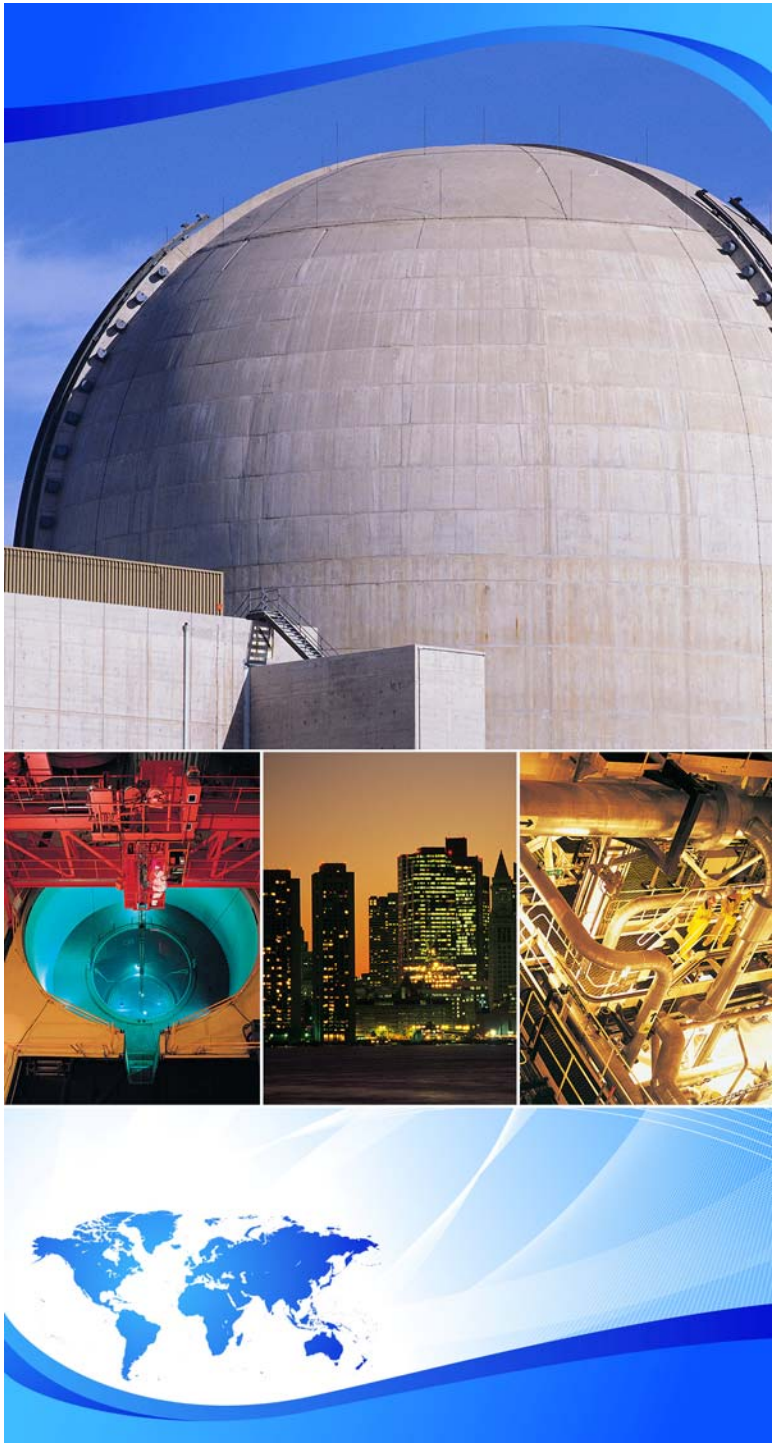

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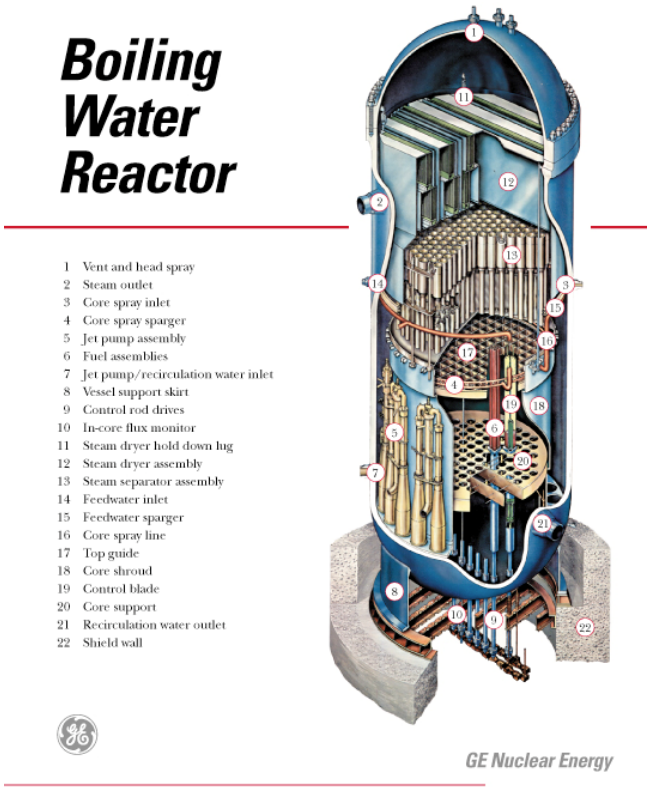
Testing of Irradiated Alloys X-750 and XM-19: Motivation for the Collaboration with Idaho National Laboratory

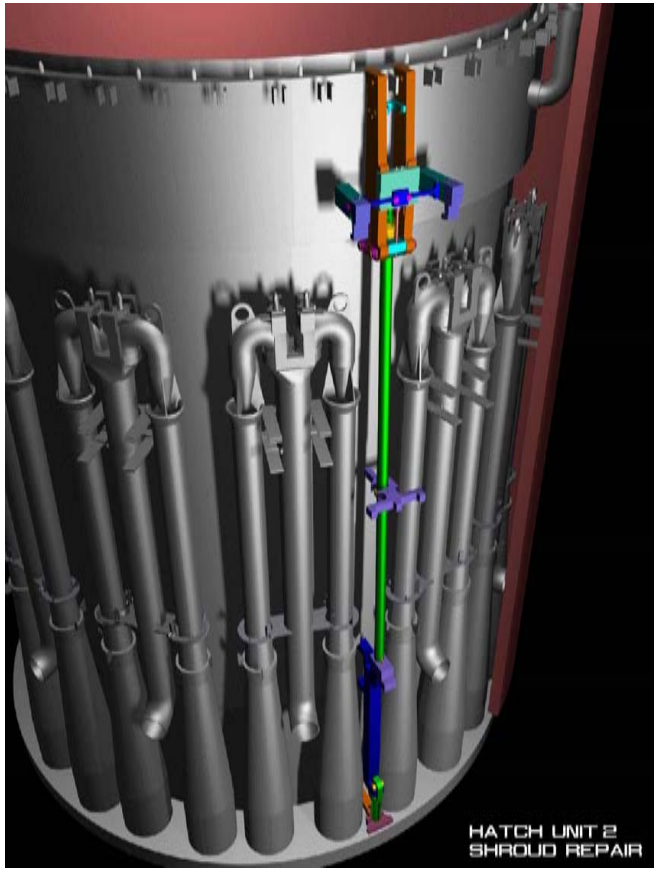
ATR NSUF Users' Week
Idaho National Laboratory
June 1, 2009

Peter Chou, Bob Carter, Raj Pathania

Irradiation Effects : X-750 and XM-19

- X-750 and XM-19 are used in repair hardware of BWR core shrouds





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Core Shroud Repairs: A Snapshot

- 14 U.S. plants and several international plants have installed tie-rod repairs (initial installations in the mid-1990's)
- Designed to remain in reactor through 60+ years of plant operation
- Effects of irradiation on fracture toughness and crack growth rates (NWC and HWC) are very limited
- Cost to replace all installed repairs could be well in excess of $100M

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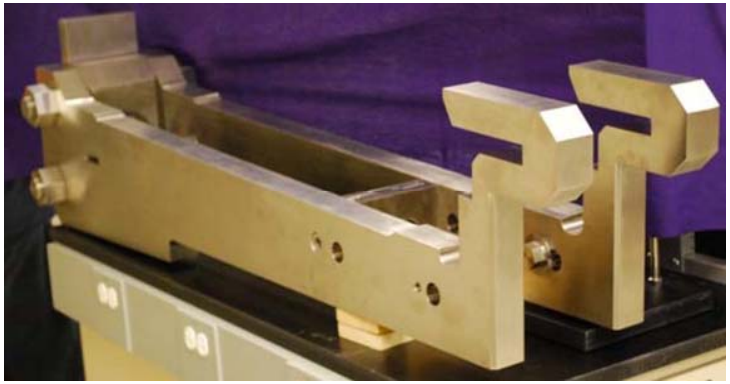
The Alloy X-750 Material to be Tested

- Higher-strength, precipitation-hardened Ni-base alloy
- Material from spare tie-rod-repair bracket
  - Documented Heat Treatment:
    - Temperature/duration: as in HTH Condition
    - No record of HTH "35% reduction minimum on last roll"



	Al	C	Co	Cr	Cu	Fe	Mn	Nb	Nb+Ta	Ni	Ni+Co	P	S	Si	Ta	Ti	???
X-750: UNS N07750	0.40-1.0	0.08 max	1 max(1)	14-17	0.5 max	5-9	1 max		0.7-1.2		70 min		0.01 max	0.5 max		2.25-2.75	
Measured: Heat 2750 5 7656 (2)	0.77	0.04	0.726	14.99	0.0151	7.8063	0.197	0.98	0.99	70.83	71.55	<0.005	0.002	0.253	<0.01	2.42	0.9762
Alloy 600: UNS N06600		0.15 max		14-17	0.5 max	6-10	1 max				72 min		0.015 max	0.5 max			

(1) determination not required for routine acceptance (2) measurement conducted on billet end

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The XM-19 Material to be Tested

- Higher-strength nitrogen-strengthened stainless steel
- Plate Material: 3.25" X 14.25" X 12"
  - Purchased from GEH
  - Acquired under specifications that have been applied to plant components

	C	Co	Cr	Fe	Mn	Mo	N	Nb	Nb+Ta	Ni	P	S	Si	V
XM-19 UNS S20910	0.06 max		20.5-23.5	balance	4.0-6.0	1.50-3.00	0.20-0.40	0.10-0.30		11.5-13.5			1 max	0.10-0.30
Measured: Heat A17509	0.032	0.03	20.8	balance	4.53	2.05	0.30	0.18	0.20	12.70	0.028	0.003	0.42	0.19
304L	0.03 max		18-20	balance	2 max					8-12	0.045 max	0.03 max	1 max	
316L	0.03 max		16-18	balance	2 max	2-3				10-14	0.045 max	0.03 max	1 max	
347	0.08 max		17-19	balance	2 max				10xC min-1 max	9-13	0.045 max	0.03 max	1 max	

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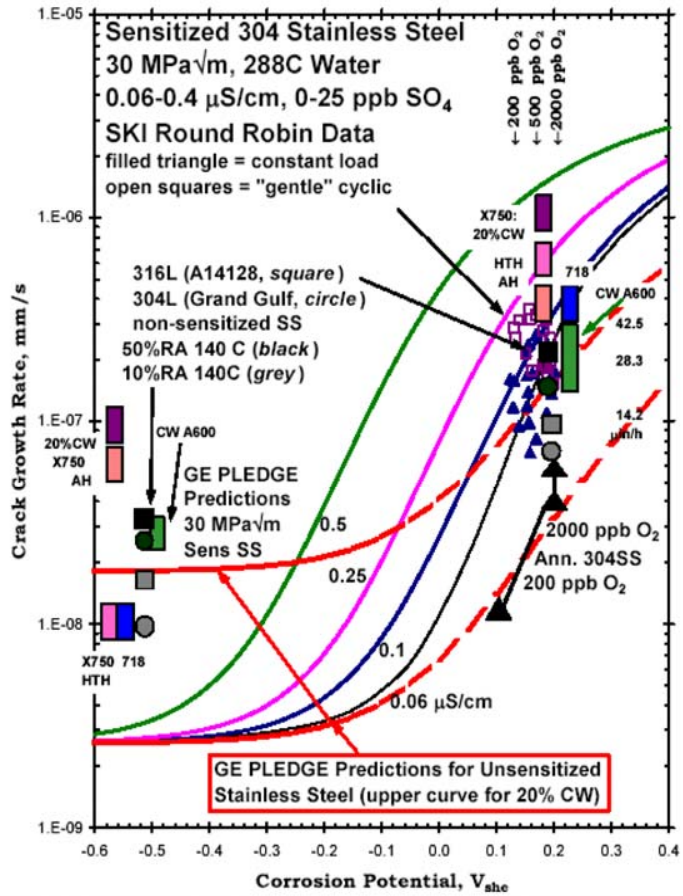
Alloy X-750: Unirradiated CGR



P. Andresen et al., 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 10-14, 2003, Stevenson, Washington

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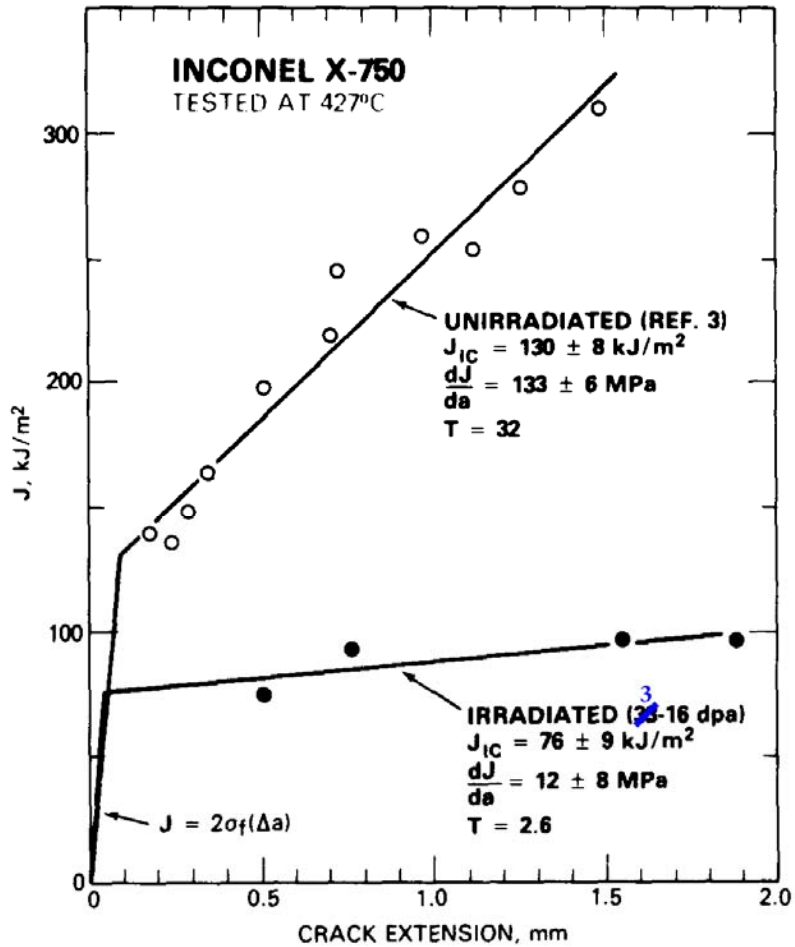
Alloy X-750: Irradiated Fracture Toughness Data… But at 427°C (3-16 dpa)



J–R curves for unirradiated and irradiated Inconel X-750.
J. Mills, Engineering Fracture Mechanics, Vol. 18, No. 3 (1983) pp. 601-607

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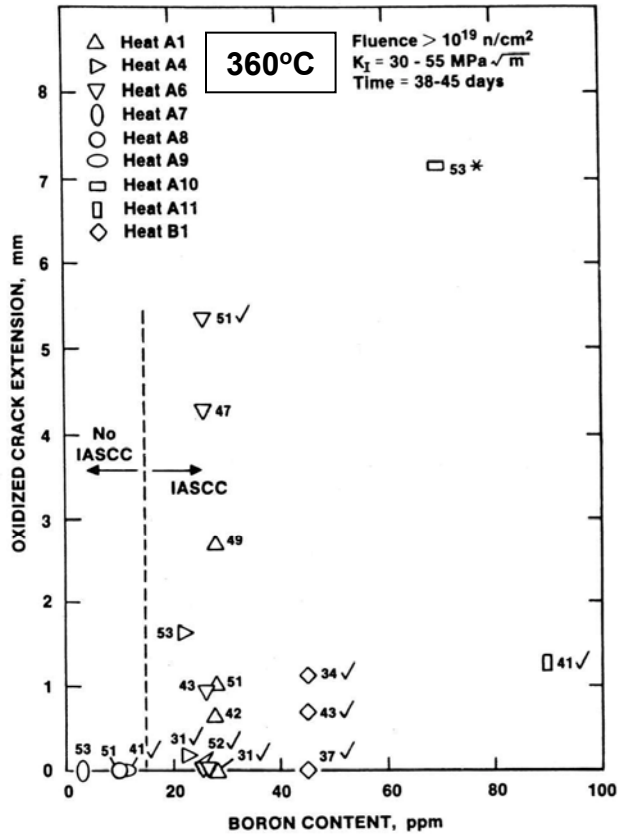
Alloy X-750: Higher IASCC Susceptibility linked to He from B^{10} Transmutation… Studies in the Right Temperature Range, But No Crack Growth Rates

- Fluence above 10^{19} n/cm^2 (E>1 MeV) [0.014 dpa] can degrade SCC performance
- Strong influence of Boron segregated at grain boundaries
  - Therefore, correlation with bulk Boron concentration
- Deleterious effect of He (α) from Boron transmutation: $B^{10}(n, \alpha)Li^7$



Extent of IASCC in HTH heats with various boron contents. The initial K_I value is provided next to each data point. Check denotes 38 day exposure and asterisk indicates that LTCP may have occurred during interim cooldowns.

W. J. Mills et al., 6th International Symposium on Environmenta Degradation of Materials in Nuclear Power Systems - Water Reactors, R. E. Gold and E. P. Simonen, eds., TMS, 1993

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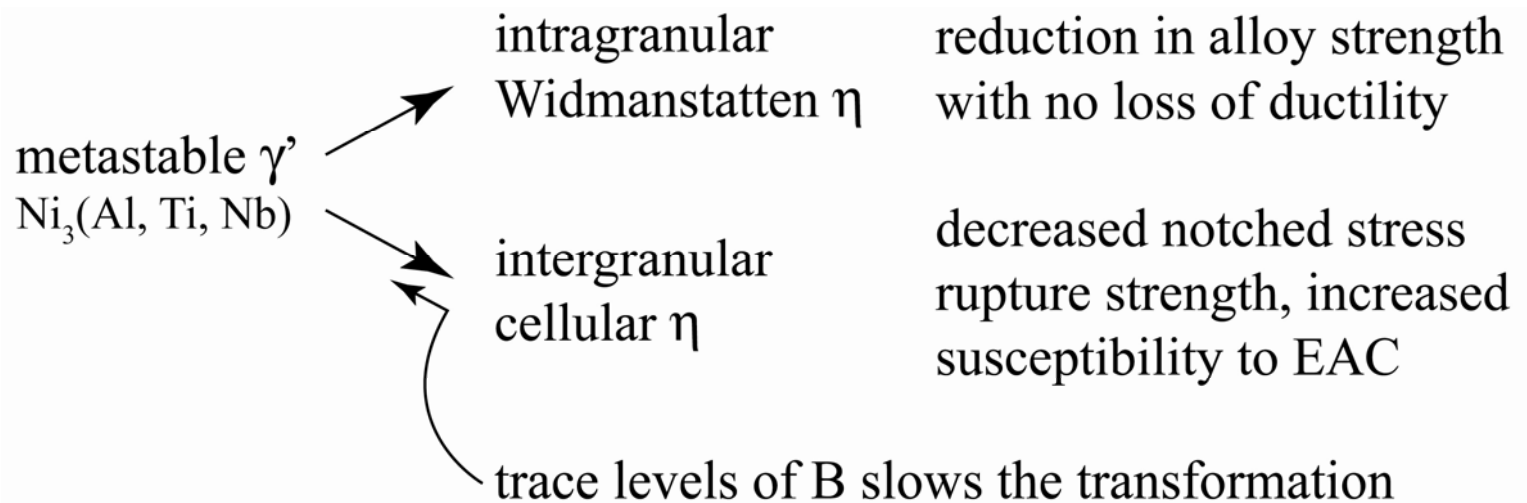
Alloy X-750: Boron Added Because…

- Trace Amounts of Boron in Ni-base superalloys improves
  - Hot working properties
  - Creep Properties
  - Notch Rupture Strength



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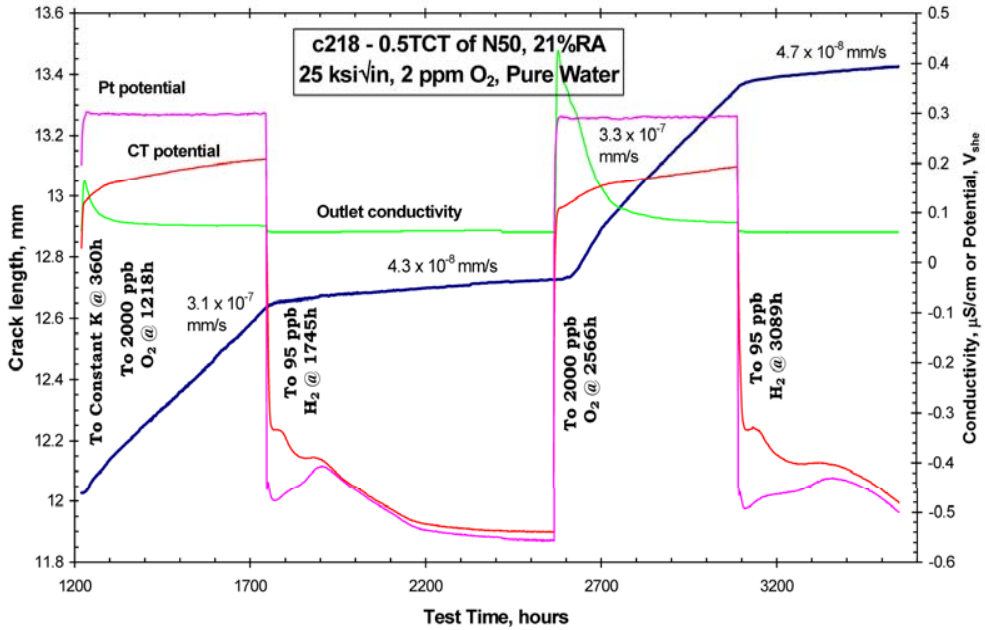
XM-19: Unirradiated CGR

Nitronic 50 (XM-19) +20% CW @140C



Growth rate behavior similar to other SSs

P. Andresen et al., 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 10-14, 2003, Stevenson, Washington

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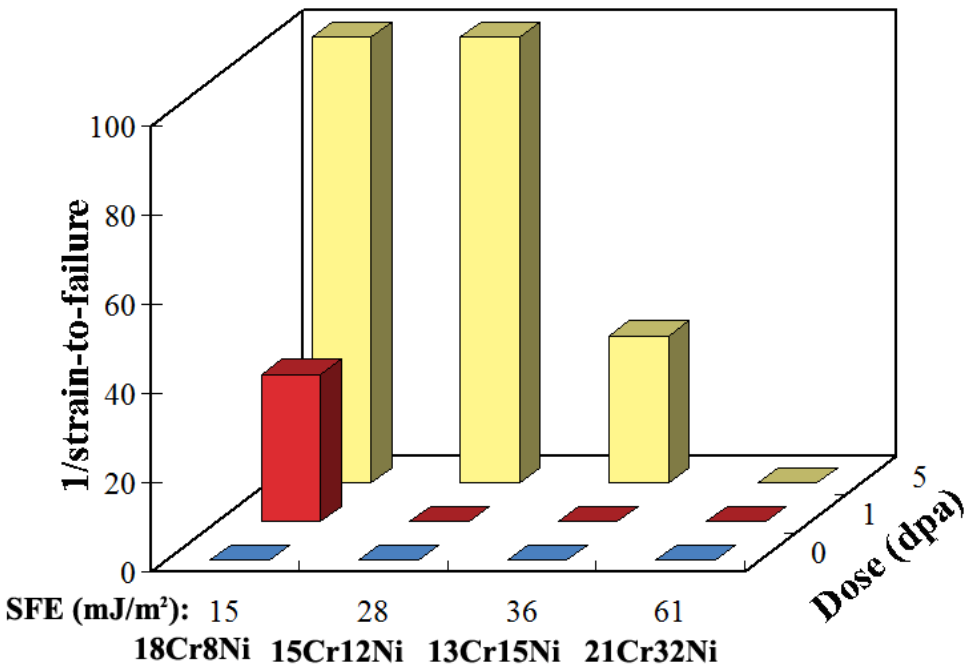
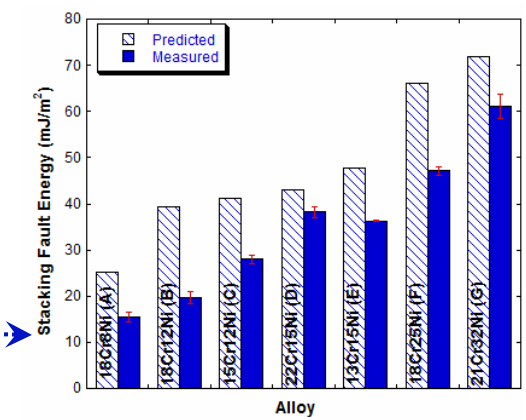
XM-19: Influence of N? ↔ Stacking Fault Energy, Localized Deformation, and IASCC Susceptibility



- **Low SFE alloy 18Cr8Ni: susceptible to IASCC at 1 dpa**
- **High SFE alloy 21Cr32Ni: resistant to IASCC even at 5 dpa**

XM-19 Material: 11.5 mJ/m²



Predicted and measured SFE

Predicted using Pickering's equation:

SFE = 25.7 + 2%Ni + 410%C - 0.9%Cr - 77%N -13%Si - 1.2%Mn

Courtesy of G. Was, University of Michigan

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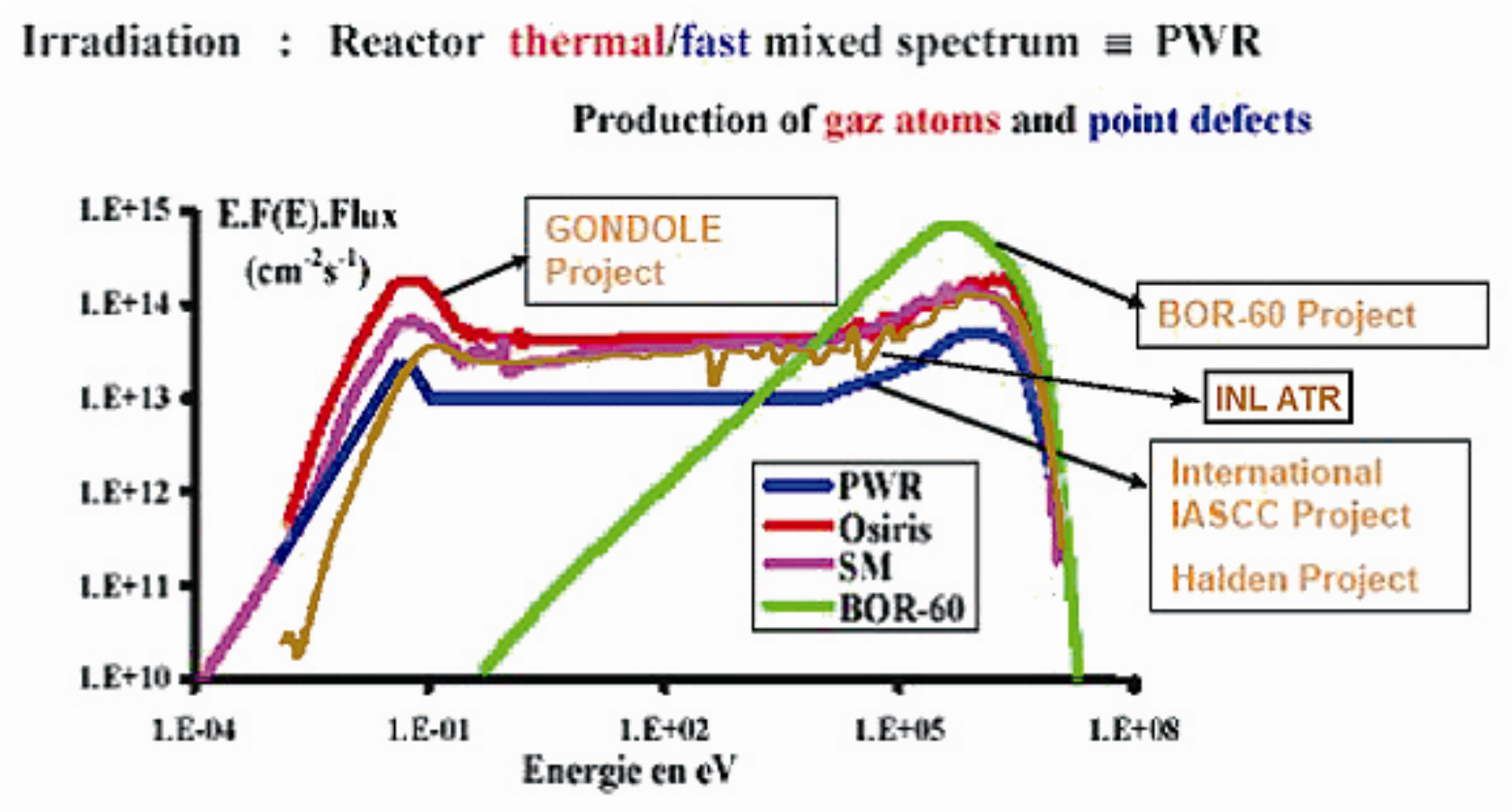
Water-Moderated (vs. Fast) Test Reactors Provide Thermal Neutrons for Boron Transmutation (Helium Production)



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INL Irradiation Program

- X-750 and XM-19 will be irradiated at the INL Advance Test Reactor (ATR) Loop 2A
- Temperature control using water
  - Temperature = 288°C
  - Conductivity = 0.15-0.20 μS/cm
  - Deaerated
- Target Upper Value Fluence is 1.0×10^{21} n/cm^2 (E>1MeV)
  - Estimates of Bounding Fluence at Core Shroud OD:
    - 40-years: 1.7×10^{21} n/cm^2
    - 60-years :2.6×10^{21} n/cm^2
  - Take some allowance for the water gap between the components and shroud OD and/or for components not at peak flux location.

Experiment Designation Capsule ID	ATR Insertion	Approximate Fluence (E> 1 MeV)	Estimated ATR Removal Date	
EPRI-1	July 2010 (147B)	5.0×10^{19}	July 2010*	0.071 dpa
EPRI-2	August 2010 (148A)	2.0×10^{20}	October 2010	0.29 dpa
EPRI-3	July 2010 (147B)	1.0×10^{21}	February 2011	1.4 dpa

*14 day cycle required to meet low fluence goal may be delayed until July 2011

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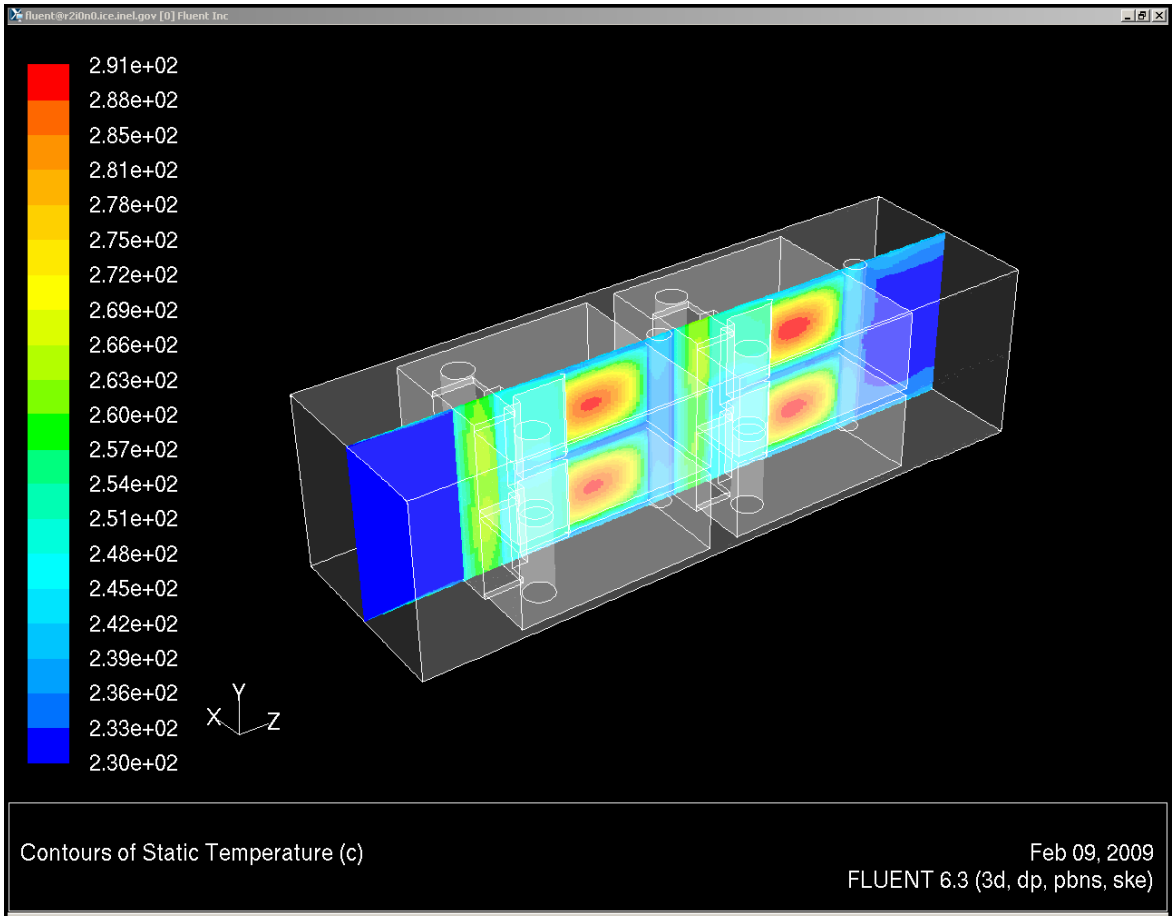
There Will be Gamma Heating. Minimize Impact by Machining Deeper Side Grooves and Limiting the Final a/W of the Tests.



Rectangular 0.4T-CT

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Testing and Specimens

Experiment Designation Capsule ID	X-750 0.4T-CT	XM-19 0.4T-CT	X-750 Tensile	XM-19 Tensile	X-750 TEM	XM-19 TEM	X-750 Optical/ SEM	XM-19 Optical/ SEM
EPRI-1	6	6	2	2	4	4	2	2
EPRI-2	6	6	2	2	4	4	2	2
EPRI-3	12	12	2	2	4	4	2	2
Total	24	24	6	6	12	12	6	6

- Fracture Toughness Tests in Air (at Temperature)
- Crack Growth Tests in NWC and HWC
- Additional CT specimens in the highest fluence capsule for possible investigation of environmental fracture toughness in HWC

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Summary

- EPRI engaged in developing irradiated fracture toughness and crack growth rate data for Alloy X-750 and XM-19
- Information will be used to construct disposition curves and to validate long-term operation of BWR internals

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Backup Slides

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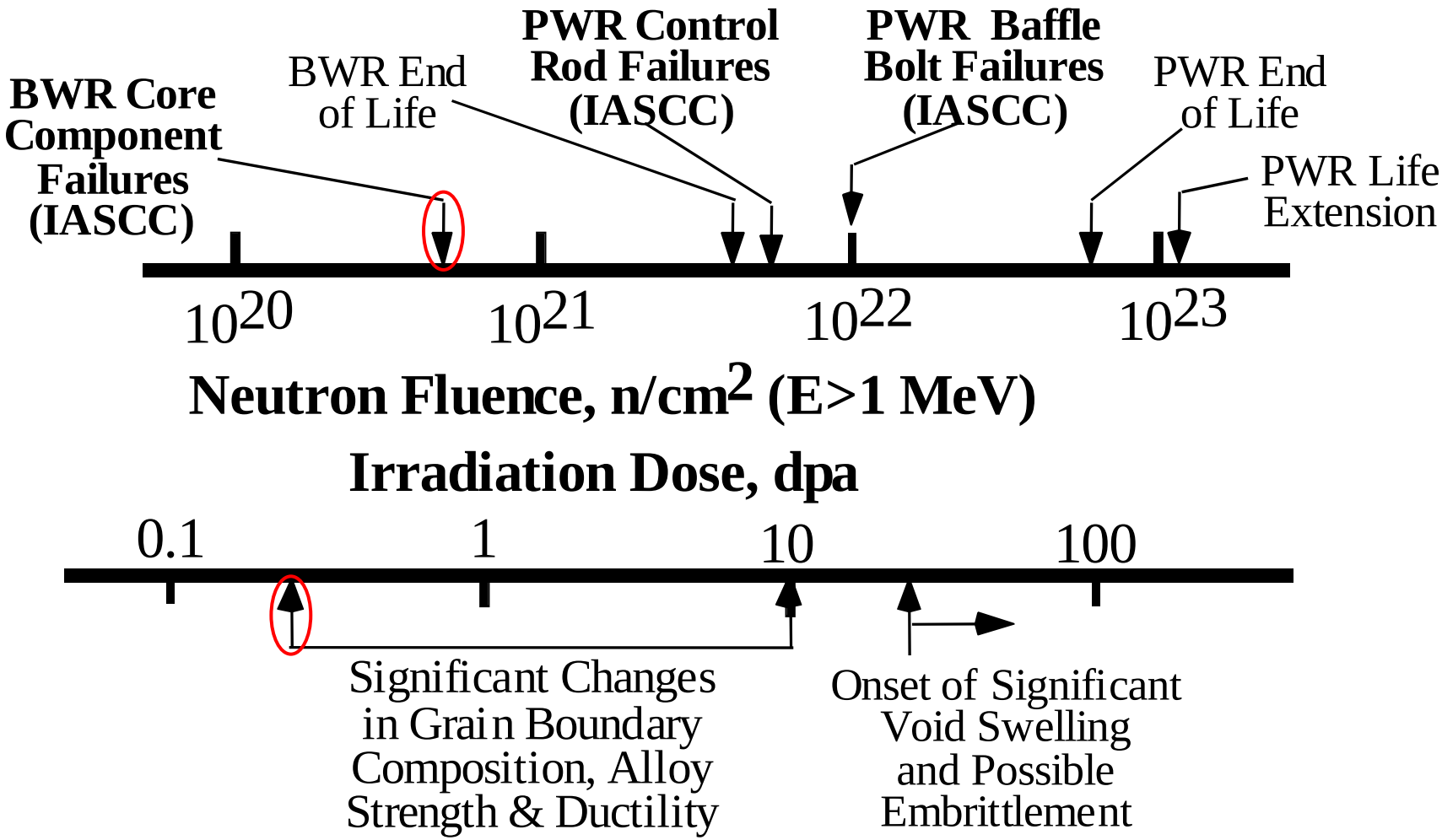
Reference: Onset of Neutron-Irradiation Effects in Stainless Steels



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