



Status of IASCC Crack Growth Model & Advanced Radiation Resistant Materials (ARRM) Program

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EPRI IASCC Data Compilation & Analysis: Objectives

- Compile crack growth rate data on irradiated stainless steels developed by EPRI, NRC, Halden and other international programs
- Convene an Expert Panel to review and screen the available CGR data on irradiated materials using appropriate screening criteria
- Develop crack growth models and disposition curves for application to BWR and PWR internals
- This work is coordinated between PSCR, BWRVIP and MRP Programs at EPRI

EPRI IASCC Expert Panel

- Industry Expert Panel Members:
 - *Peter Andresen, GE-GRC*
 - *Steve Fyfitch, AREVA*
 - *Rich Jacko, Westinghouse*
 - *Ron Horn, GE Hitachi*
 - *Torill Karlsen, IFE Halden*
 - *Pal Efsing, Vattenfall*
 - *Anders Jenssen, Studsvik*
 - *Peter Scott, Consultant*
- EPRI Participants
 - *Raj Pathania*
 - *Bob Carter*
 - *Robin Dyle*
 - *Jean Smith*
- Principal Investigator
 - *Ernie Eason, Modeling & Computing Services, LLC*

EPRI IASCC Expert Panel Meeting: Screening Criteria

- Review original crack length (a) vs. time (t) curves for each specimen and test segment in the database
- For each test segment consider stress intensity (K) validity for irradiated materials vs. ASTM E 399 and alternative criteria
- Crack extension (da)
- Test segment interval (dt)
- Variation of K with crack dK/da
- Constant load or K vs. partial unloading
- Load ratio (R)
- Correlation coefficient on the linear fit to crack growth rate over a given test segment
- The consensus was that these criteria should not be applied individually without considering the entire context of the test

Range of Selected Variables in the Low-ECP Calibration Data

Variable	Symbol	Range	Units
Crack growth rate	da/dt or	2.2×10^{-9} to 3.1×10^{-5}	mm/s
Dose	dose	0.23 to 47.5	dpa
SCC test temperature	T	275 to 340	°C
Irradiated yield stress	$\sigma_{0.2}$	257 to 1111	MPa
Stress intensity factor	K	6.5 to 32.9	MPa \sqrt{m}
Electrochemical potential	ECP	-359 to -894	mV _{SHE}
Dissolved H ₂ concentration	[H ₂]	0.02 to 3.29	ppm
Outlet conductivity	κ	0.06 to ≤ 0.3 (HWC only)	mV _{SHE}
Lithium concentration	[Li]	1.88 to 3 (PWR only)	ppm
Boron concentration	[B]	500 to 1298 (PWR only)	ppm

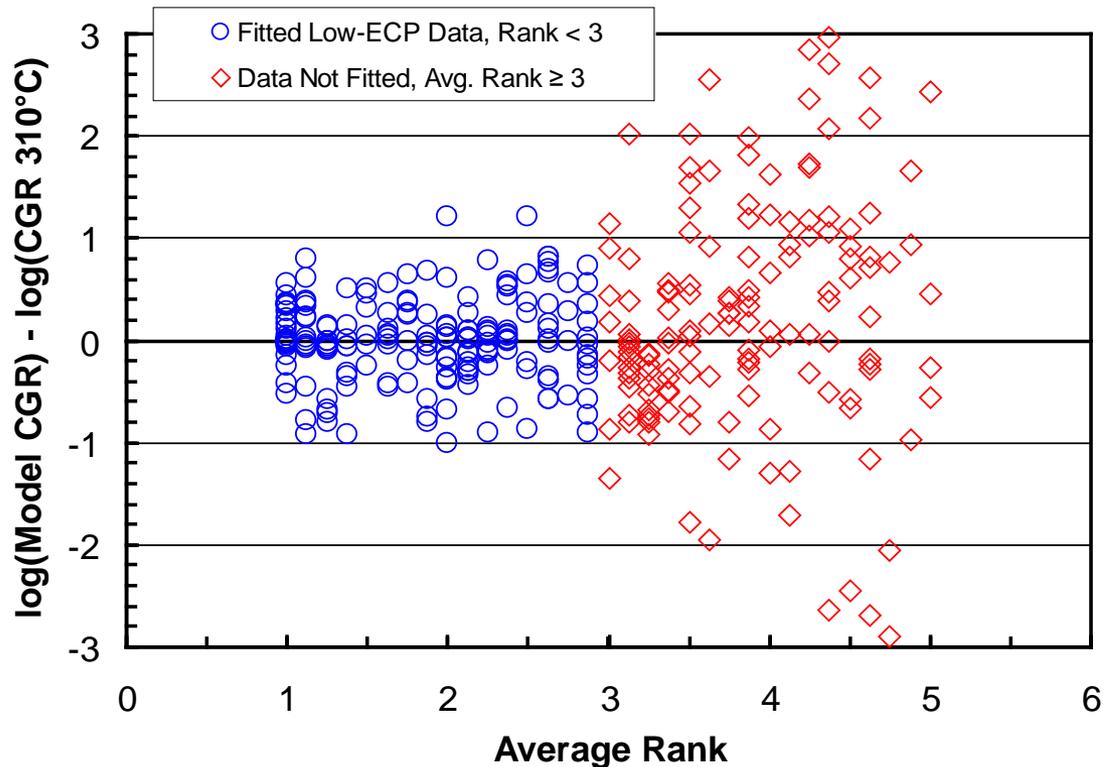
Range of Selected Variables in the NWC Calibration Data

Variable	Symbol	Range	Units
Crack growth rate	da/dt or	7.7×10^{-9} to 1.4×10^{-5}	mm/s
Dose	dose	0.21 to 47.5	dpa
SCC test temperature	T	272 to 330	°C
Irradiated yield stress	$\sigma_{0.2}$	254 to 925	MPa
Stress intensity factor	K	3.85 to 28.7	MPa \sqrt{m}
Electrochemical potential	ECP	70 to 273	mV _{SHE}
Dissolved O ₂ concentration	[O ₂]	0.2 to 32	ppm
Outlet conductivity	κ	0.08 to 0.3	$\mu\text{S/cm}$

EPRI IASCC Expert Panel Status

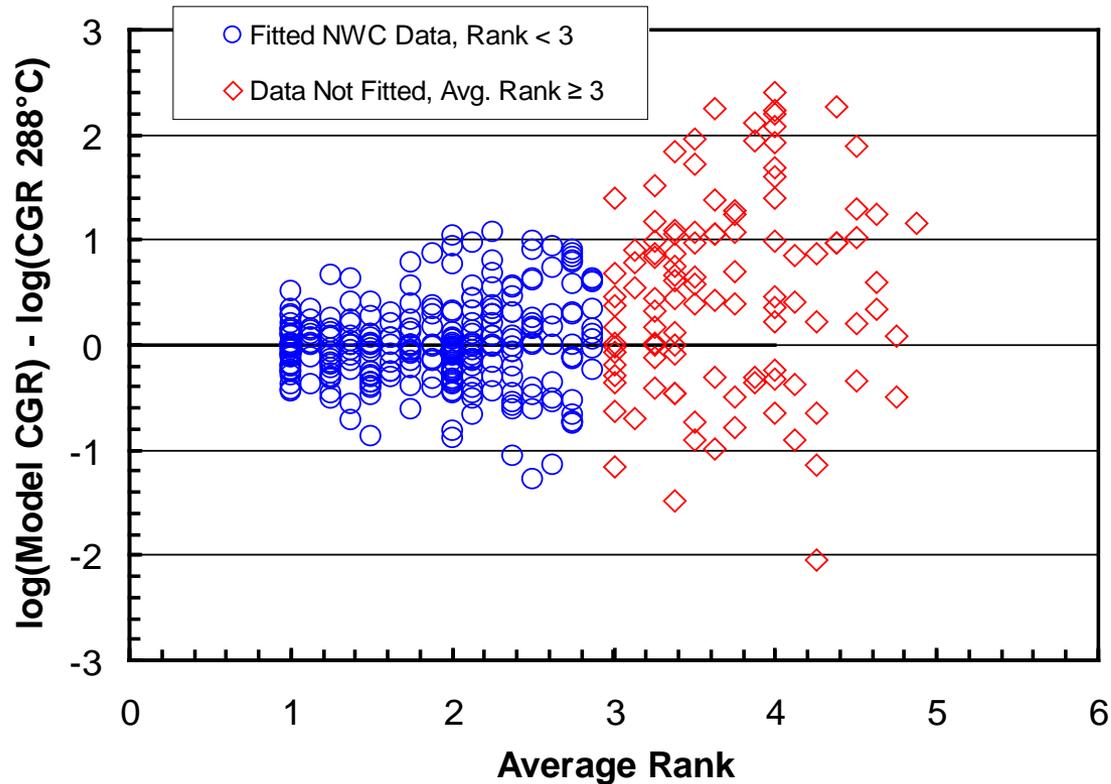
- The database includes more than 1600 test segments including those under cyclic loading
- For IASCC crack growth rates only test segments under constant load or periodic partial unloading were considered
- The reviewers ranked the test segments on a scale of 1 (best) to 5 (unsuitable) after examining the raw data from crack length vs. time plots
- Data with average ranks from 1 to 3 was considered suitable for development for IASCC crack growth models for BWR NWC, HWC and PWR environments

Low-ECP Residuals vs. Average Rank



- 187 CGR observations in proposed model (73 in HWC & 114 in PWR primary water)
- 22 heats
- 52 specimens
- Tested in 6 laboratories

NWC Residuals vs. Average Rank



- 283 CGR Observations in proposed model
- 22 Heats
- 56 Specimens
- Tested in 5 laboratories

Alternative Yield Stress Term

- Dose and irradiated yield stress are strongly correlated, as shown in the next slide
- Yield stress is known to strongly affect SCC CGR, in both unirradiated and irradiated conditions
- Irradiated yield stress can be measured or estimated from the correlation with dose (MRP-135 Rev1)
- For the above reasons, yield stress based models were developed as an alternative to the dose models

High and Low ECP Models

- Overall form:

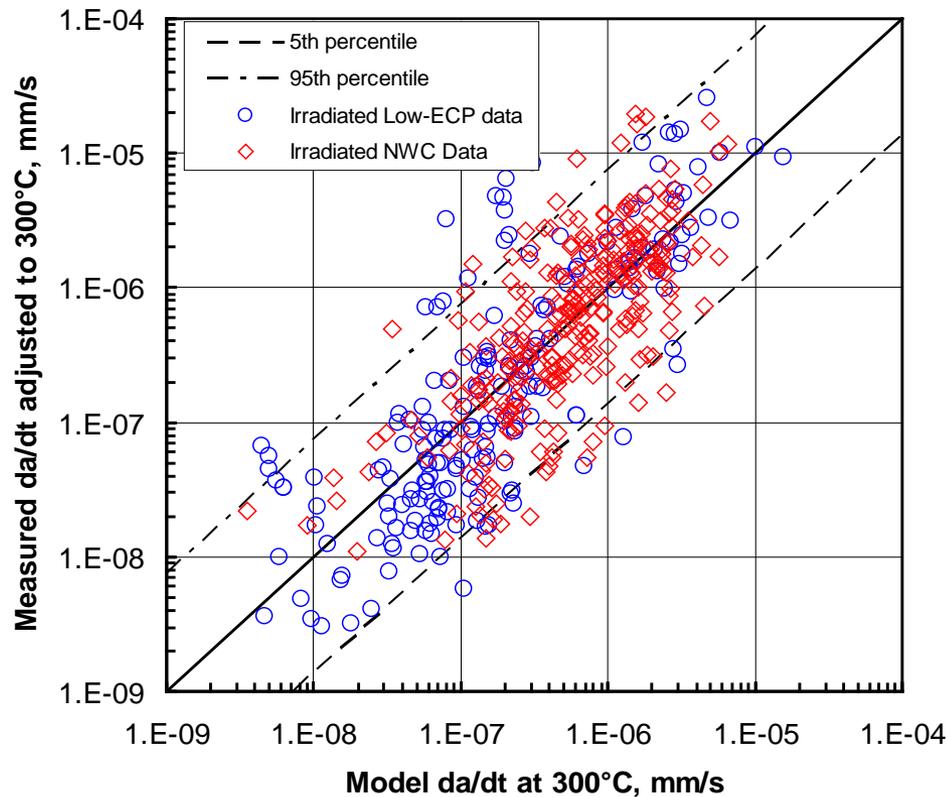
$$\frac{da}{dt} = C_i \cdot f(PPU) \cdot \sigma_{0.2}^v \cdot \exp\left[-\frac{Q}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right] \cdot K^u$$

- da/dt is crack growth rate (mm/s)
- C_i is a material/environment coefficient calibrated to each specimen or heat
- $f(PPU)$ partial periodic unloading constant = CGR_{PPU}/CGR_{CL}
- $\sigma_{0.2}$ is the irradiated yield stress at the temperature and dose of the IASCC test specimen (MPa)
- Q is the activation energy (kJ/mol)
- R is the universal gas constant 8.314 J/(mol-K)
- T is the SCC test or model application temperature
- T_{ref} is a reference temperature at which the model is calibrated
 - $T_{ref} = 583.15$ kelvin (310°C) for low-ECP data
 - $T_{ref} = 561.15$ kelvin (288°C) for NWC data
- K is the stress intensity factor, (MPa \sqrt{m})
- v and u are fitted exponents

EPRI IASCC Expert Panel: Draft Low and High ECP Models

- The models account for effects of dose, stress intensity K , ECP, temperature, type of loading (constant load vs. PPU)
- The CGRs under PPU were approximately 2X higher than under constant load
- After accounting for above parameters there is still a significant heat to heat (or specimen to specimen variability) reflected in coefficient C_i
- A similar heat to heat variability has been observed in Ni-base alloys (MRP-55 and MRP-115)
- There was no significant difference in the CGRs between BOR 60 and LWR irradiated materials after accounting for the above factors
- Analysis showed that a common model can be used for BWR-HWC and PWR environments with a temperature term to account for higher PWR temperatures
- The ratio between NWC CGR and low-ECP CGR, is a factor of ~ 6 for mean coefficients and a factor of ~ 4.5 for 75% coefficients

Overall fit of proposed model



Based on Heat Coefficients

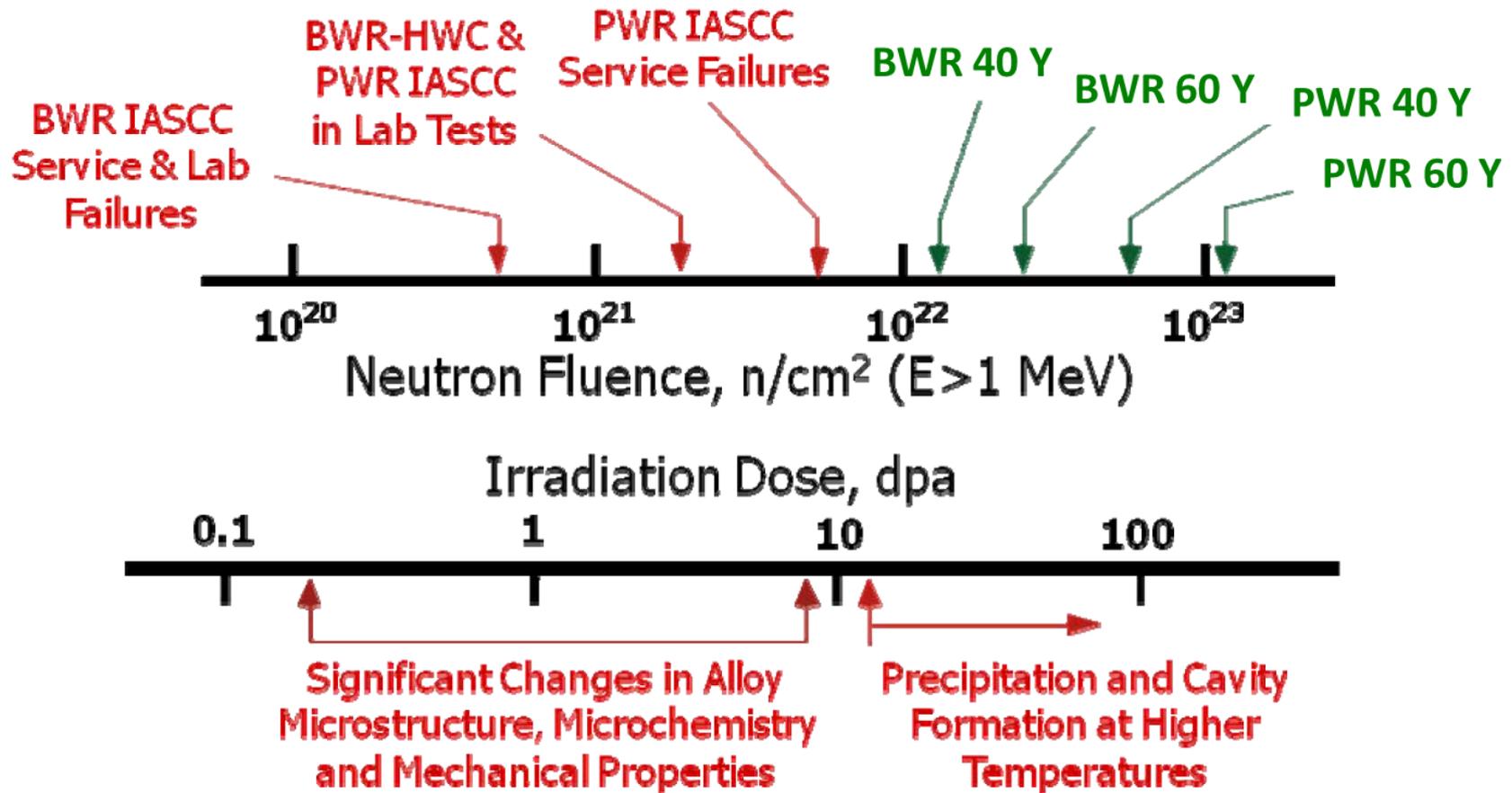
EPRI IASCC Crack Growth Report–Status

- Prepare a draft report on the IASCC database and low ECP (PWR and HWC) and high ECP (NWC) IASCC models for Expert Panel /EPRI review-*completed*
- Expert panel conference call to discuss and resolve comments-*completed*
- Address expert panel comments, revise report and send to PSCR, MRP and BWRVIP committees for review: *July-August 2014*
- Resolve industry comments, revise report and publish by end of 2014
- The report will provide the technical basis for crack growth disposition curves for irradiated BWR and PWR stainless steel internals
- Expect to provide report to NRC for review in 2015



Advanced Radiation Resistant Materials (ARRM) Program

Irradiation Induced Degradation of Austenitic Stainless Steels



Objectives of ARRM Program

- EPRI and the U.S. Department of Energy (DOE) are initiating a global, collaborative research effort to develop the next generation of materials for in-core structural components and fasteners.
- The two primary research goals are:
 - By 2022, to develop and test a degradation-resistant alloy that is within current commercial alloy specifications
 - By 2024, to develop and test a new advanced alloy with superior degradation resistance

ARRM Critical Issues Report and Roadmap

- EPRI and DOE have published “Critical Issues Report and Roadmap for Advanced Radiation Resistant Materials Program”, (EPRI Report 1026482, December, 2012)
- Objectives:
 - Identify candidate materials for use in structural applications in LWR reactor internals that have increased resistance to irradiation-induced degradation compared to currently used materials
 - Develop a systematic test program for further evaluation of these candidate materials in a phased manner, including irradiation out to high neutron doses

Commercial Alloys for Further Development

Low Strength Applications

- Hastelloy C22
- Alloy 690
- Alloy 625 (solution-annealed, not precipitation-hardened)
- Alloys 800/800H/825
- Alloys 309/310
- 12Cr F-M alloys (Optimized HT9)
- 9Cr F-M alloys (NF616)
- Zr-2.5Nb

High Strength Applications

- Alloy 625 Plus (precipitation-hardened condition)
- Alloy 725/718

Advanced Alloys for Further Development

Low Strength Applications

- Ti Alloys (Grade 26 + 0.1Ru)
- Higher Cr (>14Cr) F/M alloys (Type 439)
- ODS alloys (8-14Cr and Higher Cr with Al) (14YWT)
- High Cr, High Al ODS Alloys (?)

High Strength Applications

- ODS alloys (8-14Cr and Higher Cr with Al) (14YWT)

Advanced alloys will need ASME code qualification

Key Deliverables

- Critical Issues Report and Roadmap for the Advanced Radiation-Resistant Materials Program (Report 1026482, December 2012).
- Interim report documenting the results of microstructural, mechanical and stress corrosion cracking studies on proton irradiated alloys to identify promising alloys for further evaluation (December 2015).
- Final report recommending alloys for further evaluation under irradiation (December 2017).
- Interim report documenting microstructural, mechanical and stress corrosion cracking studies on neutron irradiated alloys (December 2019).
- Final report on qualification of radiation resistant commercial alloys for LWR internals (December 2022)



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