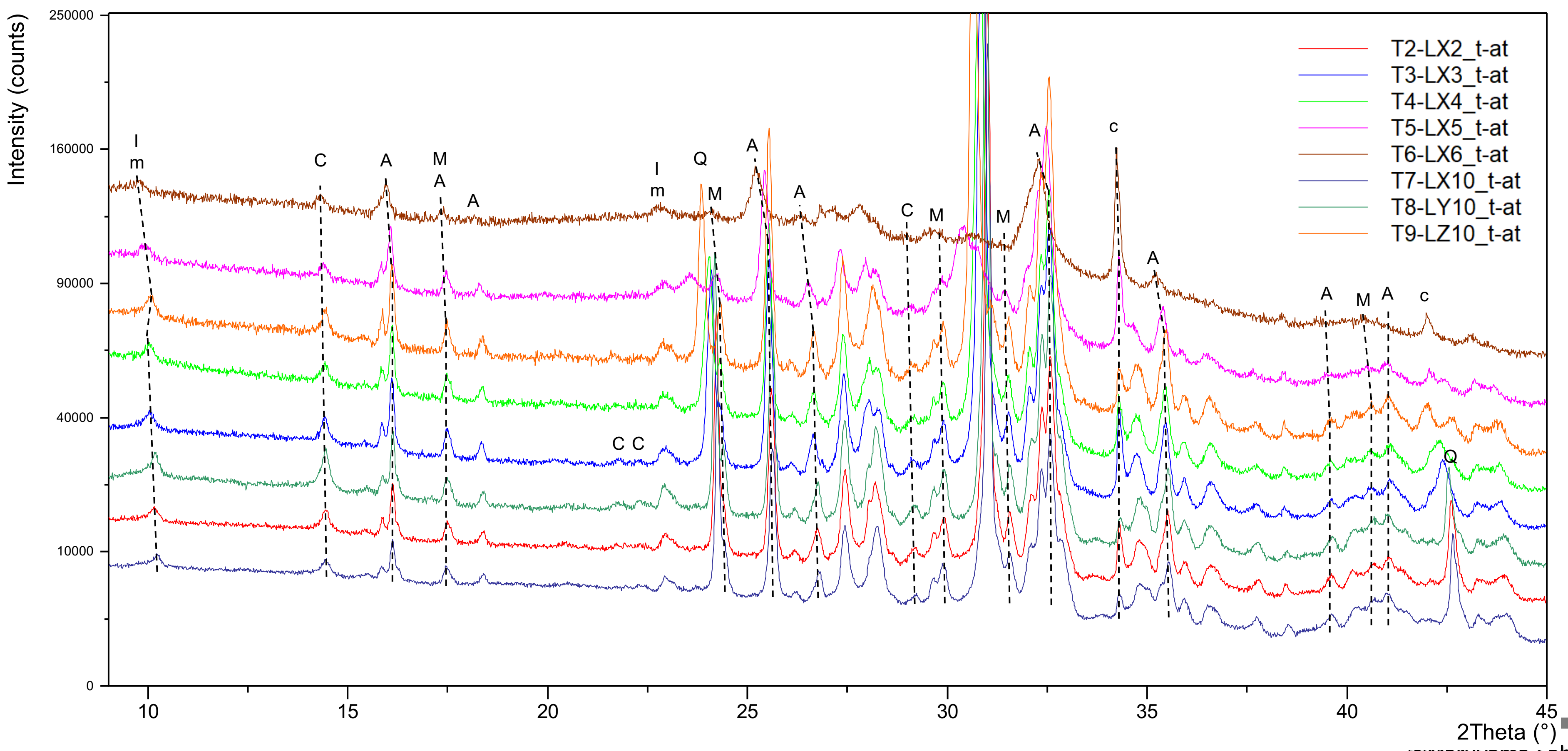


- Previous IFE results proved that the density of X-ray amorphous region = density of expanded cell volume, within our irradiated experiment.

XRD/Rietveld analysis : quartz-cell volume expansion evaluation

- XRD was measured before and after irradiation

A – albite, C – clinochlore, I – illite, M – microcline, m – muscovite, Q – quartz.

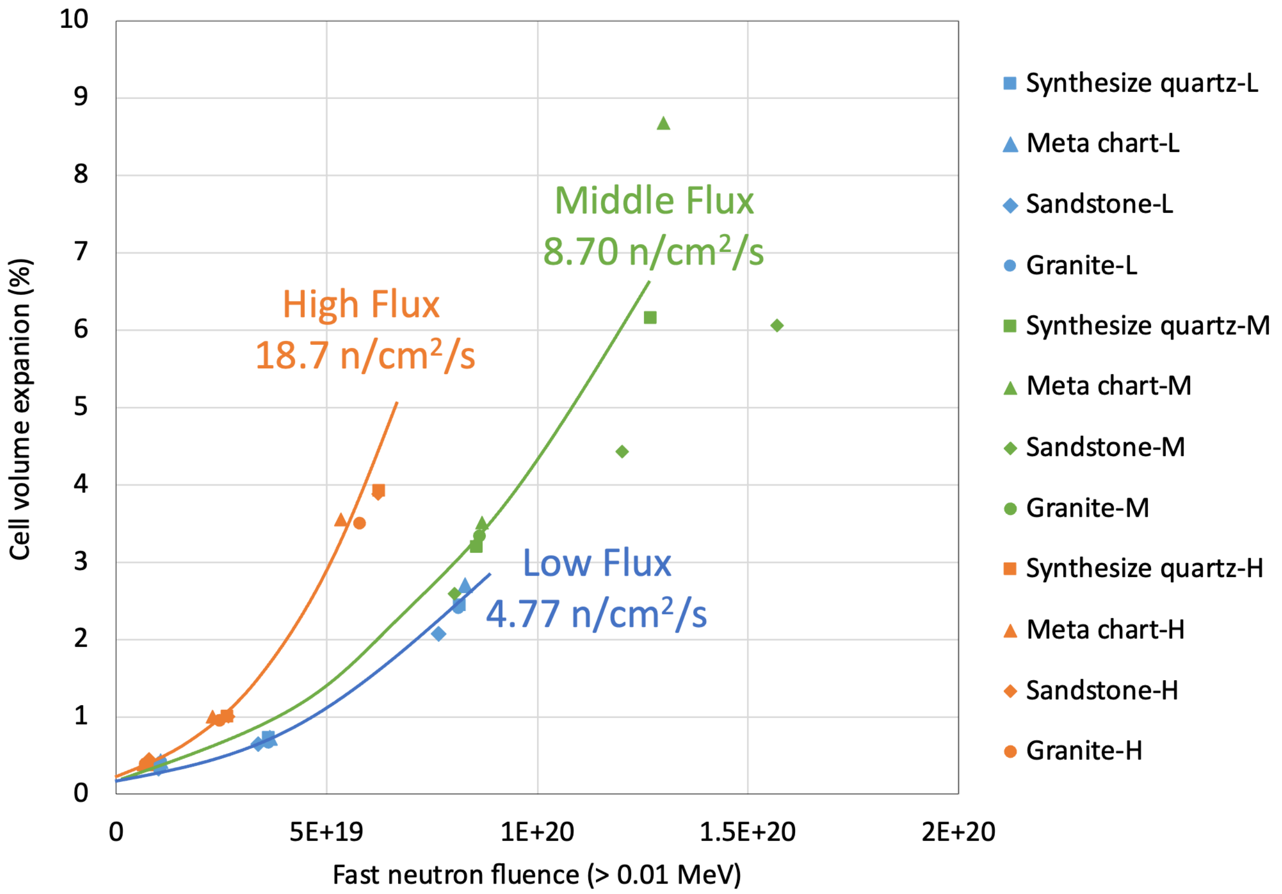


Unpublished data:
Further data validation process is needed.

Experimental results: Rietveld analysis

- α-Quartz, α-Quartz HT, β-Quartz were evaluated.
- All the cell volumes were identified.
- α-Quartz volume expansion was evaluated by the average of expansion of crystalline phases of quartz.

→ Identify the flux impact



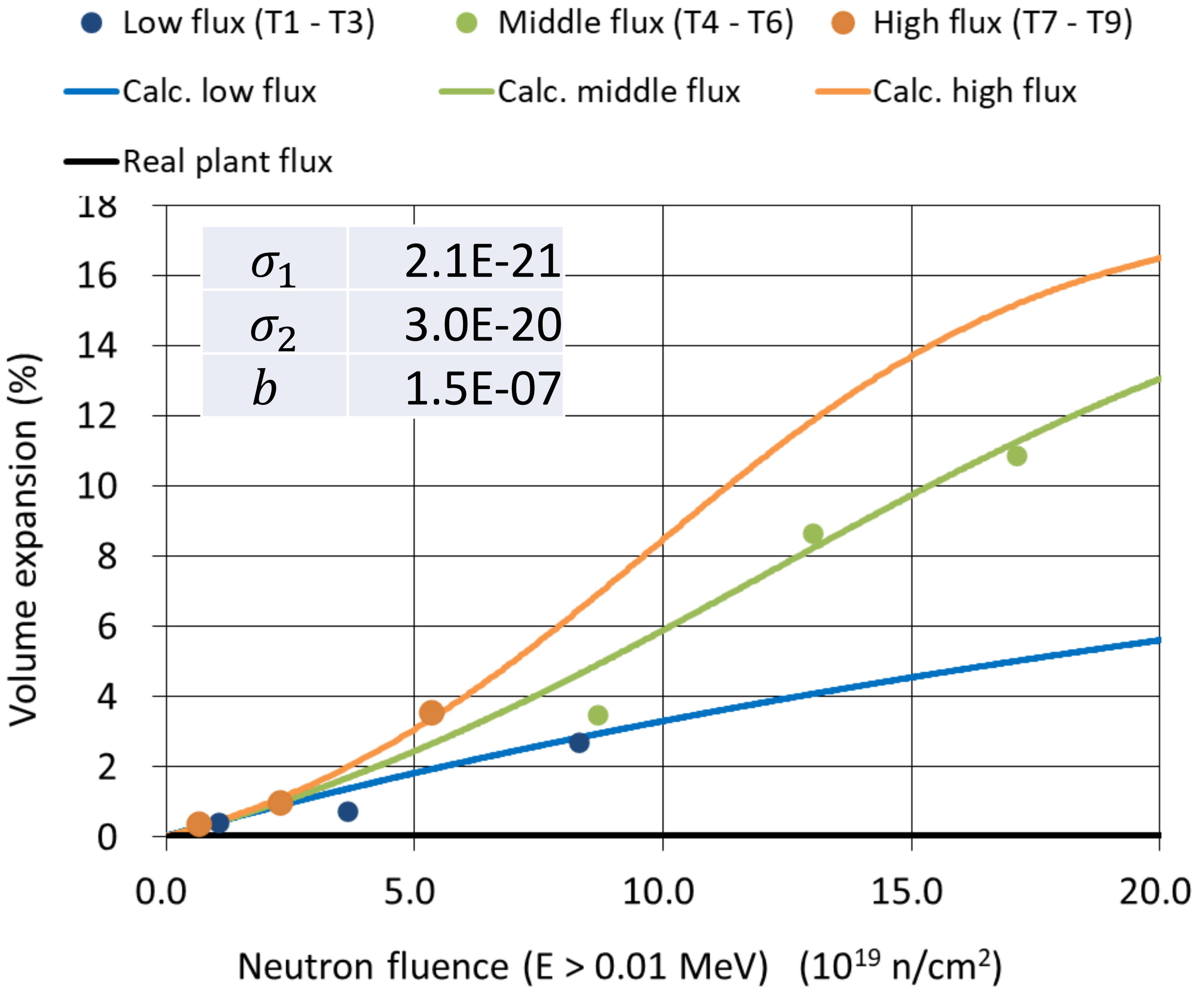
Unpublished data:
Further data validation process is needed.

Results

Here, we propose a crystalline – amorphous 2 phase model which takes into account the recovery at the interface of 2 phases:

$$R = R_1 C_1 + R_2 C_2$$
$$1 = C_1 + C_2$$

$$\begin{cases} \dfrac{dC_1}{dt} = -\emptyset \sigma_1 C_1 - \emptyset \sigma_2 C_1 C_2 + b C_1 C_2 \\ \dfrac{dC_2}{dt} = +\emptyset \sigma_1 C_1 + \emptyset \sigma_2 C_1 C_2 - b C_1 C_2 \end{cases}$$



Simple model predicts that irradiation of existing reactor's flux is less harmful.

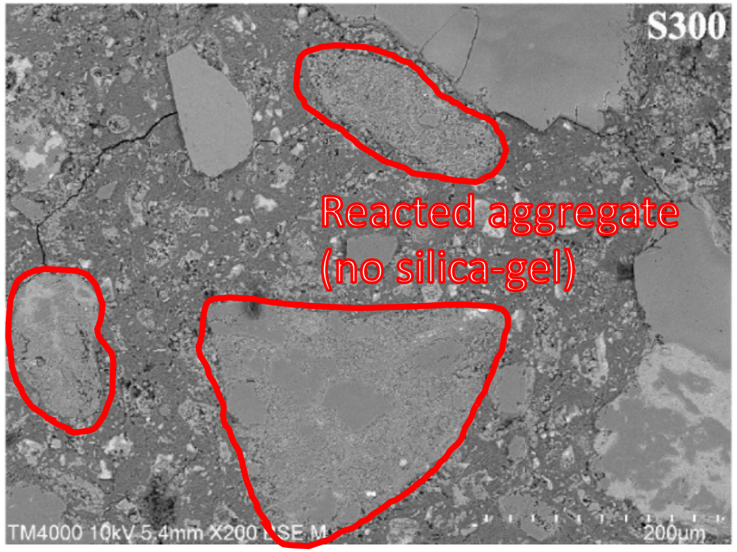
Summary

- JCAMP team identified the flux impact on the rate of expansion of alpha-quartz. This is the first evidence that the realistic neutron flux irradiation may cause less impact than those drawn by the accelerated experiments.
- Further evidences are needed. Taking cores from the real plants is meaningful.
- JCAMP team are preparing the evaluation methods for the cored samples which may have the damage distribution with steep gradient and the depth of potential damage area is very narrow.

Hamaoka Project

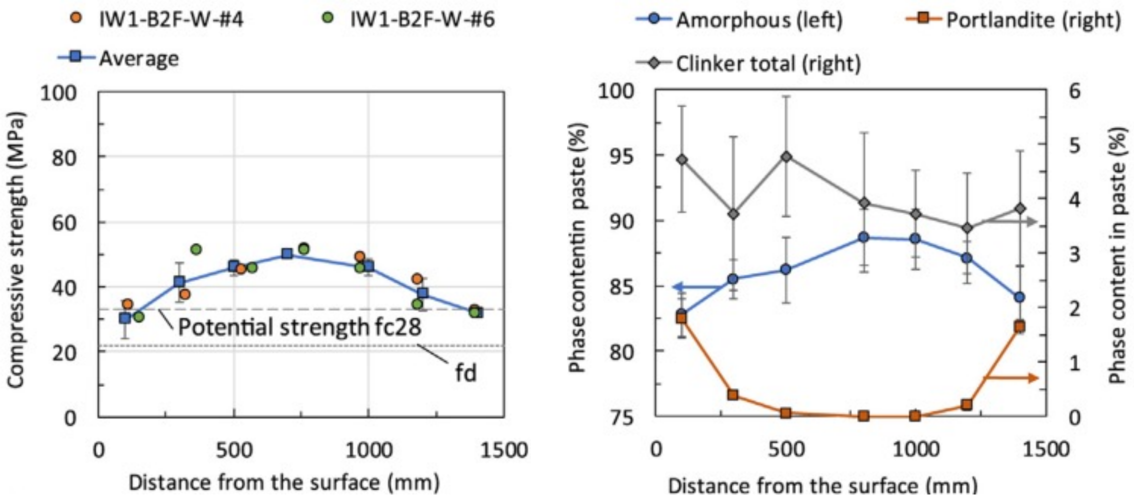
Main findings of Project phase I

- Strength increase of concrete in inner region of thick concrete wall.
- Reaction between aggregate and cement paste
    - hcp: Portlandite, Calcium silicate hydrates
    - Agg.: silica, alumina, alkali and other oxides
    - Reaction path: dissolution-precipitation
    - Confirmed by: Portlandite depletion, C-A-S-H increase in XRD
    - Characterized by aggregate reaction degree from ICP-AES



- Homogenous hydration
- No evident carbonation

Strength increase is due to Agg.-hcp reaction



Strength (left) and content of portlandite and C-S-H (right) along the wall thickness, in H1-IW1

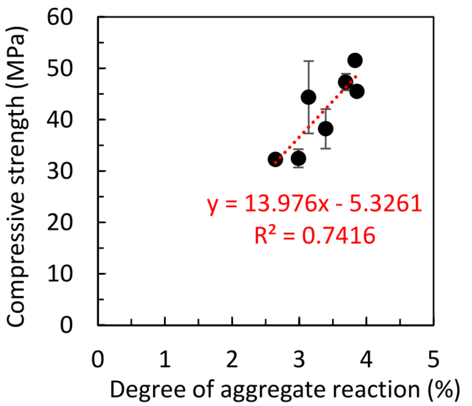


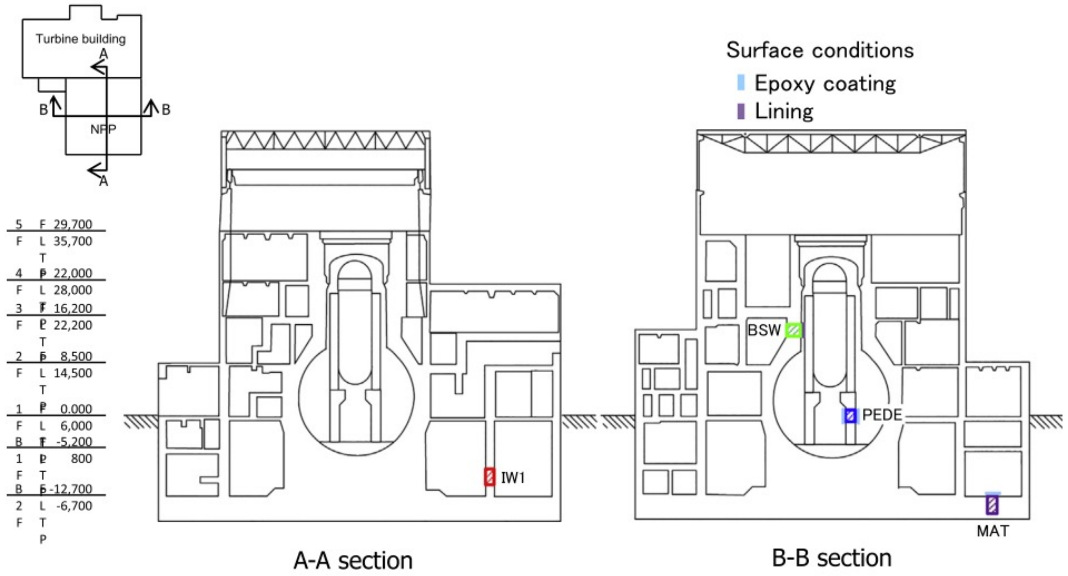
Fig. 33 Compressive strength (W) as a function of degree of aggregate reaction calculated from ICP-AES results.

Objectives:

- Reaction, its rate, involving factors
- Mechanism of strength increase

Project Phase II

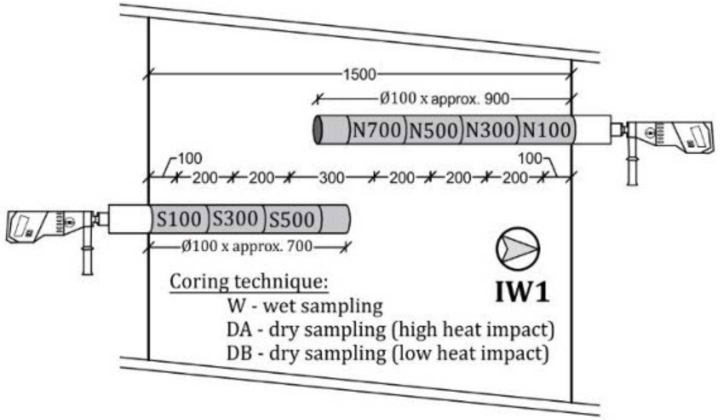
- Hamaoka power plants
  - Unit 1&2 (H1&H2) under Decom.:
  - Unit 3 (H3):
  - Unit 5 (H4):
- Members from each unit
  - Internal wall (IW1)
  - Biological shielding wall (BSW)
  - Pedestal (PEDE)
  - Mat slab (MAT)



Unit/ Member	Age of construction (years)	Cement type	Design strength (MPa)	Water to cement ratio (%)	Sand/Agg. volume ratio (%)
H1-IW1	47	OPC	22	48.3	38.5
H1-BSW	47	MPC	22	48.0	39.7
H1-PEDE	47	MPC	22	49.0	42.0
H2	47	MPC	24	48.0	43.0
H3	36	MPC	24	52.0	45.2
H5	16	MPC	32	49.0	45.5

Experimental data

- Temperature history of the members:
  - IW1: 20-30°C
  - BSW: 30-38/50-55°C during operation, 20-30°C afterwards
  - PEDE: 20-30/50-55°C during operation, 20-30°C afterwards
- Cored samples from various thick walls for:
  - Mechanical properties such as strength and elasticity, etc.
  - Physical properties such as water content, RH, porosity, etc.
  - Chemical composition by TG, XRD, ICP-AES

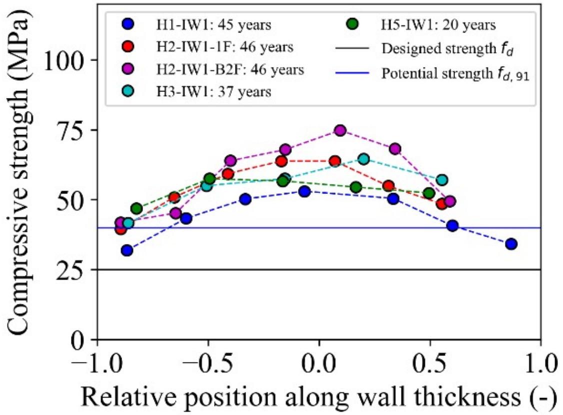
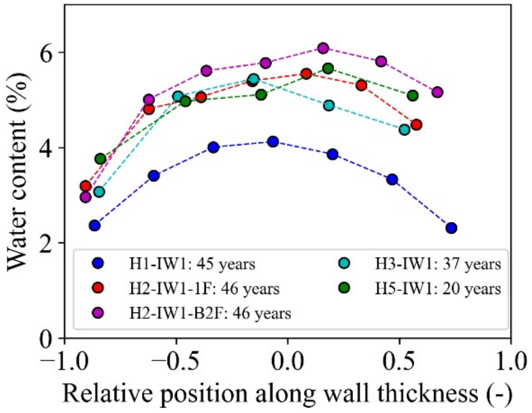
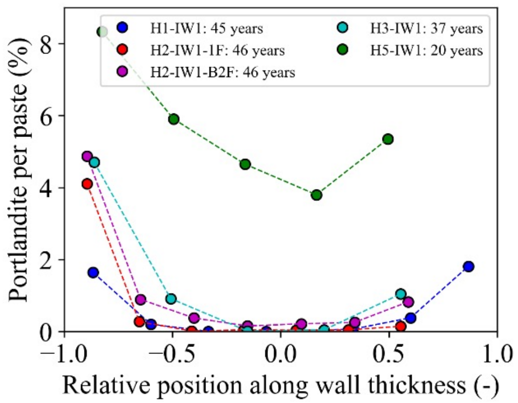
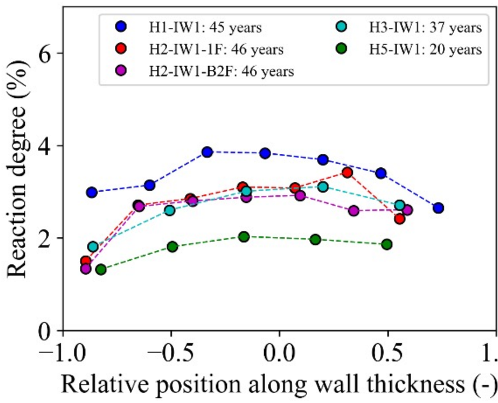
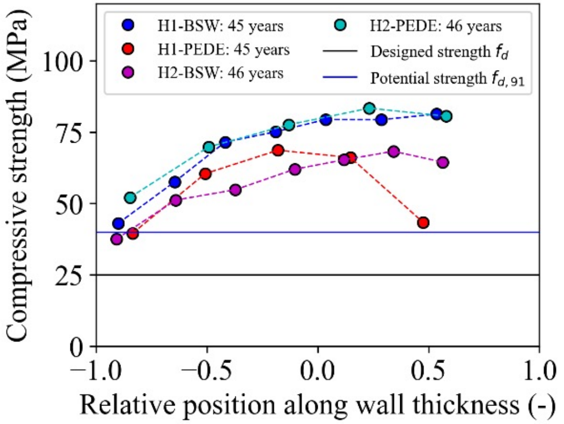
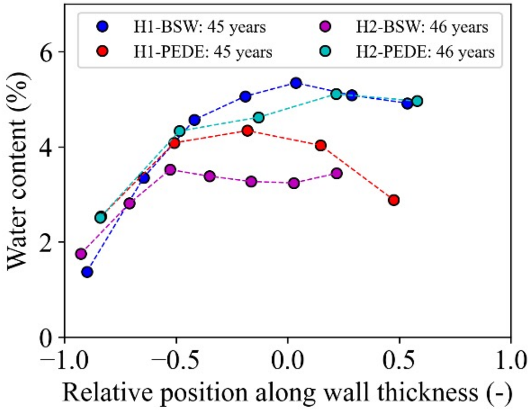
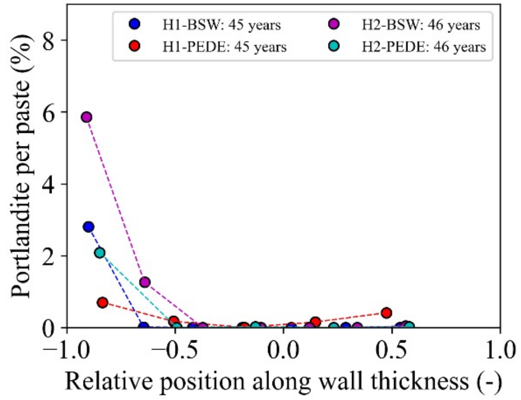
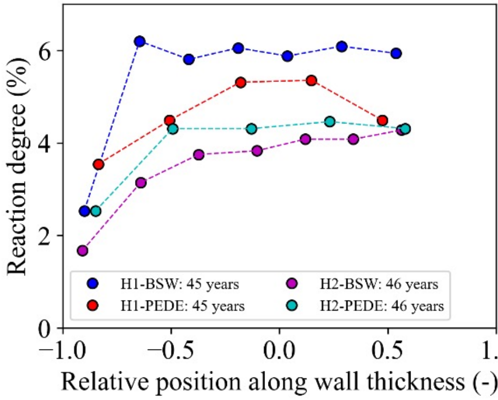


Schematic representation of coring, example of H1-IW1

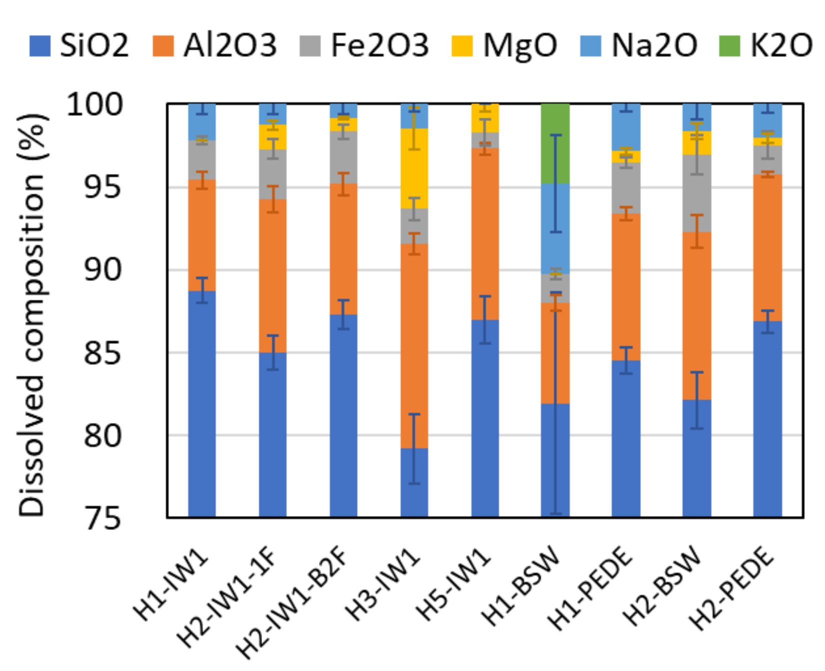
Target wall	Wall thickness (mm)	Number of samples per core	Surface condition[a]	Temperature during operation (°C)	Temperature after operation (°C)	Duration of operation (years)
H1-IW1	1500	7	N/N	20-30	20-30	16.5
H1-BSW	2200	7	N/L	30-38/50-55	20-30	16.5
H1-PEDE	1220	5	E/E	20-30/50-55	20-30	16.5
H2-IW1-1F	1700	7	N/N	20-30	20-30	18.4
H2-IW1-B2F	1700	7	N/N	20-30	20-30	18.4
H2-BSW	2200	7	N/L	30-38/50-55	20-30	18.4
H2-PEDE	1380	5	E/P	50-55	20-30	18.4
H3-IW1	1300	5	N/N	20-30	20-30	18.4
H5-IW1	1000	5	N/N	20-30	20-30	3.1

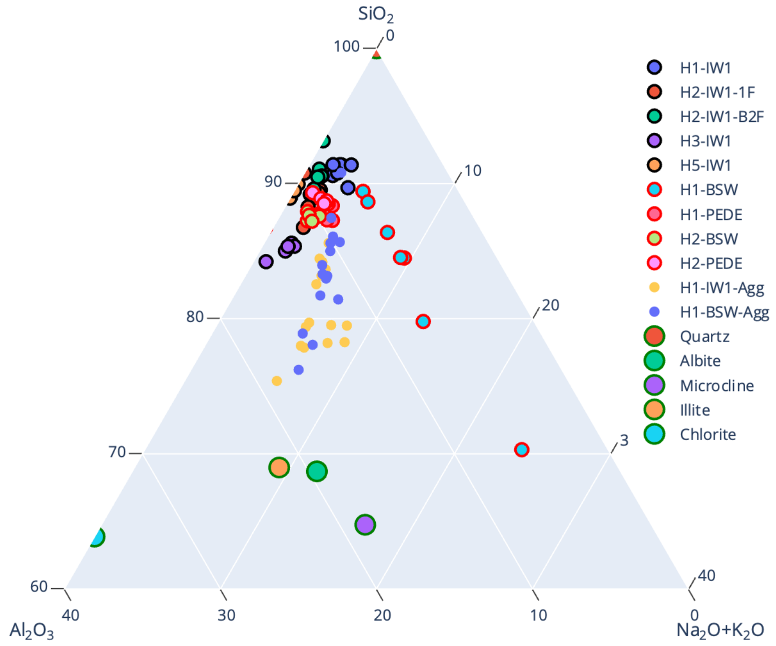
[a]N: bare surface; L: steel liner; E: epoxy resin coating; P: steel plate.

Results

	Strength	Water content	CH amount	Agg. reaction
Internal walls				
Irradiated walls (PEDE / BSW)				

What were reacted?

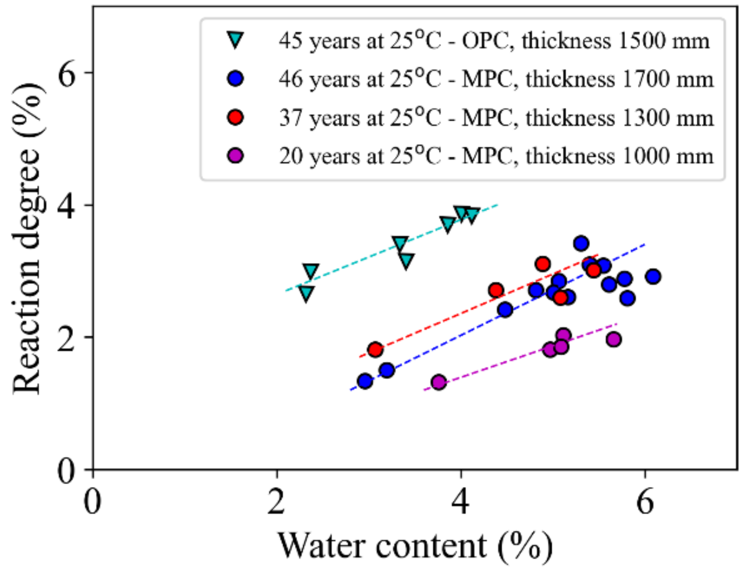




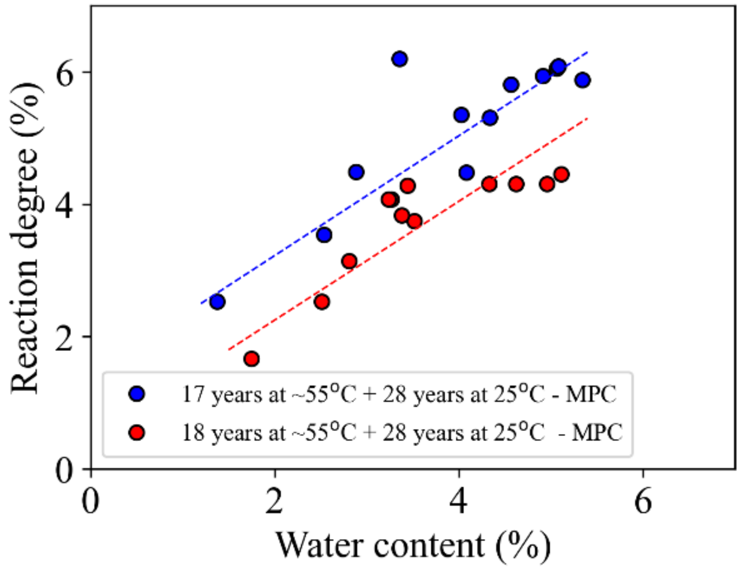
- quartz, chlorite, albite, and illite can be reacted in the system.

Rate of reaction

Internal walls



Irradiated walls
(PEDE / BSW)

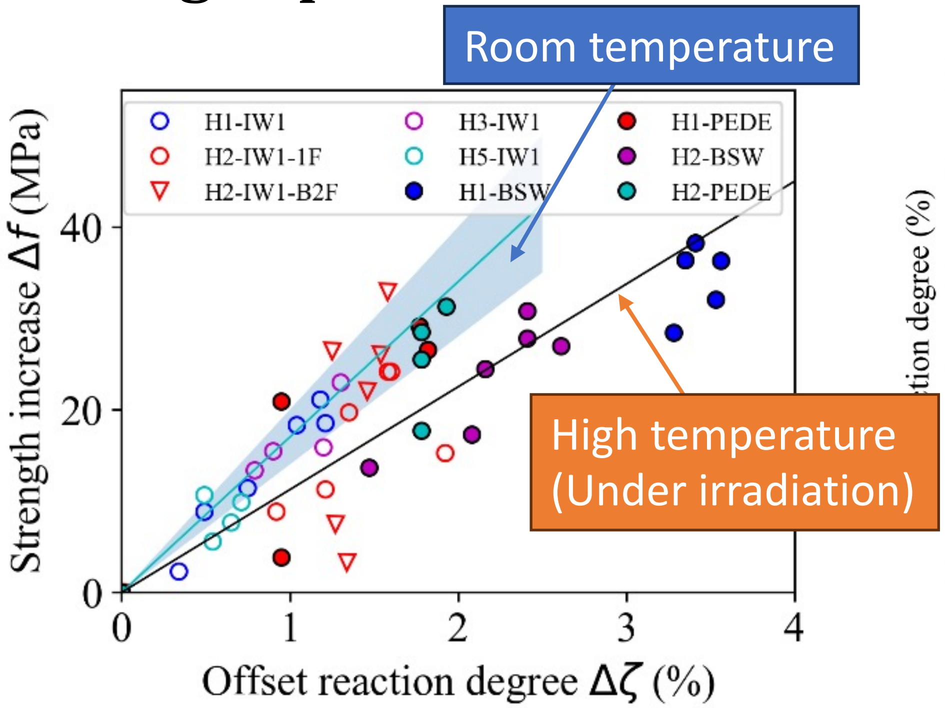


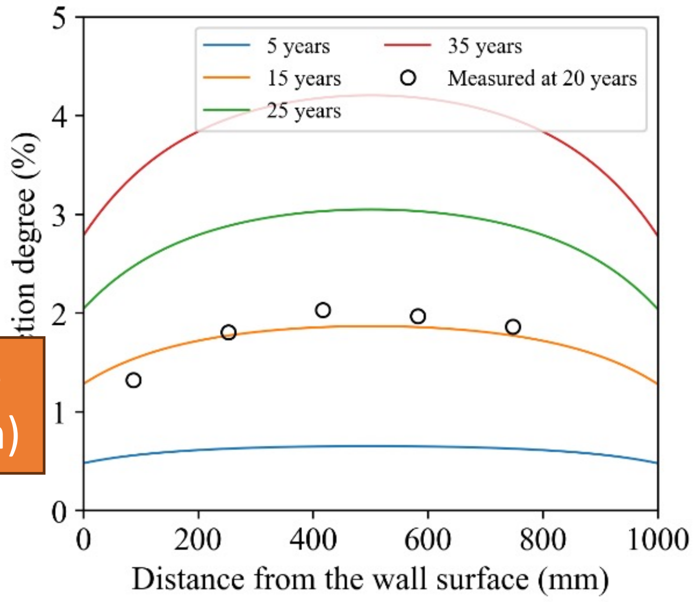
- Rate of reaction (mass%/year) and Activation of energy were obtained.

Without high-temperature history			With high-temperature history		
Walls	β (-)	$k(20°C)$ [a] (%/year)	Walls	β (-)	$k(20°C)$ (%/year)
α_0=0.1, θ=0.1, E_a=28 kJ/mol					
H1-IW1	10	0.086	H1-BSW	20	0.094
H2-IW1	100	0.056	H1-PEDE	20	0.108
H3-IW1	100	0.079	H2-BSW	20	0.064
H5-IW1	100	0.108	H2-PEDE	50	0.075

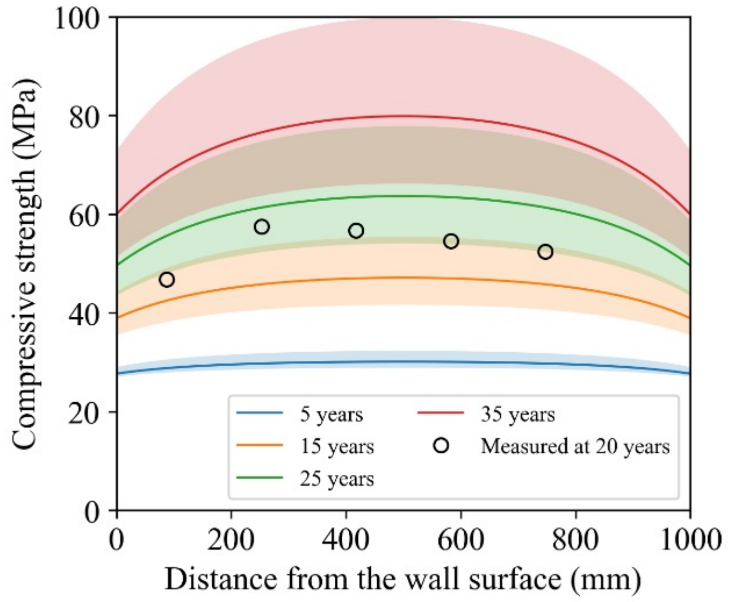
[a]For ease of understanding physical meaning, reaction rate $k(20°C)$ is given instead of A. The relation between A and $k(20°C)$ is shown in Eq.(4).

Strength prediction





Reaction degree of Agg.



Strength development

- FDM → water content + temperature distribution → Rate of reaction degree.
  → Microstructure change, Diffusion coeff. + water consumption → FDM
- Strength development of thick concrete wall can be predicted.

Summary and comments

- General sandstone fine aggregate may be reactive for long-period.
- But aggregate did not show the ASR. The dissolution rate vs Ca movement is the key. (Another paper is in preparation.)
- Slow reaction of aggregate enhance the strength, which contributes to the high performance of shear wall.
- Temperature (Gamma-ray induced) has accelerated this phenomenon.
- Neutron also may influence on increasing in dissolution rate of minerals by metamictication (neutron-irradiated amorphization)
- This influence should have also an important role in the integrity evaluation of RC member exposed to irradiation.

Thank you for your attention.

Free to ask:

Ippei Maruyama
i.maruyama@bme.arch.t.u-tokyo.ac.jp

Global Expertise • One Voice

PWROG-18068, "Use of Direct Fracture Toughness for Evaluation of RPV Integrity"

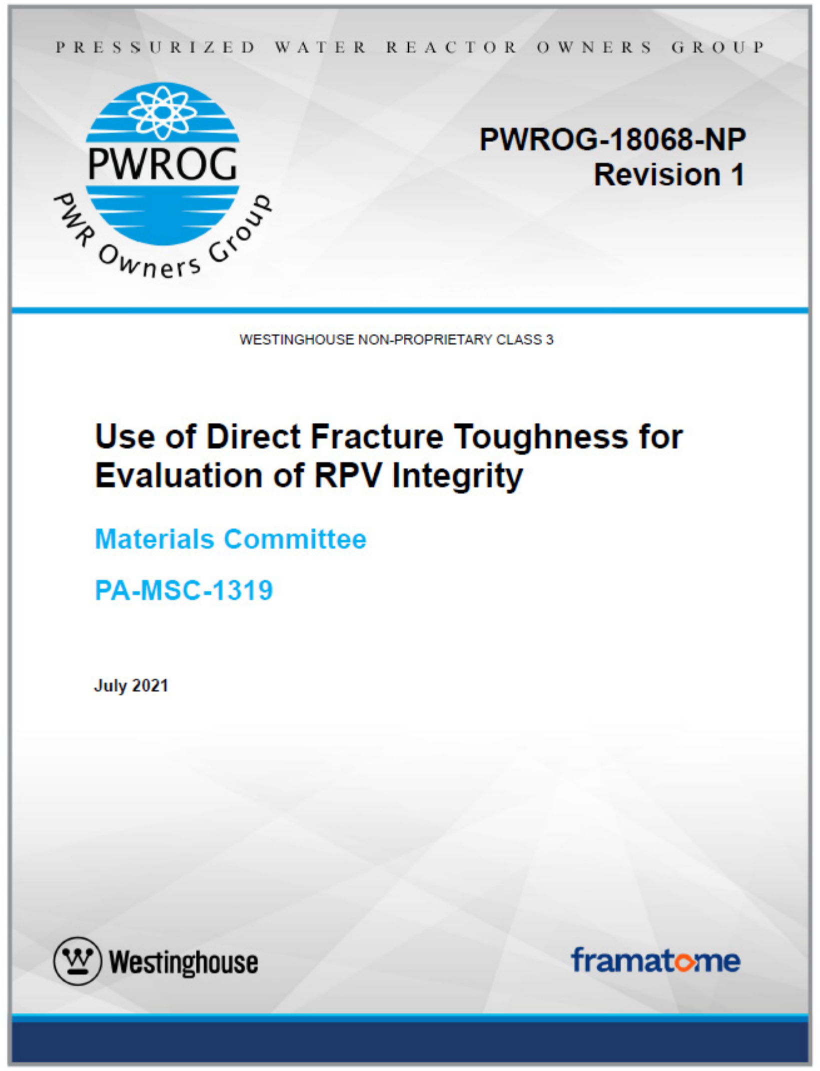
Brian Hall - Westinghouse

LWRS Spring meeting April 30 – May 1, 2024

PWROG
PWR Owners Group

PWROG-18068-NP, "Use of Direct Fracture Toughness for Evaluation of RPV Integrity"

- The methodology justifies the use of direct fracture toughness data to evaluate RPV integrity as an alternative to the requirements/methods of pressurized thermal shock (PTS) (10 CFR 50.61) and pressure-temperature (P-T) limit curves (10 CFR 50, Appendix G). The topical report discusses a methodology to:
  - Generate irradiated or unirradiated ductile-brittle transition reference temperature (T_0) according to the industry consensus ASTM E1921-20 Standard Test Method
  - Adjust the data for differences between the tested material using industry consensus ASTM E900-15 Standard Guide for predicting embrittlement
  - Account for test result uncertainty and material variability
  - Apply the data using NRC-endorsed methods


PRESSURIZED WATER REACTOR OWNERS GROUP
PWROG
PWR Owners Group
PWROG-18068-NP
Revision 1
WESTINGHOUSE NON-PROPRIETARY CLASS 3
Use of Direct Fracture Toughness for Evaluation of RPV Integrity
Materials Committee
PA-MSC-1319
July 2021
Westinghouse
framatome

PWROG
PWR Owners Group

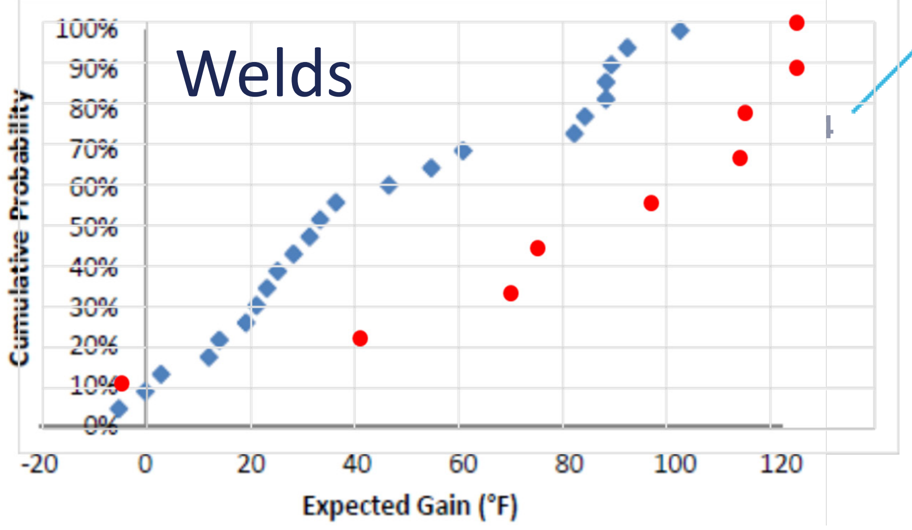
Direct Fracture Toughness Activities

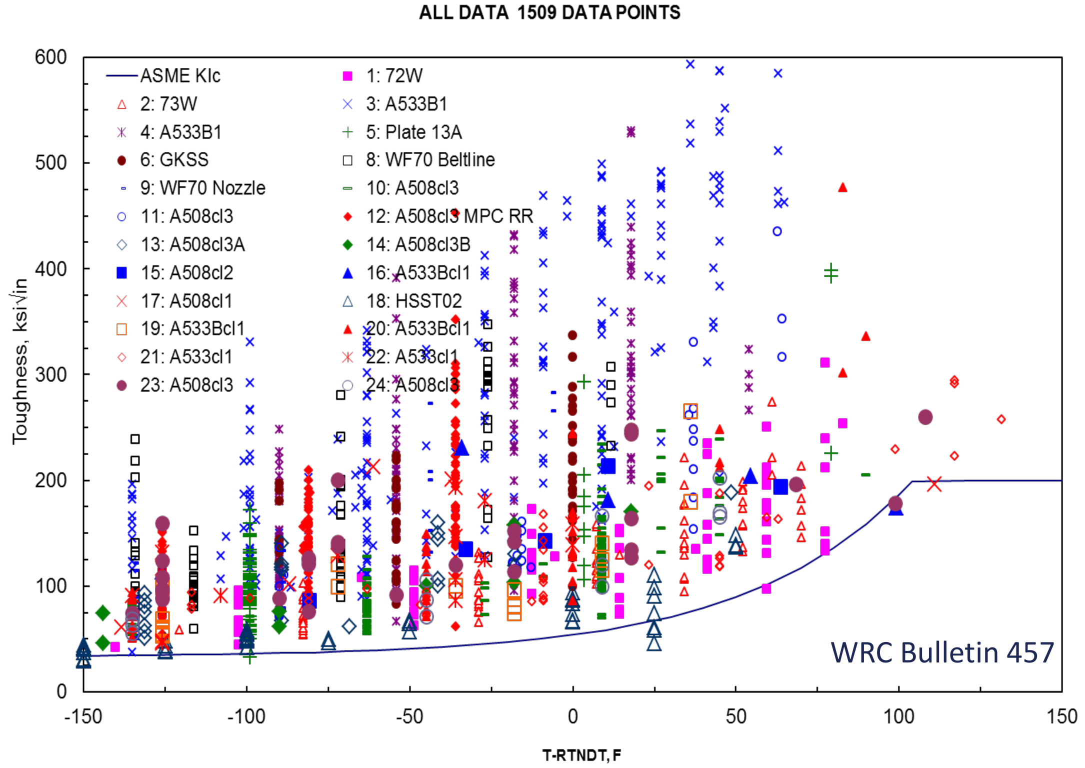
- PWROG-18068-NP submitted to NRC for review in July 2021
- Provides a methodology to use fracture toughness data as an alternative to specific sections of NRC-approved topical reports for generating pressure-temperature curves
  - WCAP- 14040-A
  - BAW-10046A
- Applicable to all PWRs
- NRC accepted PWROG-18068 for review
- 25 multi-part requests for additional information received March 2022
  - A number of meetings and changes made to address NRC questions
  - Final RAI responses and PWROG-18068 markup submitted March 8, 2024
- Parallel complimentary, different method in ASME Code with ballot of Code Case N-914 – *Methods to account for embrittlement*
  - Basis in MRP-462, Rev. 1 Draft (Feb. '23)
  - Addressing reviewer comments

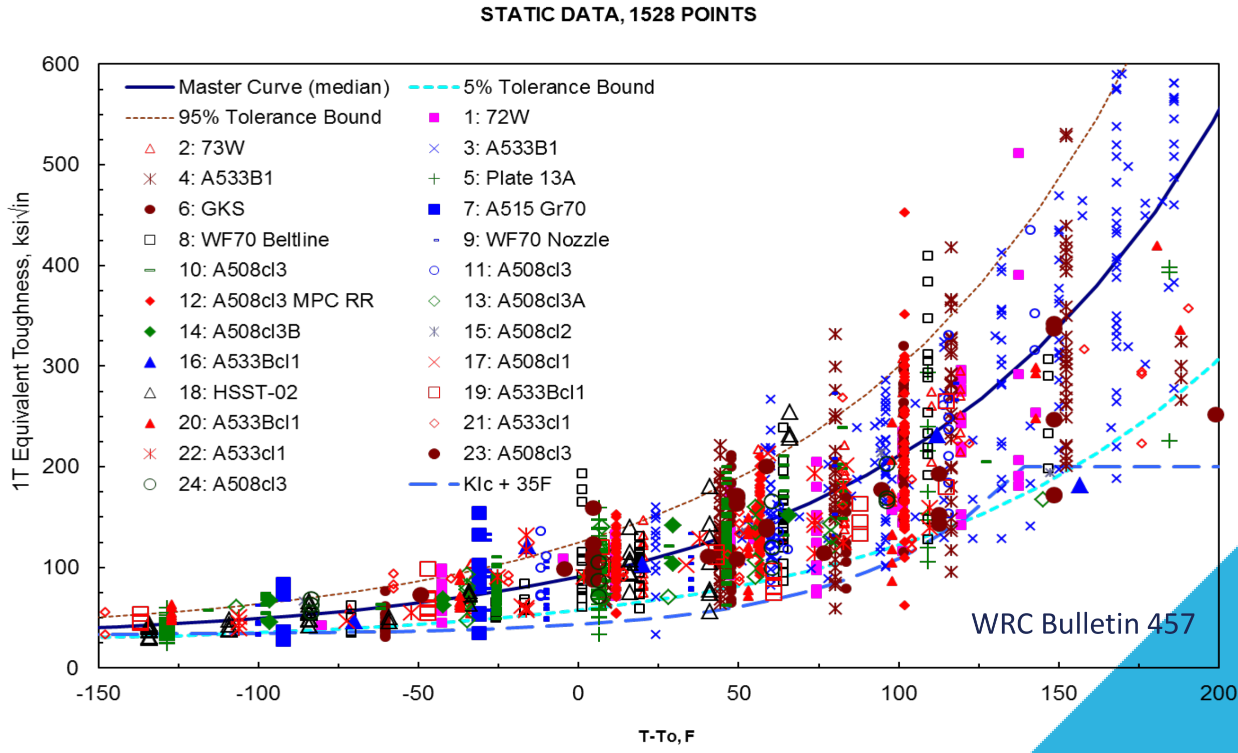
PWROG
PWR Owners Group

Why Direct Fracture Toughness

- **Master Curve**
  - Reduced uncertainty
  - Reduced inconsistency
  - Characterizes margin statistically
  - Based on actual fracture toughness measurement
- **Testing Irradiated Material**
  - Reduced embrittlement prediction uncertainty
  - Reduced embrittlement prediction error (bias)
    - e.g., RG1.99R2 high fluence non-conservatism
  - Uncertainties are accounted for explicitly







Methodology for Application of Master Curve Test Data

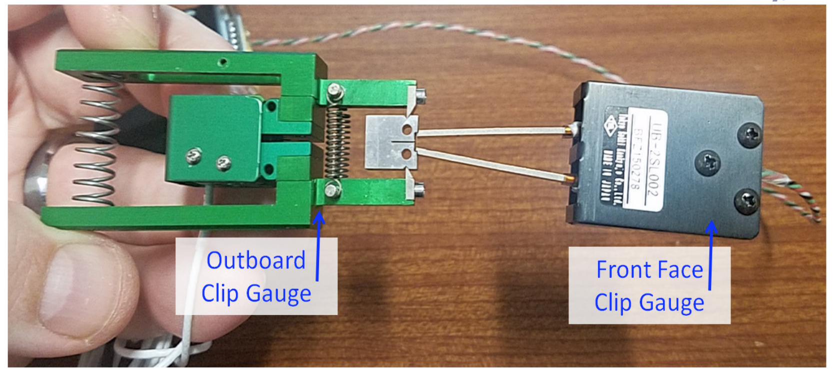
- For PTS evaluations, the following is used:

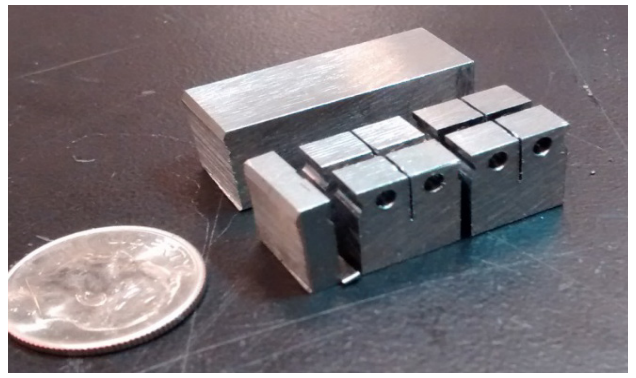
  $RT_{PTS} = RT_{T0} + \textit{adjustment} + \textit{margin}$

- Using ASME Section XI, Appendix G 2013
  - $K_{Ic} = 33.2 + 20.734 \exp[0.02\ (T - \{T_0 + 35 + \textit{adjustment} + \textit{margin}\})]$ **(K_{Ic} curve with RTT_0)**
- OR
- Using Code Case N-830-0 as modified by the NRC condition
  - $\mathbf{K_{Jc\text{-}lower95\%}} = 22.9 + 33.3 \exp[0.0106\ (T - \{T_0 + \textit{adjustment} + \textit{margin}\})]$

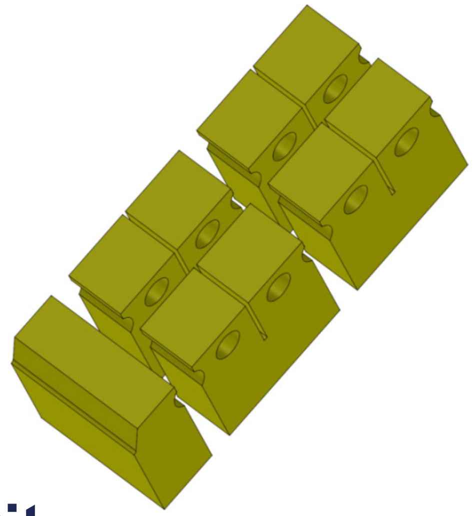
- This topical report provides a methodology to determine the *adjustment* and *margin* terms

Generation and Validation of T_0 Data





- **Irradiated T_0 can be obtained by**
  - Using existing data
  - Testing specimens machined from unirradiated archive material
  - Testing specimens machined from material irradiated in a PWR surveillance capsule, or
  - Irradiating specimens in at high flux & testing; e.g. material test reactor (MTR)
    - MTR irradiation must include validation material in each Cu group that have test materials
      - Low Cu: Cu weight percent (wt. %) ≤ 0.053
      - Medium Cu: Cu wt. % between 0.053 and 0.28
      - High Cu: Cu wt. % > 0.28
    - Ensures that MTR irradiated specimens are representative of PWR irradiated specimens
      - Potential Flux effect
      - Other differences: spectrum, temperature, unknown
      - Ensures a well-designed MTR irradiation of specimens



Specimen Testing

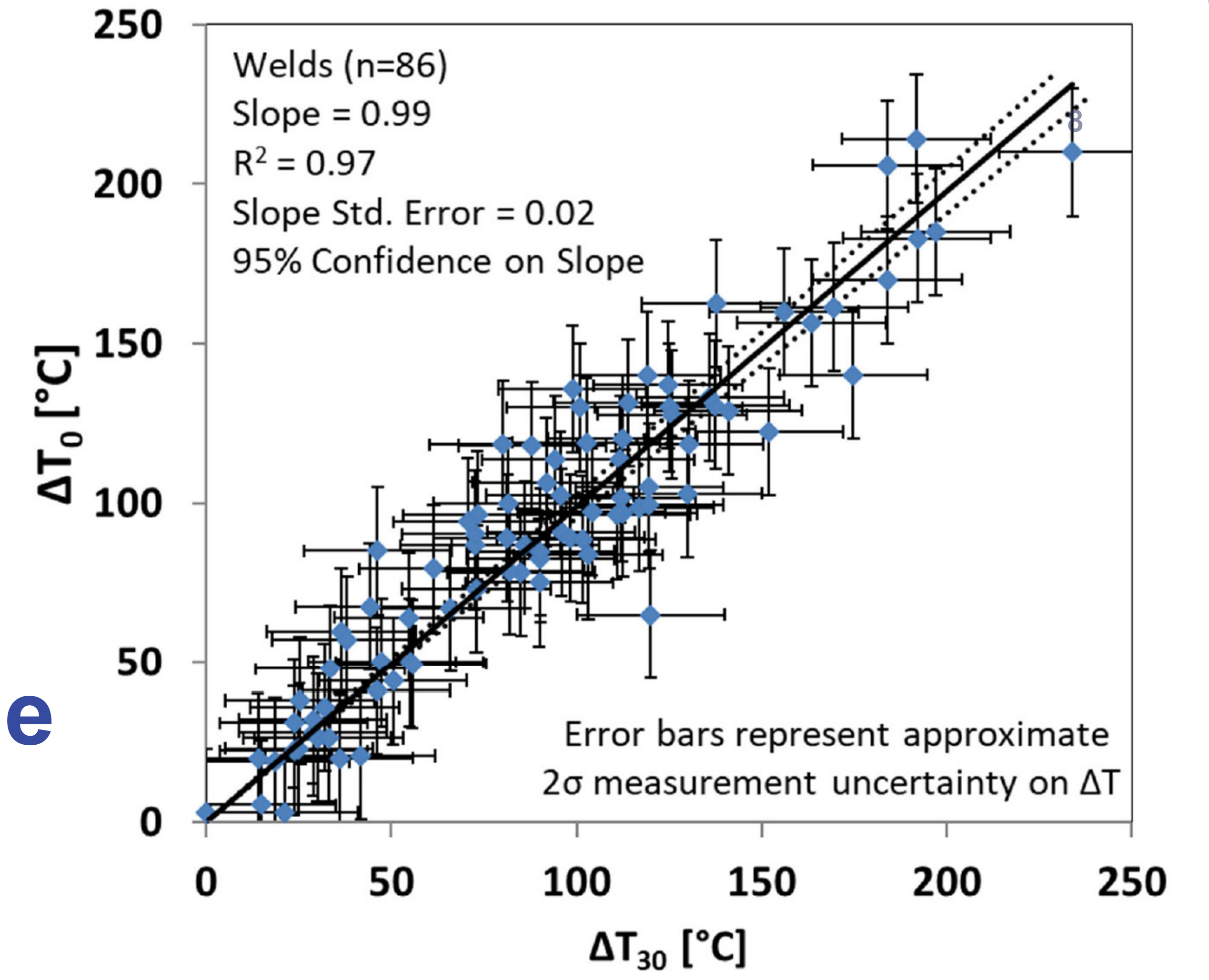
- **Irradiation of the same heat of material is required to evaluate the RPV material of interest, except**
  - Generic unirradiated T_0 method is described
    - Minimum 4 valid T_0 from same type, manufacturer, or class
    - 95/95 one-sided tolerance limit factor (k1) is used rather than 2 which is typically used for large populations
- **Testing in accordance with ASTM E1921-20**
  - Data sets are screened for inhomogeneity in accordance with 10.6 of ASTM E1921-20
  - Data sets that fail the screening criterion are evaluated in accordance with Appendix X5 “Treatment of Potentially Inhomogeneous Data Sets,” of ASTM E1921-20 with T_{0IN} (as calculated in Appendix X5) substituted for T_0.
  - Any geometry that meets ASTM E1921-20
    - A 10°C bias is added for the SEB Charpy size (10x10mm) specimen (ASTM E1921)

Data Adjustment

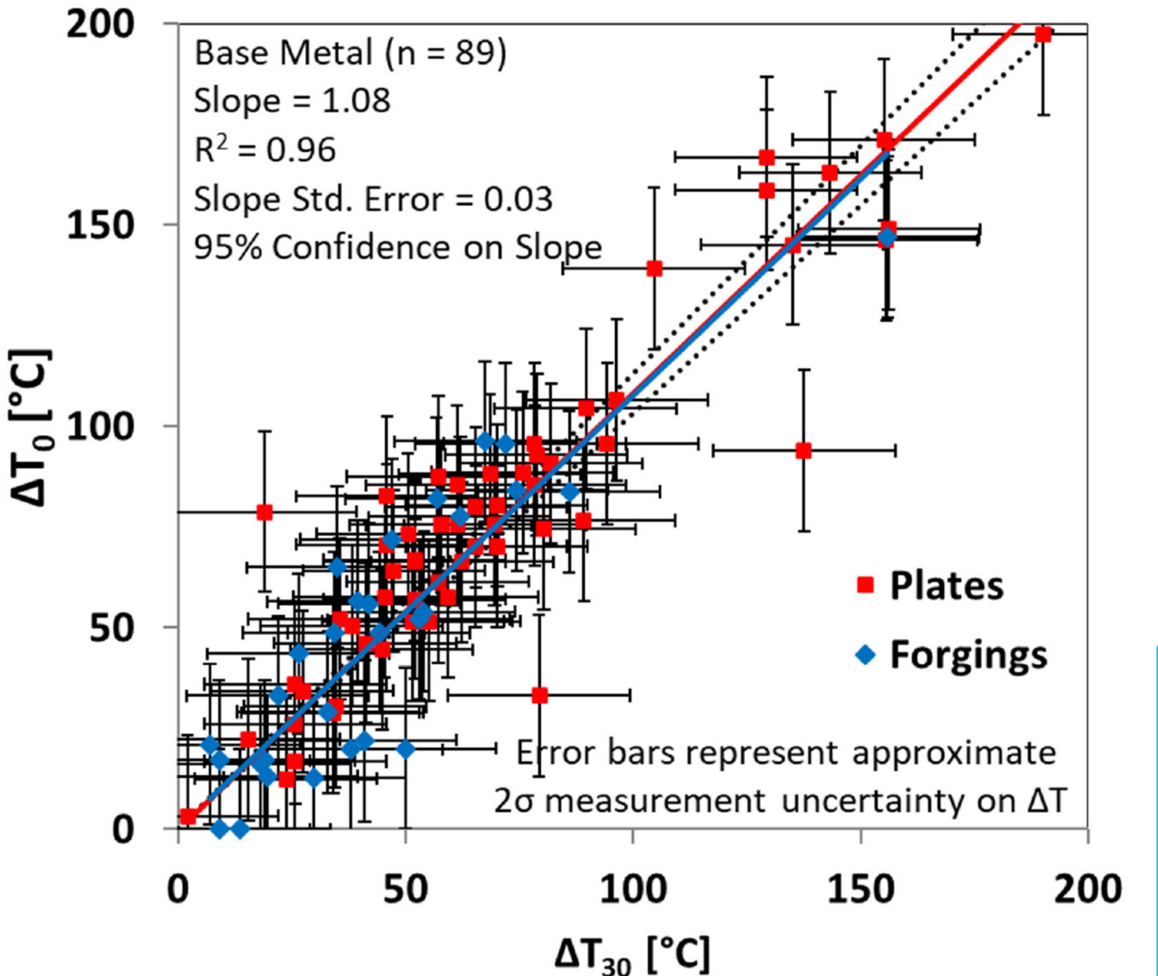
- **Tested specimens will rarely reflect the exact same irradiation conditions and chemistry as the represented RPV material**
  - Adjustments presented herein are made using the embrittlement trend curve (ETC) in ASTM E900-15 (other ETCs could also be used)

$$adjustment = \left(\Delta T_{30\,RPV} - \Delta T_{30\,Specimens}\right) \bullet (If\ BM, 1.1)$$

    - Best-estimate inputs are used for the irradiated data adjustments (Cu, Ni, Mn, P, Temp., Fluence)
    - An NRC-approved method of fluence evaluation consistent with the plant licensing basis, or another NRC-approved method of fluence evaluation
    - Weld = 1.0 and Base metal = 1.1



Consistent with draft ASME Code Case N-913 & cites EPRI MRP-462



Margin Term

$$Margin = \sqrt{\sigma^2_{E1921} + \sigma^2_{adjustment} + \sigma^2_{tempspecimen} + \sigma^2_{tempRPV} + \sigma^2_{fluencespecimen} + \sigma^2_{fluenceRPV}}$$

- **Accounts for uncertainties**
  - **Simplified, bimodal or multimodal can be used if inhomogeneous**
  - **Adjustment using ETC:** $\sigma_{adjustment} = \boldsymbol{max}\left[9°\mathrm{C}, \{\boldsymbol{C} \bullet ([If\ BM, 1.1] \bullet \Delta T_{30RPV})^{\boldsymbol{D}}\} \bullet \frac{|adjustment\ |}{(If\ BM, 1.1) \bullet \Delta\mathrm{T}_{30\ RPV}}\right]$
  - **Irradiation temperature** (effect of uncertainty on embrittlement using the ETC)
    - Test specimens; 0 if irradiated in assessed RPV
    - RPV; (2°F can conservatively be used)
  - **Fluence** (effect of uncertainty on embrittlement using the ETC)
    - Test specimens (0 if unirradiated)
    - RPV projection

PWROG
PWR Owners Group

Basis: J. B. Hall, E. Lucon, and W. Server, "Practical Application of the New Homogeneity Screening Procedure Added to ASTM E1921-20 and Appendix X5 Inhomogeneous Data Treatment," Journal of Testing and Evaluation 50, no. 4 (July/August 2022): 2190–2208. https://doi.org/10.1520/JTE20210716

Determination of σ_{E1921}

- **σ_{E1921} is calculated in accordance with paragraph 10.10 of ASTM E1921**
  - (with standard calibration practices, σ_{exp} = 4°C)
- **Uncertainty due to material variability**
  - In 2019, a homogeneity screening procedure was added to ASTM E1921, Appendix X5
    - Identifies datasets which do not follow expected normal material Weibull distribution and the 95% lower bound curve would not bound 95% of data
    - Inhomogeneity can result from initial toughness variation (i.e. segregation) or uneven embrittlement due to chemical composition variation

PWROG
PWR Owners Group

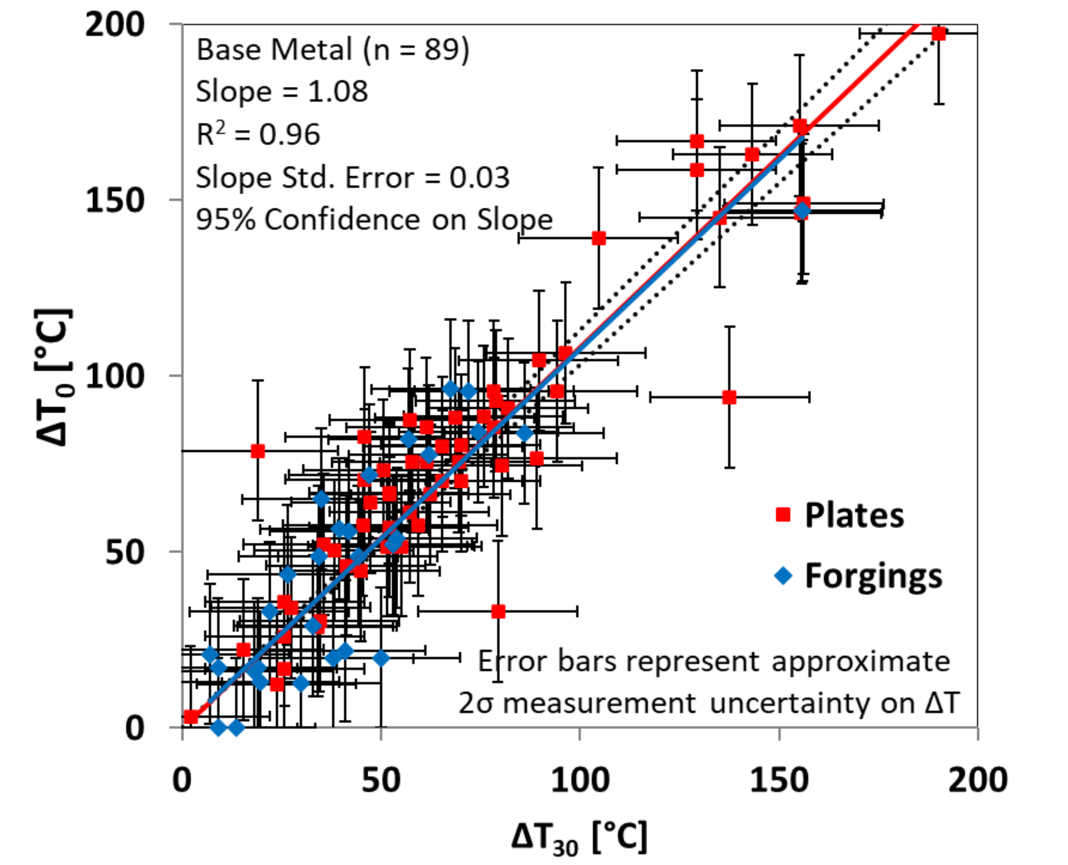
Determination of $\sigma_{adjustment}$

- $\sigma_{adjustment}$ **is proportional to ASTM E900-15 σ with a minimum value of 9°C**

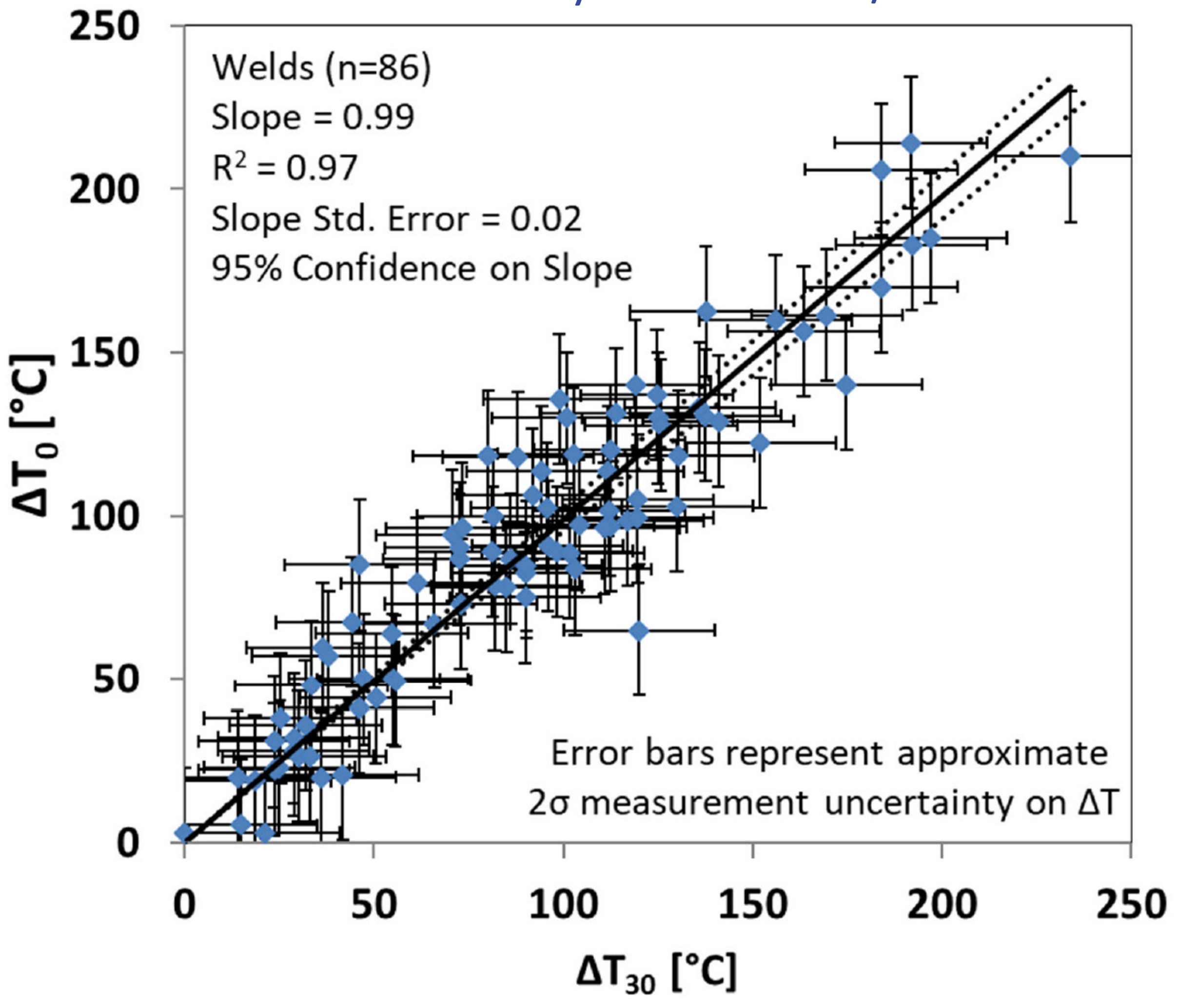
$$\sigma_{adjustment} = \boldsymbol{max}\left[9°\mathrm{C}, \{\boldsymbol{C} \bullet ([If\ BM, 1.1] \bullet \Delta T_{30RPV})^{\boldsymbol{D}}\} \bullet \frac{|adjustment\,|}{(If\ BM, 1.1) \bullet \Delta T_{30\,RPV}}\right]$$

  - Adjustment from unirradiated results in use of full σ_{E900}
  - With small adjustments, the 9°C is the value used

- **9°C uncertainty due to material variability**
  - Typical σ_{E1921} ranges from 6 to 8°C
  - Typical σ_{41J} ranges from 4 to 10°C
  - $\sqrt{T_{0init}^2 + T_{0irr}^2 + T_{30init}^2 + T_{30irr}^2}$ **=** $\sqrt{\mathbf{6^2 + 8^2 + 4^2 + 10^2}}$ **= 14.4°C**
  - Standard Deviation on Fit Residuals = 17°C for BM and Welds
  - $\sqrt{\mathbf{17^2 - 14.4^2}}$ **= 9°C (material variability)**



Data is mostly from NUREG/CR-6609



Basis: J. B. Hall, B. Golchert, and D. Simpson, "An Examination of Margins Needed to Ensure Conservative Application of T0 to RPV Fracture Toughness,"

ASME PVP2024-125225

Margin Evaluation

- **Method was used with measured fracture toughness data to evaluate if margin is sufficient**
  - Unirradiated T_0 was adjusted to irradiated T_0 with margin added from same heat (irradiated T_0 as if from RPV assessed)
  - Adjustment from unirradiated results in use of full σ_{E900}
- **98% of the data is bounded for base metals**
- **100% is bounded for welds**
- **Data is mostly from NUREG/CR-6609**

Does the method bound measured T_0 at 2nd condition?

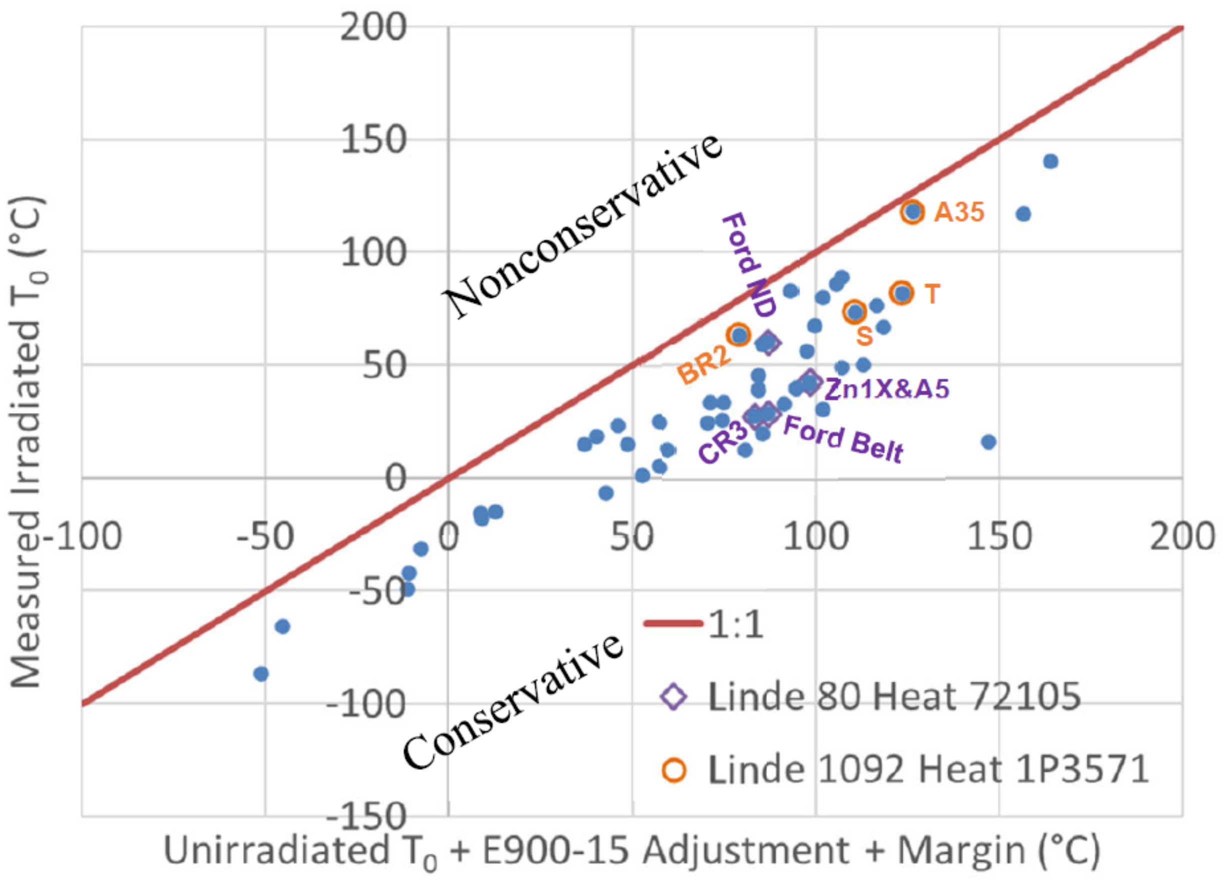


Figure 3 Bounding Adjusted T₀ Compared to Measured Irradiated T₀ for Weld Metals (labels are capsule names which are referenced later)

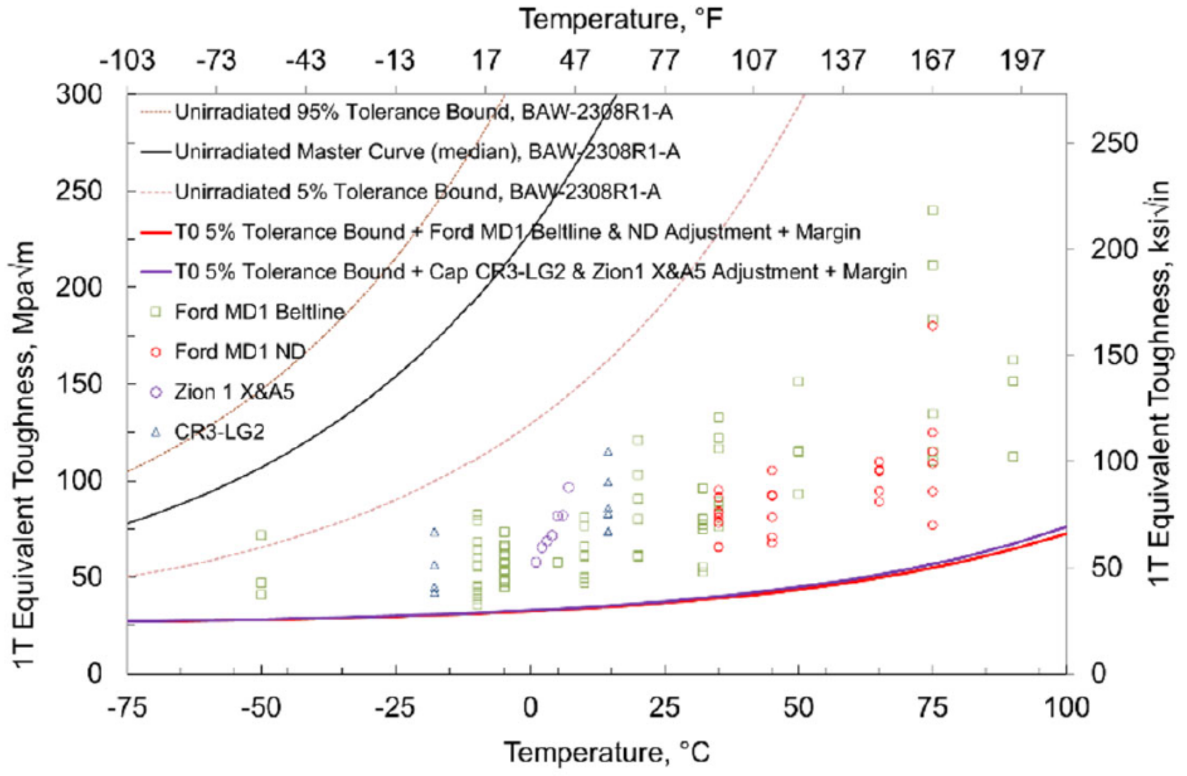


Figure 9 Comparison of Fracture Toughness Values to Bounding Curves for Weld Heat 72105 Adjusted from Unirradiated T₀

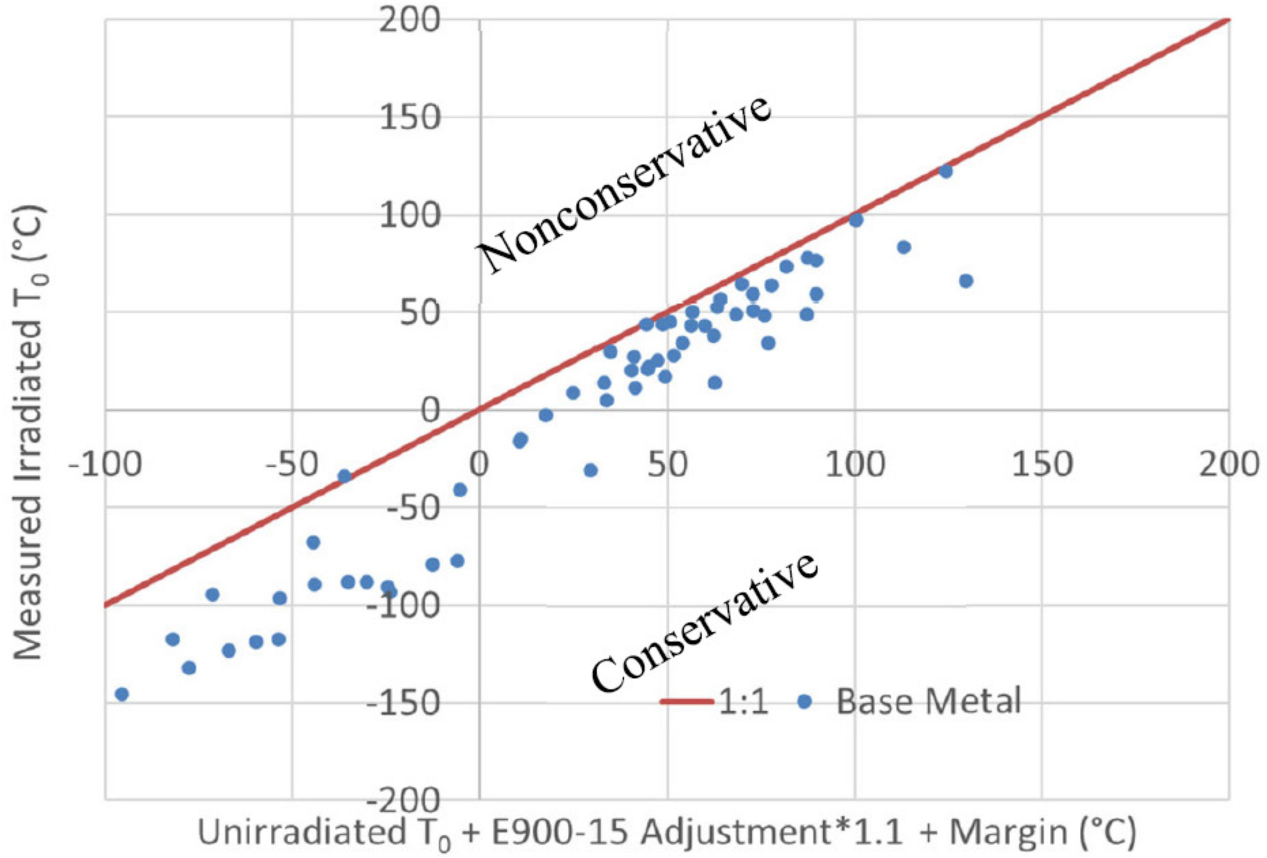


Figure 4 Bounding Adjusted T₀ Compared to Measured Irradiated T₀ for Base Metals

$$\sigma_{adjustment} = \boldsymbol{max}\left[9°C, \{C \bullet ([If\ BM, 1.1] \bullet \Delta T_{30RPV})^{D}\} \bullet \frac{|adjustment\,|}{(If\ BM, 1.1) \bullet \Delta T_{30\ RPV}}\right]$$

Margin Evaluation

- **Method was used with measured fracture toughness data to evaluate if margin is sufficient**
  - Irradiated T_0 was adjusted to another irradiated T_0 with margin added from same heat (2nd irradiated T_0 as if from RPV assessed)
  - With small adjustments, the 9°C is the value used
- **97% of the data is bounded**

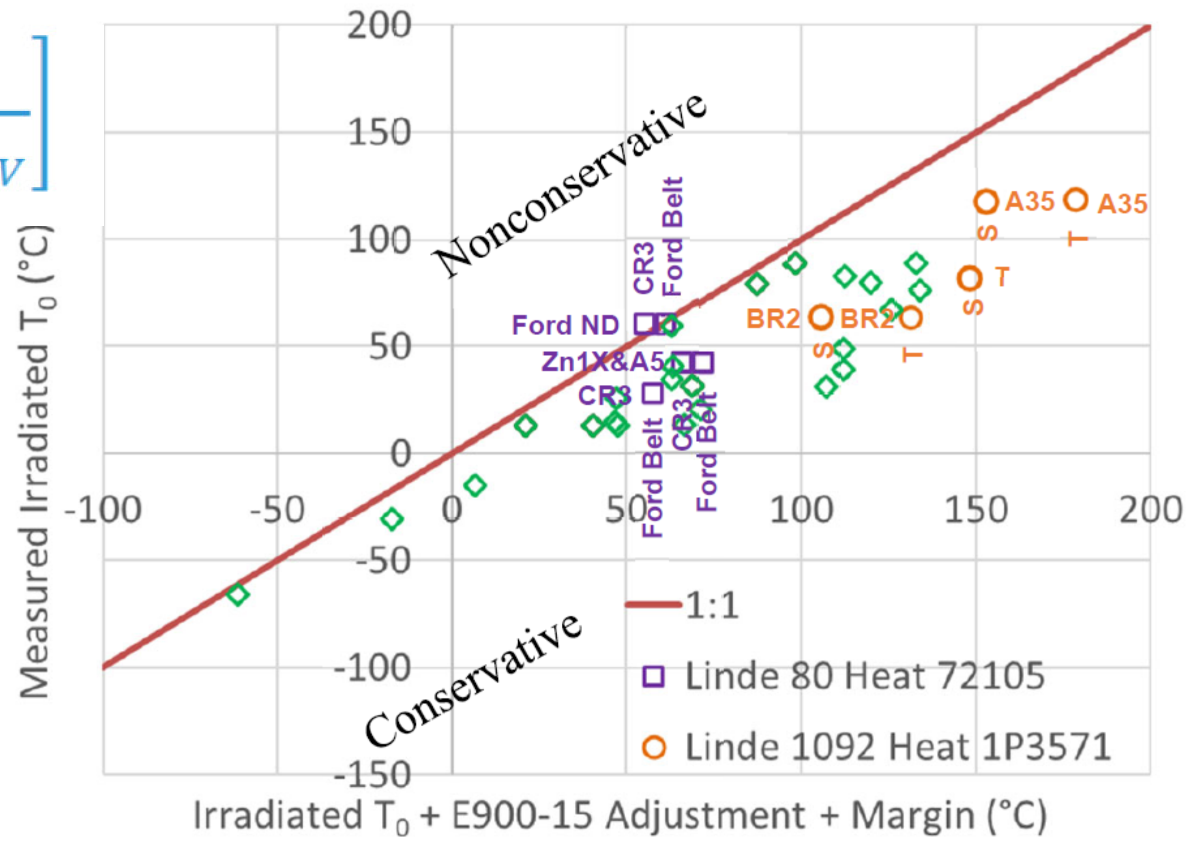


Figure 5 Bounding Adjusted T_0 Compared to Measured Irradiated T_0 for Weld Metals (horizontal labels indicate capsule names showing measured T_0; vertical labels indicate capsules from which measured T_0 was adjusted and margin added)

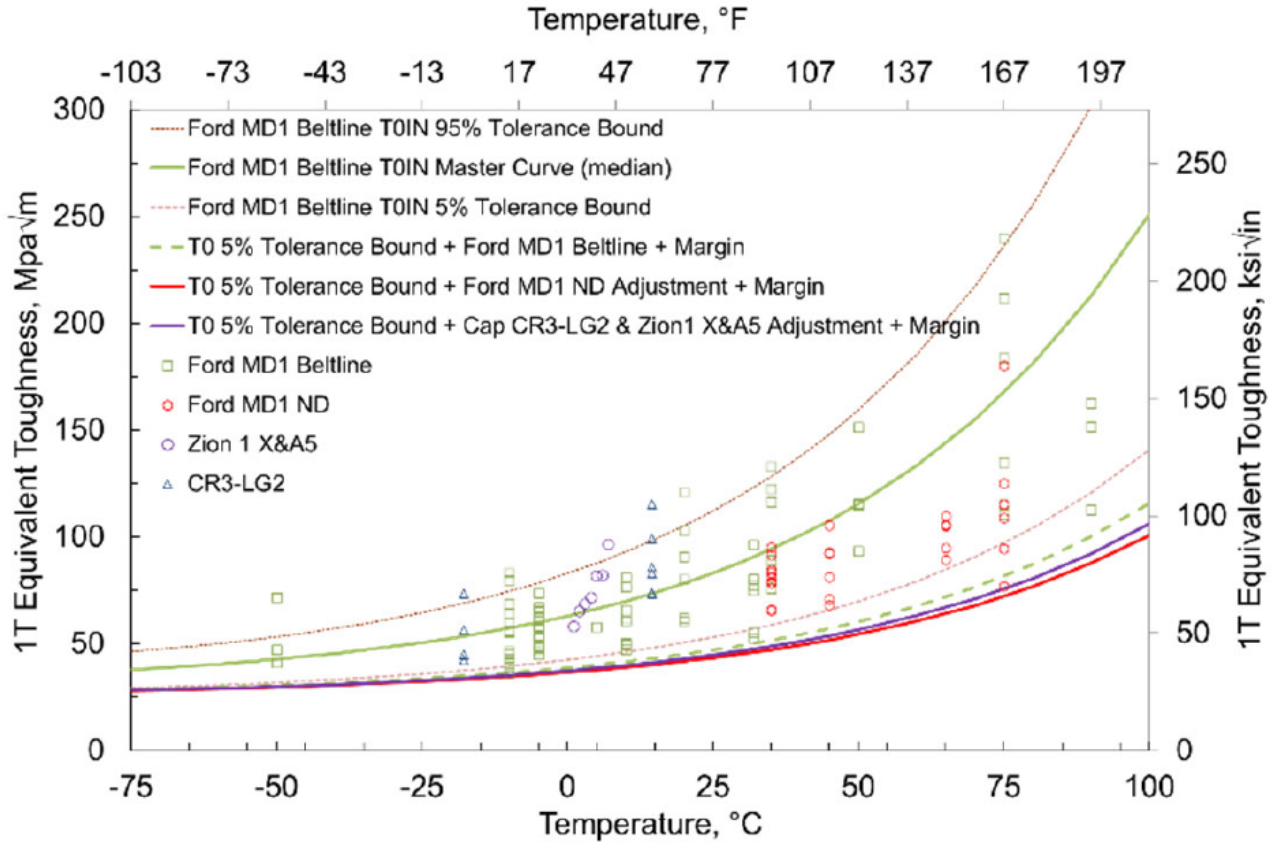


Figure 10 Comparison of Fracture Toughness Values to Bounding Curves for Weld Heat 72105 Adjusted from Ford Reactor MD1 Beltline T_{0IN}

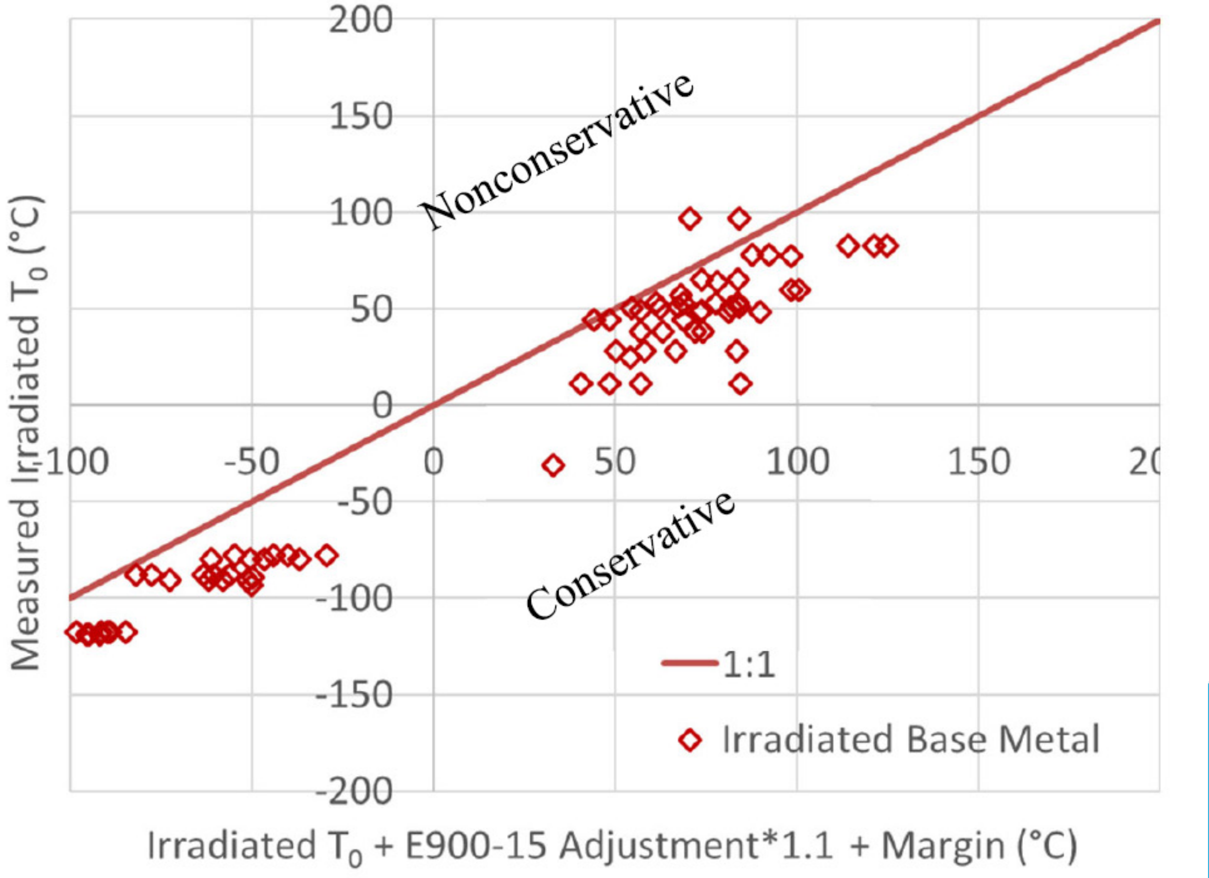


Figure 6 Bounding Adjusted T_0 Compared to Measured Irradiated T_0 for Base Metals

Basis: J. B. Hall, B. Golchert, and D. Simpson, “An Examination of Margins Needed to Ensure Conservative Application of T0 to RPV Fracture Toughness,”

ASME PVP2024-125225

PWROG-18068 Summary

The benefits of an irradiated direct fracture toughness data evaluation methodology are:

- Establishes a robust fracture toughness basis ensuring public health and safety by reducing uncertainty and enabling a statistical understanding of the actual irradiated RPV fracture toughness
- Specifically, this topical report discusses a methodology to:
  - Determine the ductile-brittle transition reference temperature (T_0)
  - Adjust the data for differences between the tested material and the RPV component of interest
  - Account for test result, adjustment and input uncertainties and material variability in the respective RPV component
  - Apply the data using the ASME Section XI Code.

Next Steps

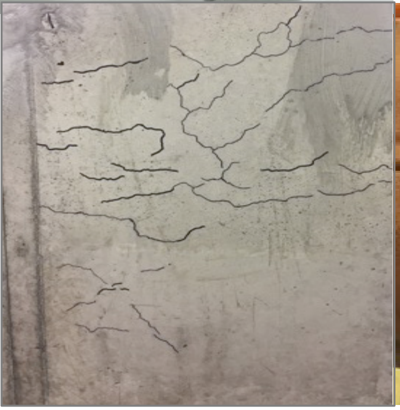
- Final RAI responses and PWROG-18068 markup submitted to NRC on March 8, 2024
  - NRC accession numbers: ML24068A101, ML24068A102, ML24068A103, ML24068A104, ML24068A105
- NRC draft safety evaluation expected in May
  - Review and provide comments
  - NRC then issues final safety evaluation (approved method utilities can use)
- Once approved via NRC safety evaluation
  - Submit pilot plant evaluations using existing T_0 data
  - Develop detailed test matrix
    - Select limiting materials most likely to benefit plants
    - Balance irradiated material testing cost vs. unirradiated vs. benefit

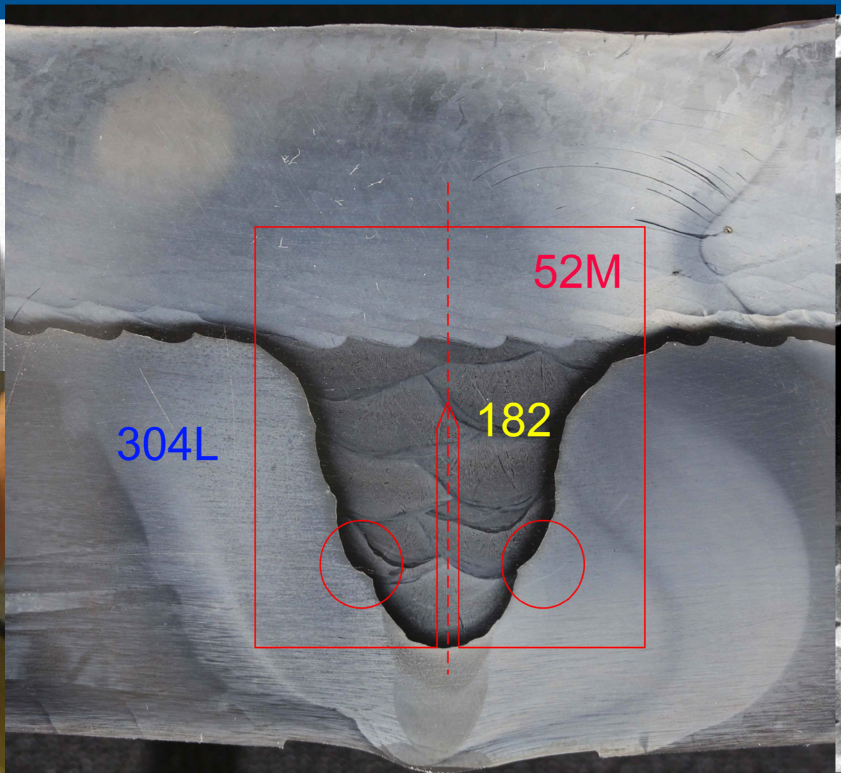
Collaboration Activities

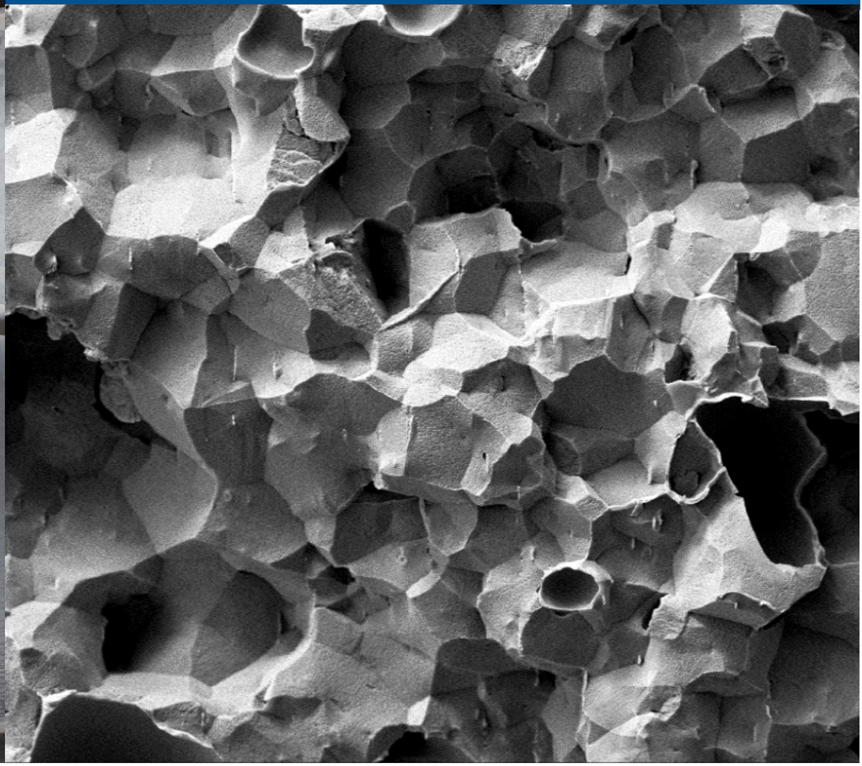
- **Recent**
  - Dr. Chen and Sokolov have attended PWROG materials committee meetings to listen to ongoing activities and present LWRS work
  - ORNL provided archive Palisades pressurizer weld for use in plant SLR application of direct fracture toughness
  - PWROG provided unirradiated archive Zion Unit 1 weld and plate to ORNL so that irradiated RPV beltline test results could be compared
  - Palisades high fluence capsule was withdrawn, shipped, disassembled with specimens sent to ORNL for testing
- **Future possibilities**
  - Test Zion Unit 1 surveillance capsule materials for T_0 to compare to RPV shell test results
  - Provide unirradiated archive Palisades weld and plate to ORNL so that irradiated high fluence capsule test results could be compared
  - Testing and expertise to help resolve observed ductile instabilities (test record crack jumps) when testing irradiated stainless and RPV steel on upper-shelf

Questions?

The Materials Committee is established to provide a forum for the identification and resolution of materials issues including their development, modification and implementation to enhance the safe, efficient operation of PWR plants.



Overview of NRC Materials Research Supporting Long-Term Operation

Jeff Poehler
Nuclear Regulatory Commission

LWRS Spring Program Review Meeting
May 1, 2024

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Materials and Aging Research

- Research objectives
  - Improve timeliness of regulatory decision-making on the use of new materials, manufacturing technologies, and inservice inspection techniques through independent and confirmatory research.
  - Address materials degradation during long-term plant operation.
  - Inform and enhance the use of risk information in regulatory decision making.
- Strategic Focus Areas
  - Support resolution of safety-significant technical issues
  - Maintain core capabilities to support emerging technical needs related to corrosion, metallurgy, component integrity assessment, and non-destructive examination
  - Enhance modeling/analytical tools to support efficient regulatory decision-making
  - Foster collaborations with domestic and international counterparts to stimulate information sharing and cooperative research approaches
- More information contained in U.S.NRC's Research Prospectus for Fiscal Years 2022 – 2024 (ML22235A651)

Long-Term Operation (LTO) & Aging Management

- **What are we doing?** Supporting guidance development, coordinating related research activities, developing a systematic approach for harvesting materials and components from reactors.
- **Motivation:** Provide assurance that aging effects will be adequately managed during LTO.
- **Regulatory Application:** Refine, as appropriate, existing aging management programs and guidance
- **Collaboration:** DOE and EPRI
  - Significant activities:
  - Draft report on knowledge gaps in online monitoring and structural health management for NPP LTO in FY24
  - Collaborating with EPRI on cables aging management workshop (June 13-14, 2024.)
  - Workshops on structural materials (metals and concrete) aging management for LTO (October 1-4, 2024)

Materials Harvesting

- **What are we doing?** Extracting materials (metallic, structural and electrical) from decommissioning or operating plants for laboratory testing.
- **Objective:** Improve understanding of material degradation associated with LTO, reduce uncertainty and unnecessary conservatism.
- **Motivation:** Harvested materials can confirm information on aging mechanisms generated through other research programs and operating experience.
- **Regulatory Application:** Inform aging management approaches for extended operation to ensure they are appropriate and adequate.
- **Collaboration:** DOE, EPRI, OECD/NEA, other international partners
- Significant activities:
  - OECD/NEA SMILE project (2021-2025), potential SMILE 2 project (2026-)
  - Updating NRC's harvesting priorities

Irradiation Assisted Degradation (IAD)

- **What are we doing?** Testing highly irradiated materials to characterize irradiation effects on fracture toughness and stress-corrosion cracking.
- **Motivation:** Confirm adequacy of reactor internals aging management programs.
- **Regulatory Application:** Support reviews of internals inspection/ evaluation guidance, ASME Code changes and associated rulemaking.
- **Collaboration:** EPRI, OECD/NEA, DOE
- Significant activities – Testing of Zorita RVI materials, participation in OECD/NEA SMILE and FIDES projects.

Primary Water Stress Corrosion Cracking (PWSCC)

- **What are we doing?** Mainly testing Alloy 690/52/152 crack growth rate (CGR) and initiation, and related evaluations.
- **Motivation:** Provide assurance of reactor coolant pressure boundary integrity
- **Regulatory Application:** Support reviews of proposed changes to the inspection requirements in the ASME Code and associated rulemaking
- **Collaboration:** EPRI, DOE
- **Significant activities:** CGR and initiation testing, participate in expert panels reviewing CGRs.

Steam Generator Tube Integrity Program (SG-TIP)

- **What are we doing?** Evaluating effectiveness of SG tube NDE.
- **Motivation:** Confirm adequacy of industry practices and new inspection approaches used for SG tube in-service inspections.
- **Regulatory Application:** Review acceptability of current and new approaches to inspection techniques plus changes to SG guidelines as proposed by industry
- **Collaboration:** EPRI, CNSC, KINS, KAERI, GRS, MPA, and IRSN
- **Significant activities:**
  - Report evaluating eddy current sizing capabilities for PWSCC at expansion transition regions of SG tubing.
  - Independent assessments of industry ET approach and probe-probe equivalency.

Probabilistic Integrity Assessment

- **What are we doing?** Developing probabilistic methods to assess structural integrity of RPV and piping components.
- **Motivation:** Confirm continued integrity of safety-critical components subject to degradation mechanisms
- **Regulatory Application:** Risk-inform regulatory decision-making on component integrity
- **Collaboration:** EPRI and CSNI (xLPR)
- **Significant activities:**
  - Development of the FAVPRO PFM code for RPV integrity (replace existing FAVOR code)
    - Issue FAVPRO v1.0.x and associated manuals and QA documentation (2024Q2)
  - Continued development and modernization of the xLPR code for probabilistic piping integrity.
    - Prob. Risk assessment of French SCC OE impact on US PWRs ML23236A080
    - Confirm LOCA frequency estimates from NUREG-1829

Piping Integrity

- **What are we doing?** Leak-before-break (LBB) and high energy line break (HELB) studies. Assess thermal aging embrittlement (TE) of cast austenitic stainless steel (CASS) and austenitic stainless steel welds (ASSW)
- **Motivation:** Confirm integrity of safety-critical piping systems during LTO
- **Regulatory Application:** Enhance guidance for piping structural integrity calculations
- **Collaboration:** EPRI and CSNI
- **Significant Activities:**
  - Developing alternative HELB framework for existing and new reactors and assessing risk-informed HELB.
  - Updating Flaw Evaluation Software (FES) for evaluating PWSCC in piping and CRDM nozzles
  - Reports on TE of CASS and ASSW.

Nondestructive Evaluation (NDE)

- **What are we doing?** Evaluating effectiveness and reliability of NDE techniques. Looking at application of machine learning (ML) to NDE.
- **Motivation:** Confirm adequacy of industry procedures and practices
- **Regulatory Application:** Support reviews of ASME Code modifications and proposed revisions of current requirements
- **Collaboration:** EPRI, IRSN, and PIONIC
- **Significant activities:**
  - Report on assessment of ML applied to UT NDE ML24046A150
  - Assess NDE capabilities for carbon fiber reinforced composite repairs (2025)
  - Assess the capabilities of machine learning and automated data analysis in NDE (2026)

Concrete Research

Overview

- **Objective:** Evaluate and improve concrete aging and performance for LTO
- **Motivation:** Confirmatory research for lower-knowledge aging mechanisms generated through other research programs and operating experience and potential higher significance for LTO
- **Regulatory Application:** Inform aging management approaches for renewal of plant licenses to ensure they are appropriate and adequate
- **Collaboration:** DOE, EPRI, and other international partners
- POC: Madhumita Sircar (Madhumita.Sircar@nrc.gov)

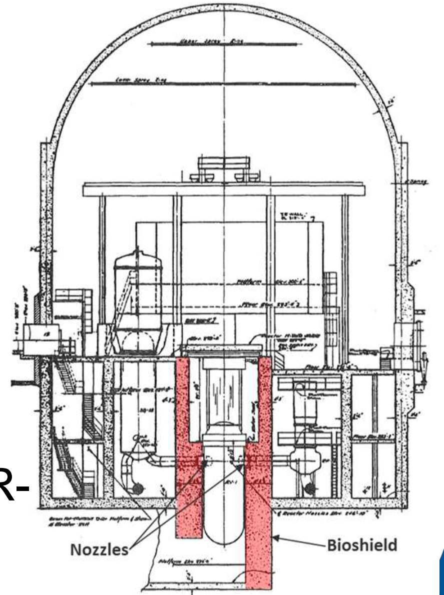
Significant Activities

Current:

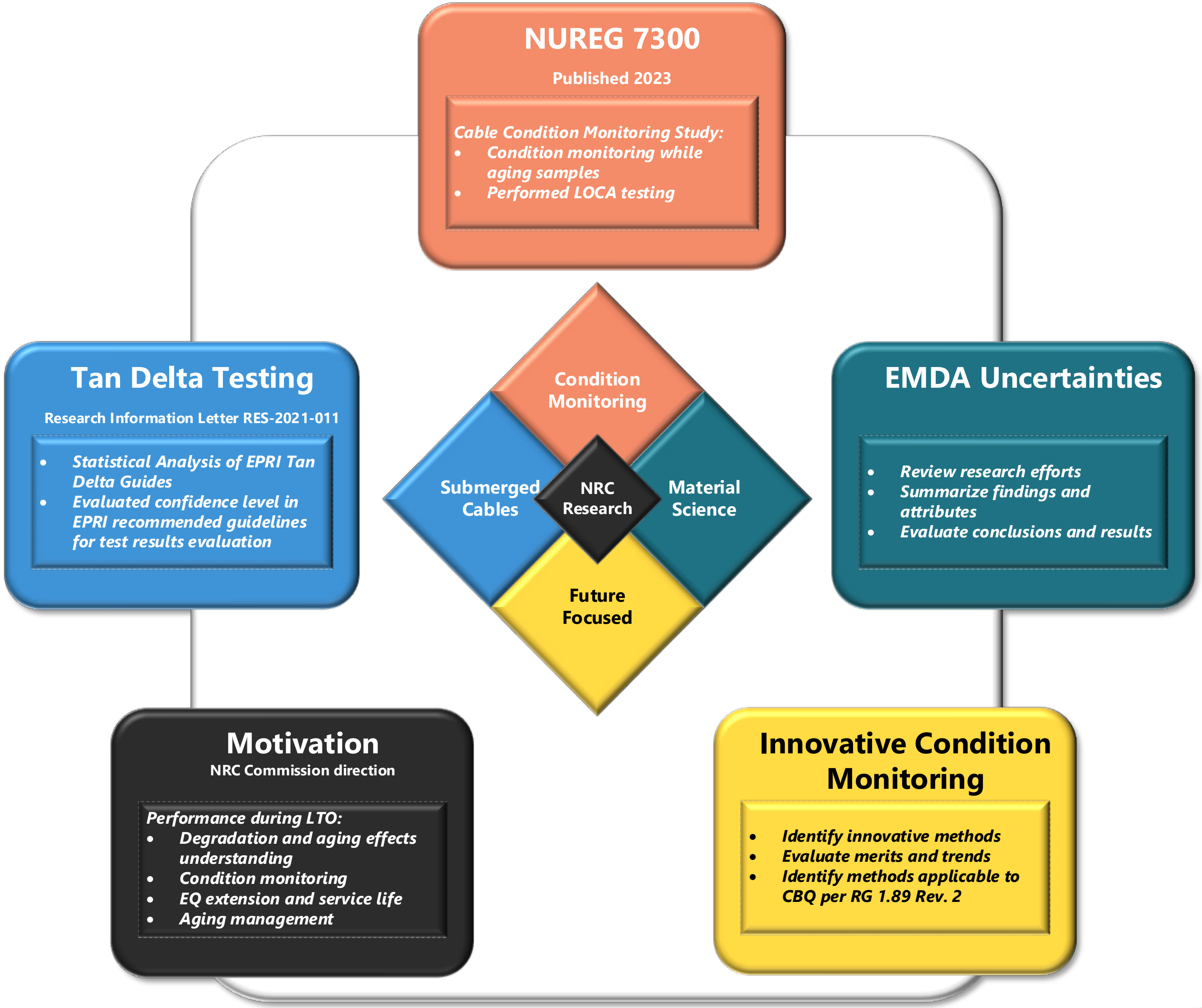
- Evaluate **effects of irradiation on concrete biological shield structures**. Experimental study to evaluate effects of irradiation on concrete-rebar bond (Using the LVR-15 reactor in the Czech Republic) and develop modeling methodology. **Report to be completed in FY2024.**
- Exploring **harvesting of irradiated concrete** materials
- Study on **creep and shrinkage** effects on PCCVs

Completed:

- Reviewed radiation-induced degradation mechanisms and potential structural implications (NUREG/CR-7280)
- Evaluated neutron fluence, gamma dose and radiation energy deposition through concrete structures (NUREG/CR-7281)
- Creep and shrinkage effects on aging of post-tensioned containment vessels (PCCVs) (RIL-2022-06)
- NRC-NIST project on Alkali-Silica Reaction (Completed)



U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment



Overview of NRC Cables Research Program

Future Work – Important Considerations

- Identify and address materials degradation during LTO.
- Develop, maintain, and implement research strategies to obtain and evaluate domestic and international operating experience on age-related degradation:
  - Harvesting aged components
  - Engage external stakeholders
  - Leverage resources
- Assess aging management approaches appropriate for extended plant operation
- Develop targeted harvesting strategies.
- Conduct workshops on topics important to safety.

Summary

- NRC Office of Nuclear Regulatory Research conducts confirmatory research to establish technical bases that support regulatory decisions and development of regulatory guidance documents.
- NRC staff exchanges information with domestic and international counterparts on materials performance and aging management of nuclear power plant structures and components, and conducts independent analyses.
  - Research results
  - Operating experience
- Research activities are prioritized to address potential safety-significant technical issues.
- Long-lead-time confirmatory research is an important consideration in proactive aging management.
- For more information contact: Jeffrey.Poehler@nrc.gov

Published Reports on Concrete Research

- NUREG/CR–7280, “Review of Radiation Induced Concrete Degradation and Potential Implications for Structures Exposed to High, Long-Term Radiation Levels in Nuclear Power Plants”, Report December 2020, published July 2021
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7280/index.html

- RIL 2021-07, “Radiation Effects on Concrete – An Approach for Modeling Degradation of Concrete Properties”, Report December 2020, published August 2021
https://www.nrc.gov/docs/ML2123/ML21238A064.pdf

- NUREG/CR-7281, “Radiation Evaluation Methodology for Concrete Structures”, Report December 2020, published July 2021
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7281/index.html

- RIL 2022-06, “Aging of PCCV with Emphasis on Creep and Creep Rupture”, Report August 2021, published March 2022
https://publish.nrc.gov/docs/ML2207/ML22075A007.pdf

- RIL 2022-07, “Assessment of the San Onofre Concrete Susceptibility against Radiation Damage”, published April 2022
https://www.nrc.gov/docs/ML2211/ML22119A092.pdf

- SMiRT27 Paper, “Effects of Neutron Irradiation on the Bond Strength of Steel Embedded in Concrete
https://confit.atlas.jp/guide/event/smirt27/subject/Tu.4.H-02/detail

15

Published Reports on Concrete Research

Alkali-Silica Reaction (ASR) Research at NIST: Tasks and Reports

- Task 1: Assessing In-Situ Mechanical Properties of ASR-Affected Concrete (NIST Technical Note 2121, February 2021)
  https://www.nist.gov/publications/structural-performance-nuclear-power-plant-concrete-structures-affected-alkali-silica-0
- Task 2: Assessing Bond and Anchorage of Reinforcing Bars in ASR-Affected Concrete (NIST Technical Note 2127, February 2021)
  https://www.nist.gov/publications/structural-performance-nuclear-power-plant-concrete-structures-affected-alkali-silica
- Task 3: Effects on seismic response characteristics (NIST Technical Note 2180, January 2022)
  https://www.nist.gov/publications/structural-performance-nuclear-power-plant-concrete-structures-affected-alkali-silica-1
- Tasks 4 and 5: Design of concrete mixes for all tests, and prediction of future and ultimate expansion, and degradation and methods to assess degree of reaction (current state of material degradation)
  https://doi.org/10.6028/NIST.IR.8415
  https://nvlpubs.nist.gov/nistpubs/ir/2022/NIST.IR.8415.pdf

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment