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Advanced Reactor Technologies Program

Fast Reactor Structural Materials

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Materials Science and Technology Division
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DOE-NE Materials Crosscut Coordination Meeting

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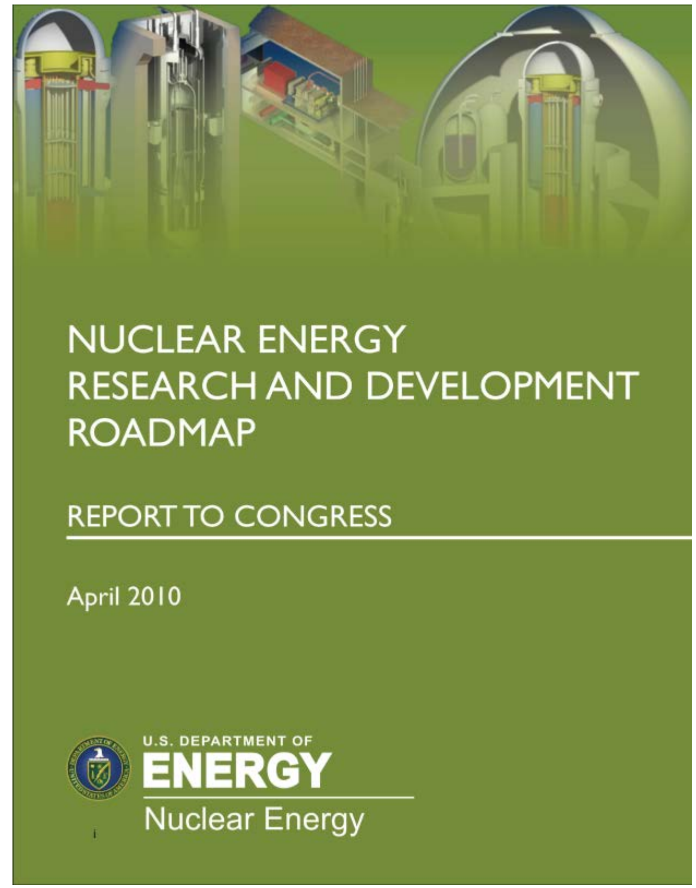
ART Program Supports Advanced Reactor Development

- **Advanced Reactor Technologies (ART) Program supports multiple high-level objectives identified in the 2010 Nuclear Energy R&D Roadmap (2 & 3)**

  (2) Develop improvements in the **affordability** of new reactors to enable nuclear energy to help meet the Administration's energy security and climate change goals

  (3) Develop **sustainable nuclear fuel cycles**

  …overall goal is to have demonstrated the technologies necessary to allow commercial deployment of solution(s) for the sustainable management of used nuclear fuel that is safe, economic, and secure and widely acceptable to American society by 2050."



Advanced Reactor Technologies Program

Program Mission:
To research and develop advanced technologies to significantly improve the efficiency, safety, and performance of advanced reactor systems

Advanced Reactor Technologies (ART)

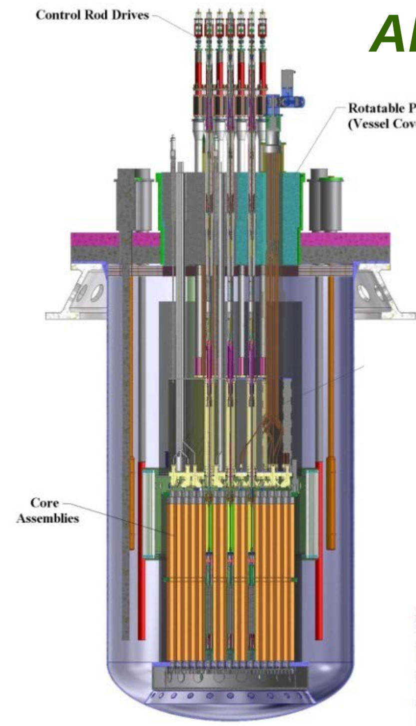
- **Fast Reactor Technologies**
- **Thermal Reactor Technologies**
- **Advanced Reactor Generic Technologies**
  - **Advanced Materials**
  - **Energy Conversion**
  - **ICHMI**
- **Advanced Reactor Licensing**
- **Advanced Reactor System Studies**

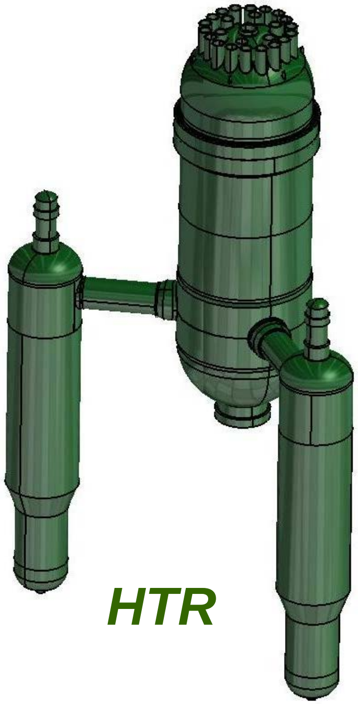
Structural Materials Are Critical for Advanced Nuclear Reactor Technologies

- **Development and qualification of advanced structural materials are critical to the design and deployment of the advanced nuclear reactor systems that DOE is developing**
  - **High and Very High Temperature Gas Cooled Reactors (HTGRs and VHTRs)**
  - **Sodium Cooled Fast Reactors (SFRs)**
  - **Fluoride Salt Cooled High Temperature Reactors (FHRs)**
- **Structural materials must perform over design lifetimes for pressure boundaries, reactor internals, heat transfer components, etc.**
- **Performance of metallic alloys and graphite for the long times and high operating temperatures required is being examined under the Advanced Reactor Technologies (ART) Program**

ART Program Includes Advanced Materials R&D Activities

- Development and qualification of graphite, improved high-temperature alloys, and ceramic composites for advanced reactor systems
- Advanced Fast Reactor-100 is an example of fast reactor systems
  - Targets local small grids with limited needs for on-site refueling
  - 250MWt/100MWe, sodium-cooled, core life (30 years), plant life (60 years)
- AREVA's High Temperature Reactor is an example of a He-cooled system
  - TRISO fueled, graphite moderated
  - 625MWt/315MWe, 750°C outlet temperature to steam generator, plant life (60 years)





Advanced Materials Program Structure

Advanced Materials

- Technical Area Lead: Sam Sham, ORNL

High Temperature Materials

- Technical Lead: Richard Wright, INL

Graphite

- Technical Lead: Will Windes, INL

Fast Reactor Structural

- Technical Lead: Sam Sham, ORNL

Active NEUP Projects (16) in High Temperature Structural Materials

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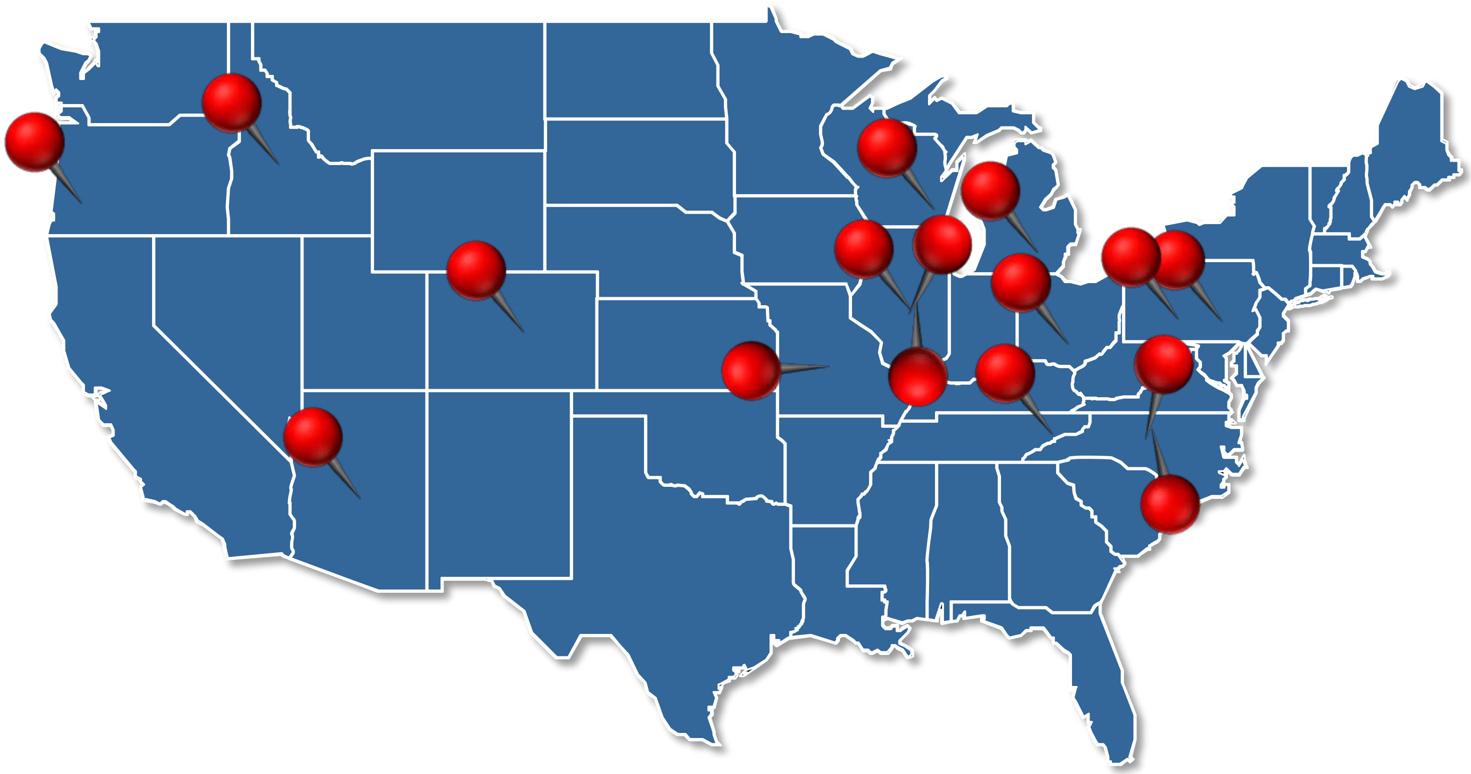
William Corwin, DOE-NE, ART Materials Technology Lead
Project 12-3541, Accelerated irradiations for high dose microstructures in fast reactor alloys (University of Michigan)
Project 12-3882, Neutron irradiation damage in pure iron and Fe-Cr model alloys (University of Illinois, Urbana-Champaign)
Project 13-4791, Mechanistic models of creep-fatigue crack growth interactions for advanced high temperature reactor components (Oregon State University)
Project 13-4900, Corrosion of structural materials for advanced supercritical carbon-dioxide Brayton cycle (University of Wisconsin-Madison)
Project 13-4948, Fundamental understanding of creep-fatigue interactions in 9Cr-1MoV steel welds (Ohio State University)
Project 13-5039, Multi-resolution testing for creep-fatigue damage analysis of Alloy 617 (Arizona State University)
Project 13-5252, Long-term prediction of emissivity of structural material for high temperature reactor systems (University of Missouri)
Project 14-6346, Integrated computational and experimental study of radiation damage effects in Grade 92 Steel and Alloy 709 (University of Tennessee-Knoxville)
Project 14-6562, Development of novel functionally graded transition joints for improving the creep strength of dissimilar metal welds in nuclear applications (Lehigh University)
Project 14-6762, Microstructural evolution of advanced ferritic/martensitic alloys under ion irradiation (University of Illinois, Urbana-Champaign)
Project 14-6803, Dissimilar joints between 800 H alloy and 2.25 Cr & 1 Mo steel (Pennsylvania State University)
Project 15-8308, Creep and creep-fatigue crack growth mechanisms in Alloy 709 (North Carolina State University)
Project 15-8432, Multi-scale experimental study of creep-fatigue failure initiation in a 709 Stainless Steel alloy using high resolution digital image (University of Illinois, Urbana Champaign)
Project 15-8548, Assessment of Aging Degradation Mechanisms of Alloy 709 for Sodium Fast Reactors (Colorado School of Mines)
Project 15-8582, Mechanistic and Validated Creep/Fatigue Predictions for Alloy 709 from Accelerated Experiments and Simulations (North Carolina State University)
Project 15-8623, Characterization of Creep-Fatigue Crack Growth in Alloy 709 and Prediction of Service Lives in Nuclear Reactor Components (University of Idaho)

Active NEUP Projects in High Temperature Structural Materials – Geographical



Fast Reactor Structural Activities

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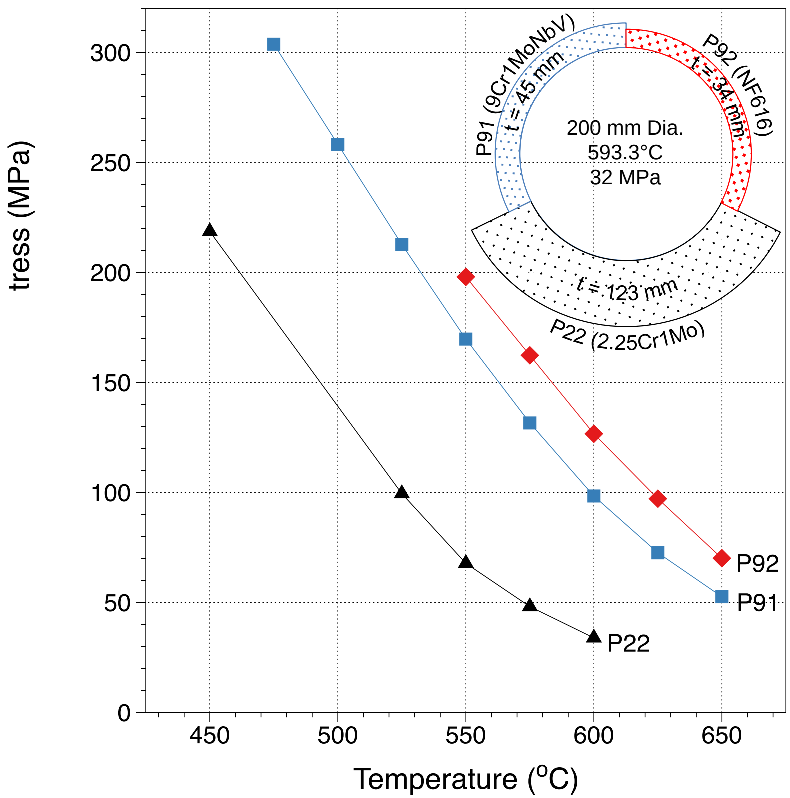
Acknowledgments

- Lizhen Tan, Yuki Yamamoto, Mikhail Sokolov, Randy Nanstad, Phil Maziasz and supporting staff (ORNL)
- Meimei Li, Ken Natesan and supporting staff (ANL)
- Laura Carroll and supporting staff (INL)

Advanced Structural Materials Provide Greater Safety Margin and Design Flexibility

- **Higher strength for constant temperature:**
  - Reduced commodities
  - Greater safety margins
  - Longer lifetimes
- **Higher temperature for constant stress:**
  - Higher plant performance (e.g., thermal efficiency)
  - Reduced commodities
  - Greater safety margins in accident scenarios
- **Combinations of above:**
  - Greater design flexibility



Fast Reactor Materials Development and Code Qualification

Enhanced structural performance of AFR construction materials would reduce capital costs, enable more flexible designs, and increase safety margins

FY 08

- Established advanced materials development strategy

FY 09-12

- Down-selected one austenitic steel and one F/M steel

- Alloy 709 (Fe-20Cr-25Ni base) austenitic steel
- Grade 92 steel with optimized chemistry and thermo-mechanical treatment

FY 13-15

- Intermediate term testing to support ASME Code Qualification assessment

- Alloy 709 – testing completed, recommended for ASME Code Qual.
- Opt Grade 92 steel - still waiting for some longer term data before assessment can be made

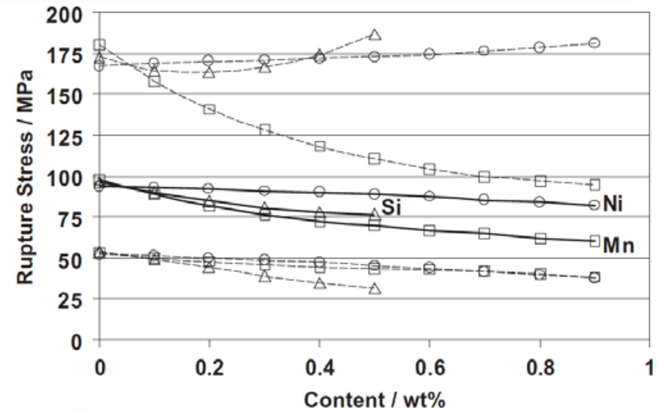
FY 16 & Beyond

- Opt Grade 92 – to complete assessment
- Alloy 709 – to initiate Code Qualification effort, and to integrate NEUP project activities

Optimized Grade 92 Alloy Chemistry

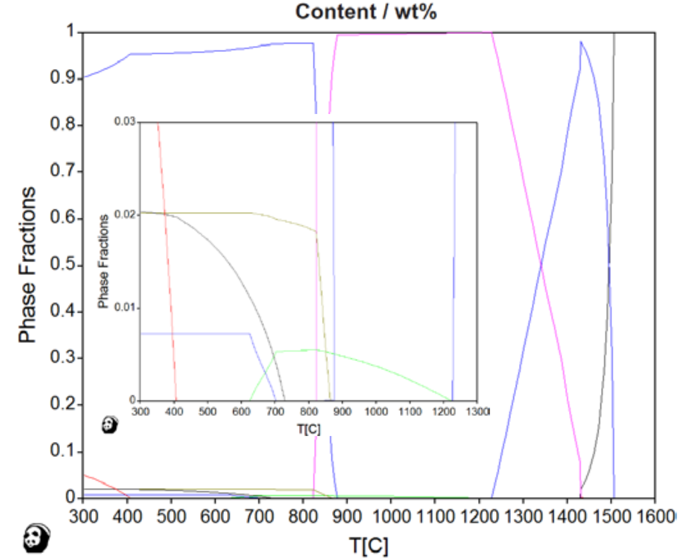
ASTM A1017 – 11 (Grade 92); A182 – 12a (F92); A335 – 11 (P92); A213 – 11 (T92)													
	C	Mn	P	S	Si	Cr	Mo	Ni	V	Nb	B	N	W	Others
Min.	0.07	0.3				8.5	0.3		0.15	0.04	0.001	0.03	1.5	
Max.	0.13	0.6	0.02	0.01	0.5	9.5	0.6	0.4	0.25	0.09	0.006	0.07	2.0	0.02Al, 0.01Ti/Zr

- **Alloy chemistry of optimized Grade 92 is adjusted to**
  - Reduce Ni, Si, and Mn contents, which tend to impair creep strength
  - Reduce $Cr_{23}C_6$-type precipitates and increase MX-type precipitates for better high temperature performance



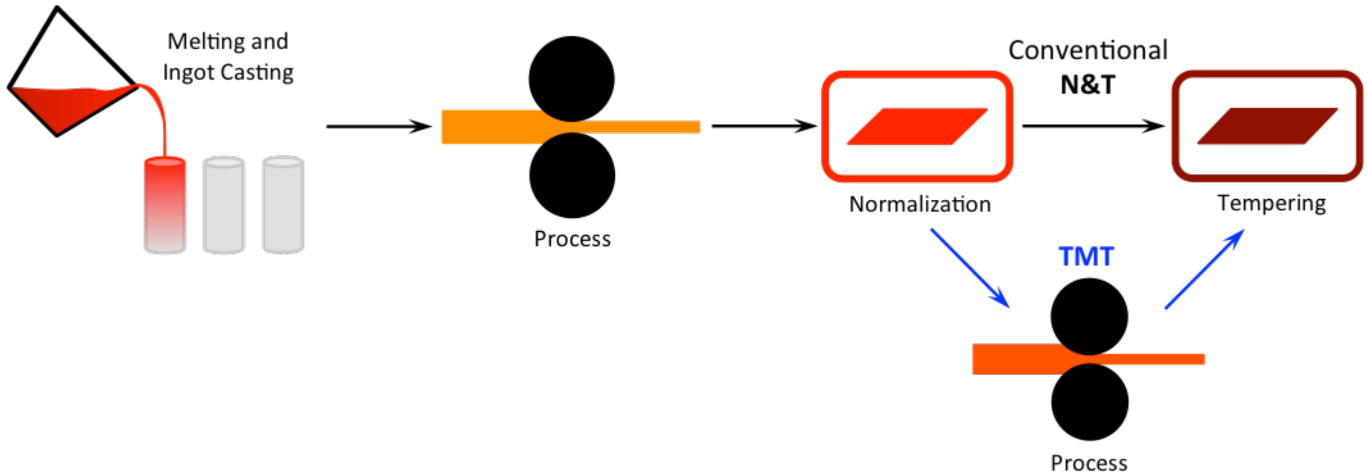
[Carlan et al. JNM (2004)]

- **Computational alloy thermodynamics is used to “visualize” the effect of alloy chemistry changes on phase constituents, which provide key information to alloy microstructure and subsequent thermomechanical treatment process.**

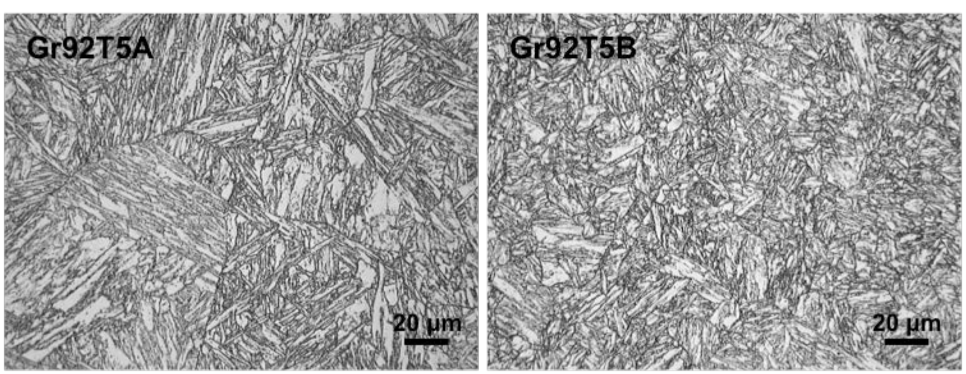


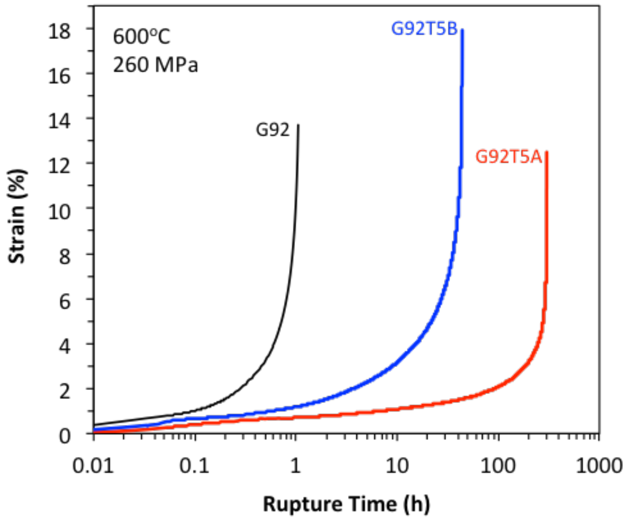
Optimized Grade 92 Thermomechanical Treatment (TMT)

- TMT can be easily implemented during conventional Grade 92 production.



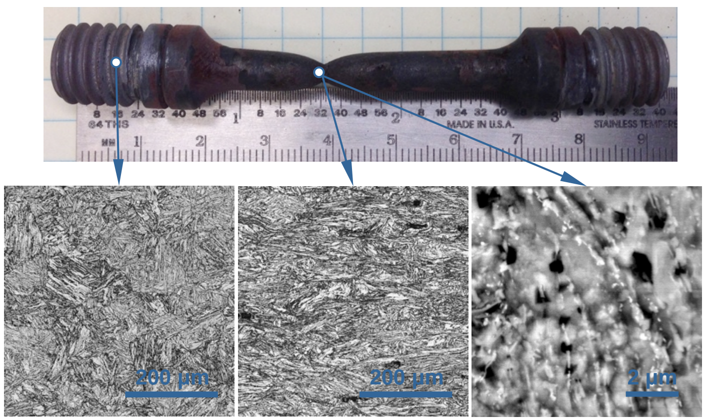
- TMT, significantly introducing additional nucleation sites for MX precipitates and possible refining grain size, would noticeably increase material's performance.



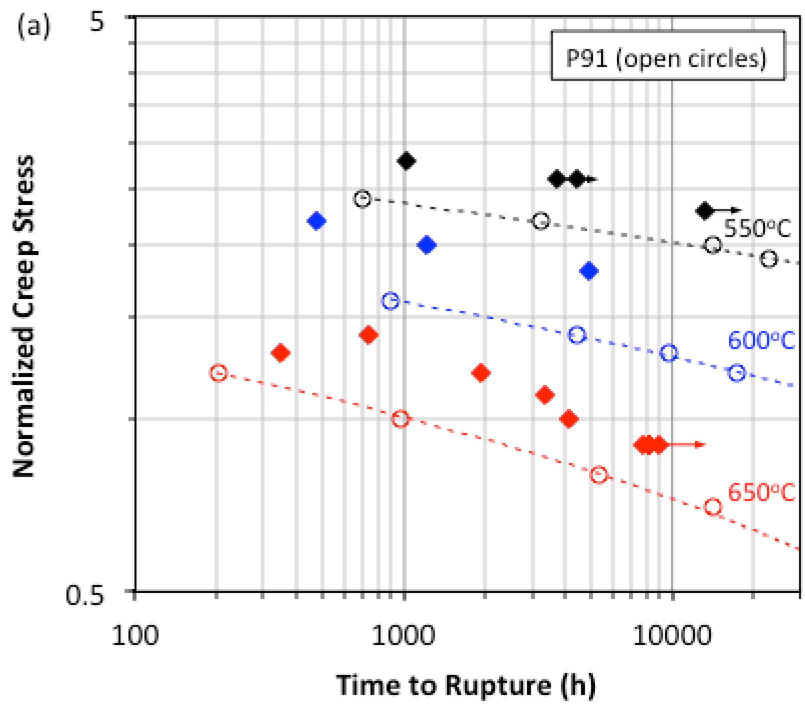


Creep Resistance

- **Creep tests have being conducted at 550, 600 and 650C and various loads. The longest test has achieved > 12,500 h at 550C.**
- **The test results indicate noticeable increases in creep strength as compared to P92 and P91.**

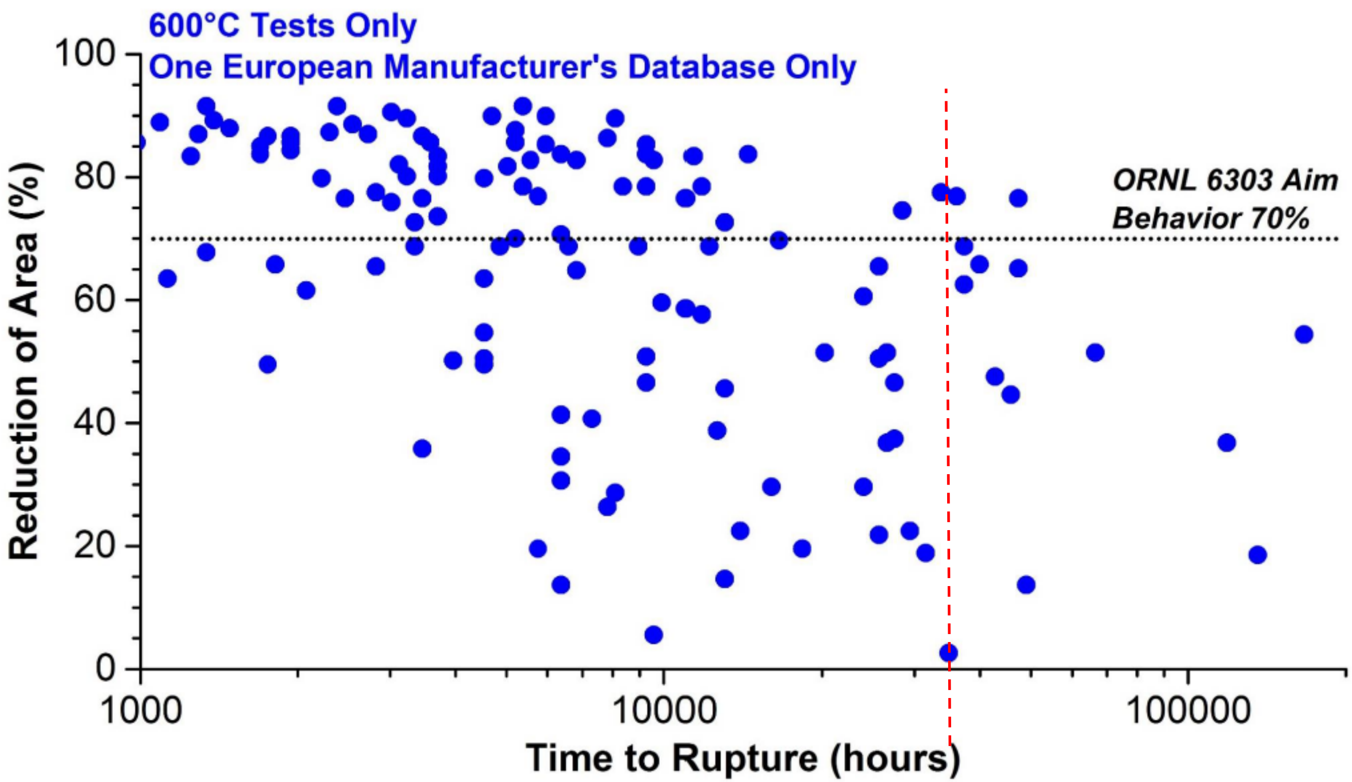


Creep cavities (lots <~2 µm and a few up to ~10 µm) formed close to the rupture site of a Opt Grade 92 specimen tested at 600C.



Creep Rupture Ductility Issue

Grade 91 Data **Plot prepared by EPRI**

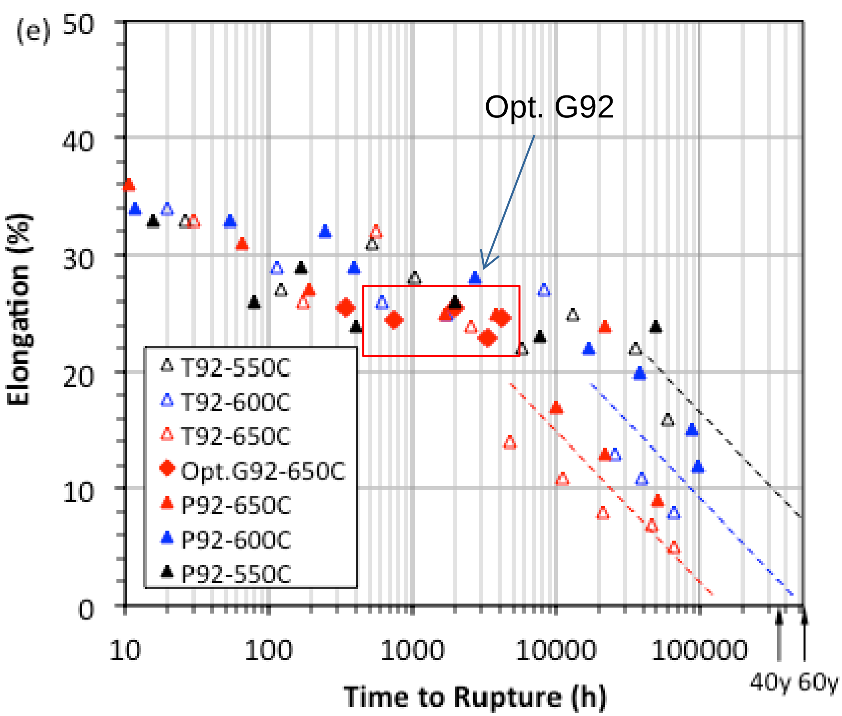


- Grade 91 base metal has been shown to exhibit wide variability of creep rupture ductility for the same life (can vary from 1 to 75%)
- Tightening of the chemistry spec and impurities of Grade 91 is being considered by ASME to mitigate the issue

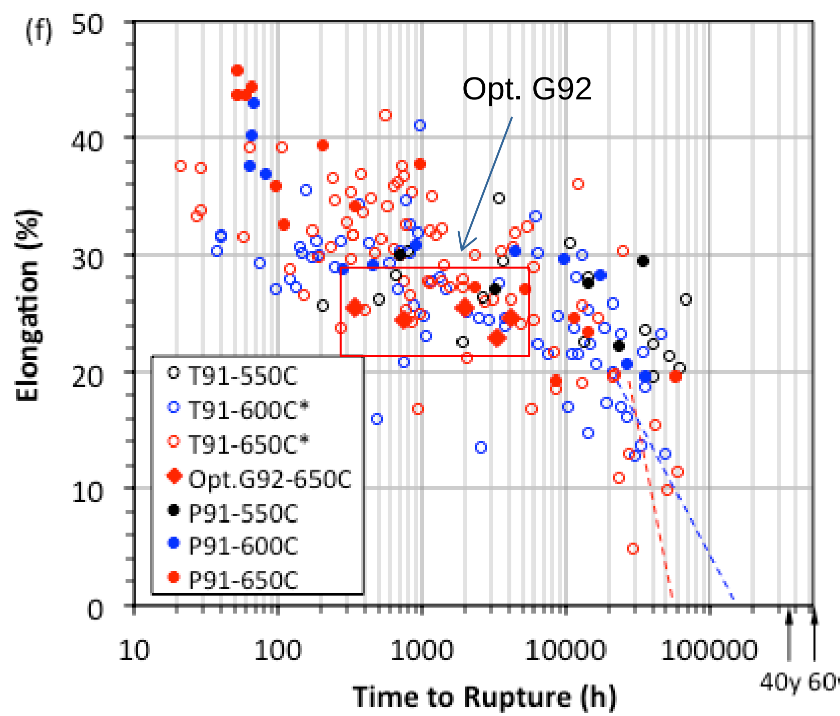
Creep Rupture Ductility for Opt Grade 92

- Higher temperature and tube products (i.e., T92 and T91) tend to result in lower elongation and reduction of area (RoA).

Grade 92 Data



Grade 91 Data

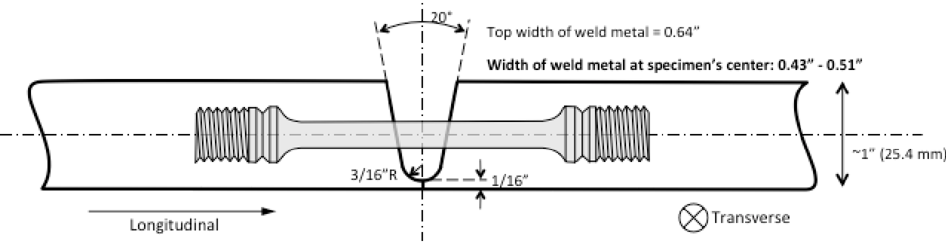


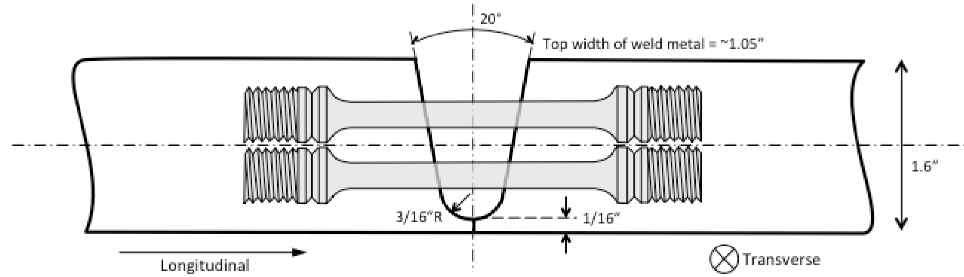
[Reference data of P/T91 and P/T92: NIMS Creep Data Sheet and ASME STP-NU-019-1 (by R.W. Swindeman *et al.* 2009)]

* The inclusion of 597C and 649C data in both tube form and plate form

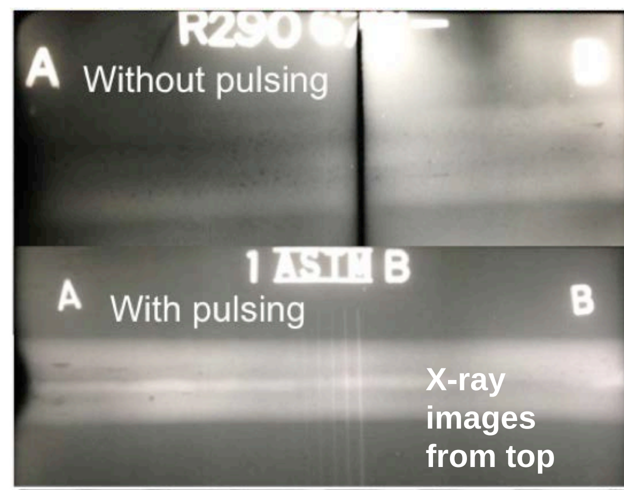
Welds Fabrication

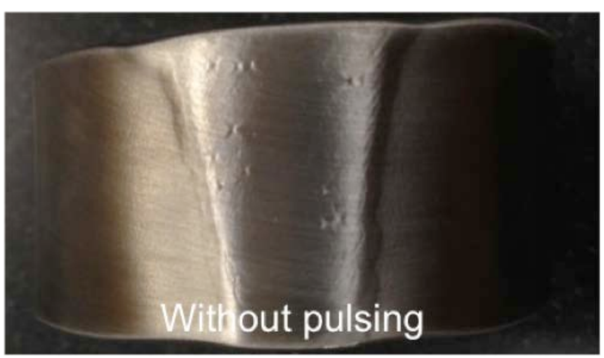
- **Single compound bevel U-shape groove was designed for gas tungsten arc welding (GTAW) of the 1”- and 1.6”-thick plates to accommodate standard size creep/tensile and other testing specimens.**

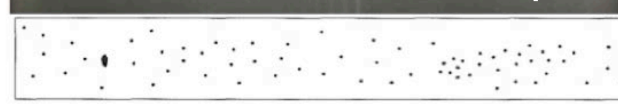




- **Welds have fabricated from heats 011365 and 011449 at ORNL and a subcontract (SWM) following ASME Section IX Welding Qualifications, e.g., QW/QB-422, A/SA-182 F92 (K92460) with welding P-No. 15E.**
  - Current pulsing was employed for the welds because non-pulsing resulted in much more cavities in the weld metal although the cavities did not result in noticeable cracks.





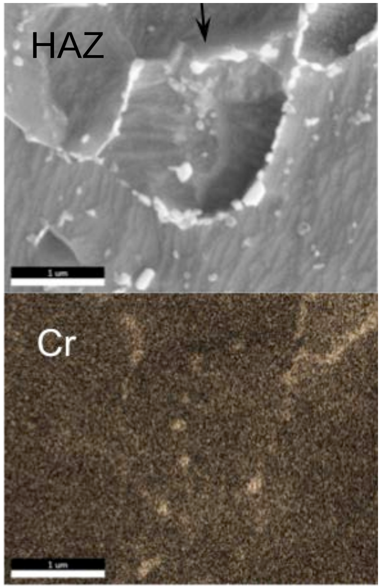


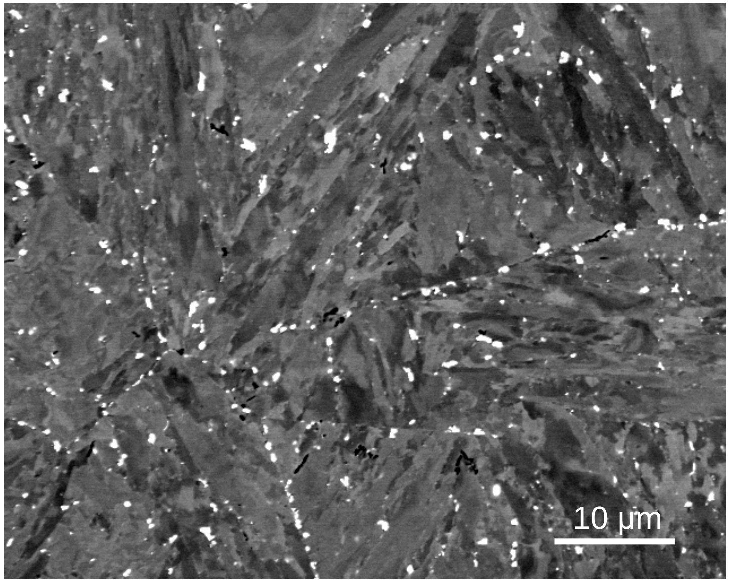
Typical Quantity and Size Permitted in 6 in. (150 mm) Length of Weld Over 1 in. (25 mm) Thickness

ASME Section IX QW-191.1.2.2(b)(4)

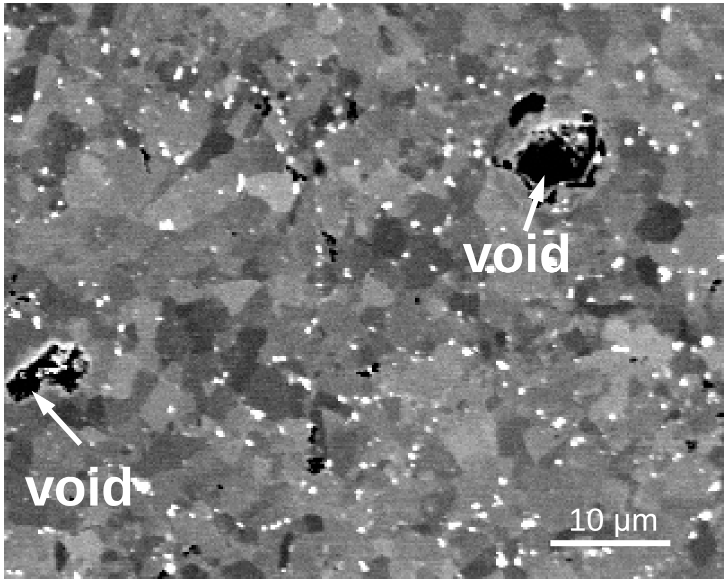
Creep Resistance of Cross-Welds (Type IV Cracking)

- **The optimized Grade 92 having refined microstructure with a high density of MX precipitates helps reducing re-precipitation of $Cr_{23}C_6$ at boundaries in the HAZ of welds, which delays Type IV cracking with less reduction in creep life.**
  - Creep voids (black) formed in the HAZ after test at 650C but not associated with dispersive Laves phase (white particles).





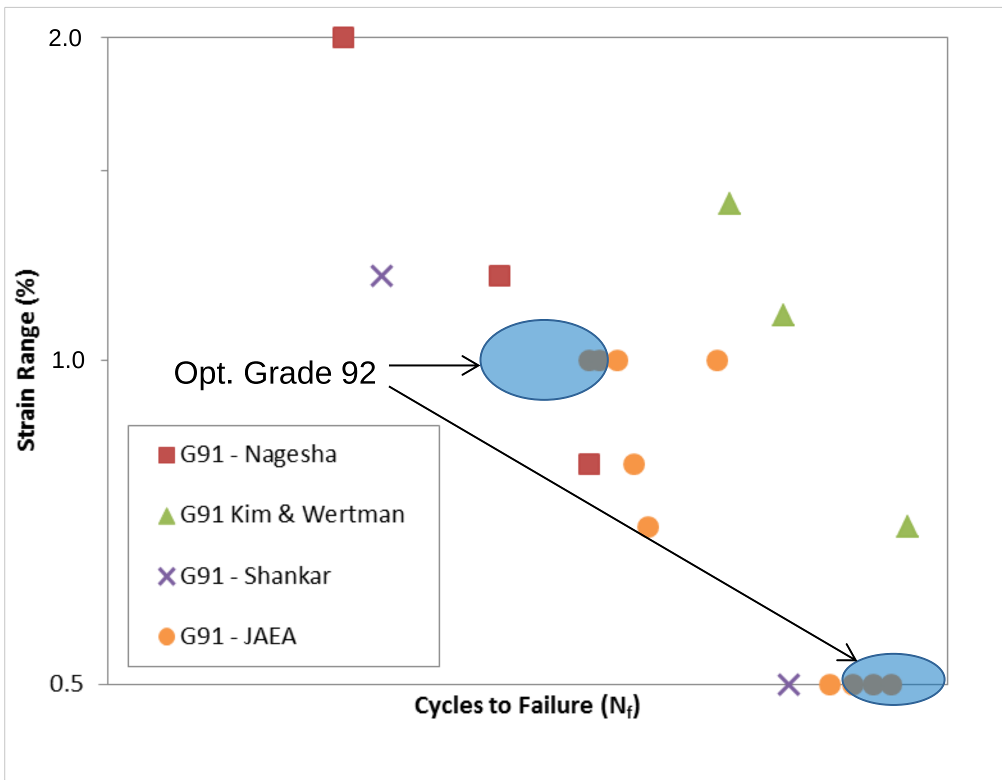
BM after creep test at 650C



HAZ after creep test at 650C

Fatigue – Cycles to Failure v. Strain Range

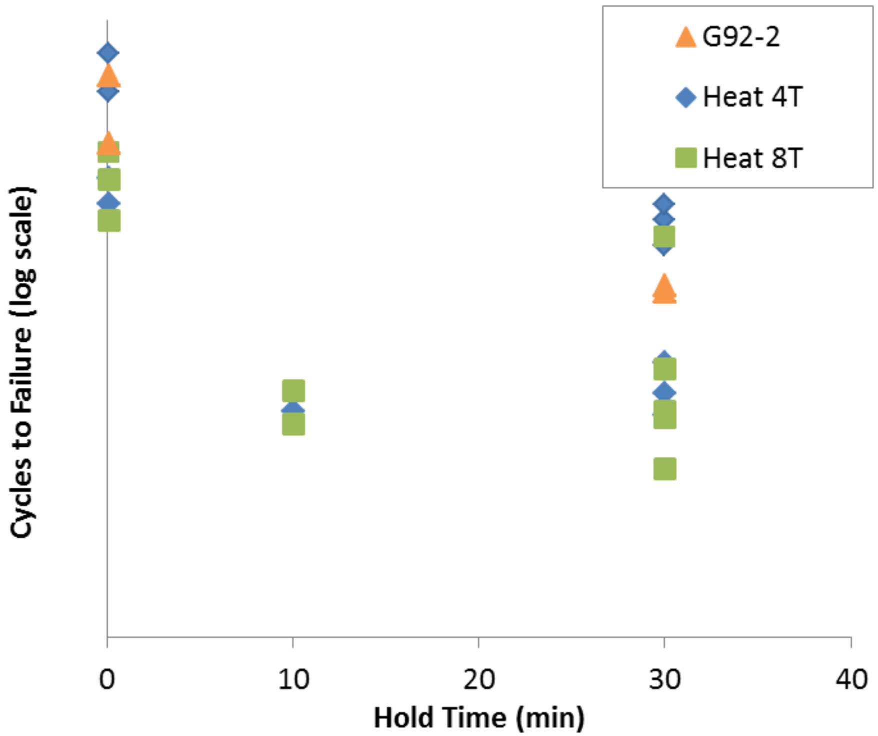
- **Opt. Grade 92 fatigue cycles to failure within the scatter of available Grade 91 literature data**



Note: All comparison data are estimates from the literature
Note: JAEA reference is ASME STP-NU-018-2009

600C Creep-Fatigue: Cycles to Failure v. Hold Time

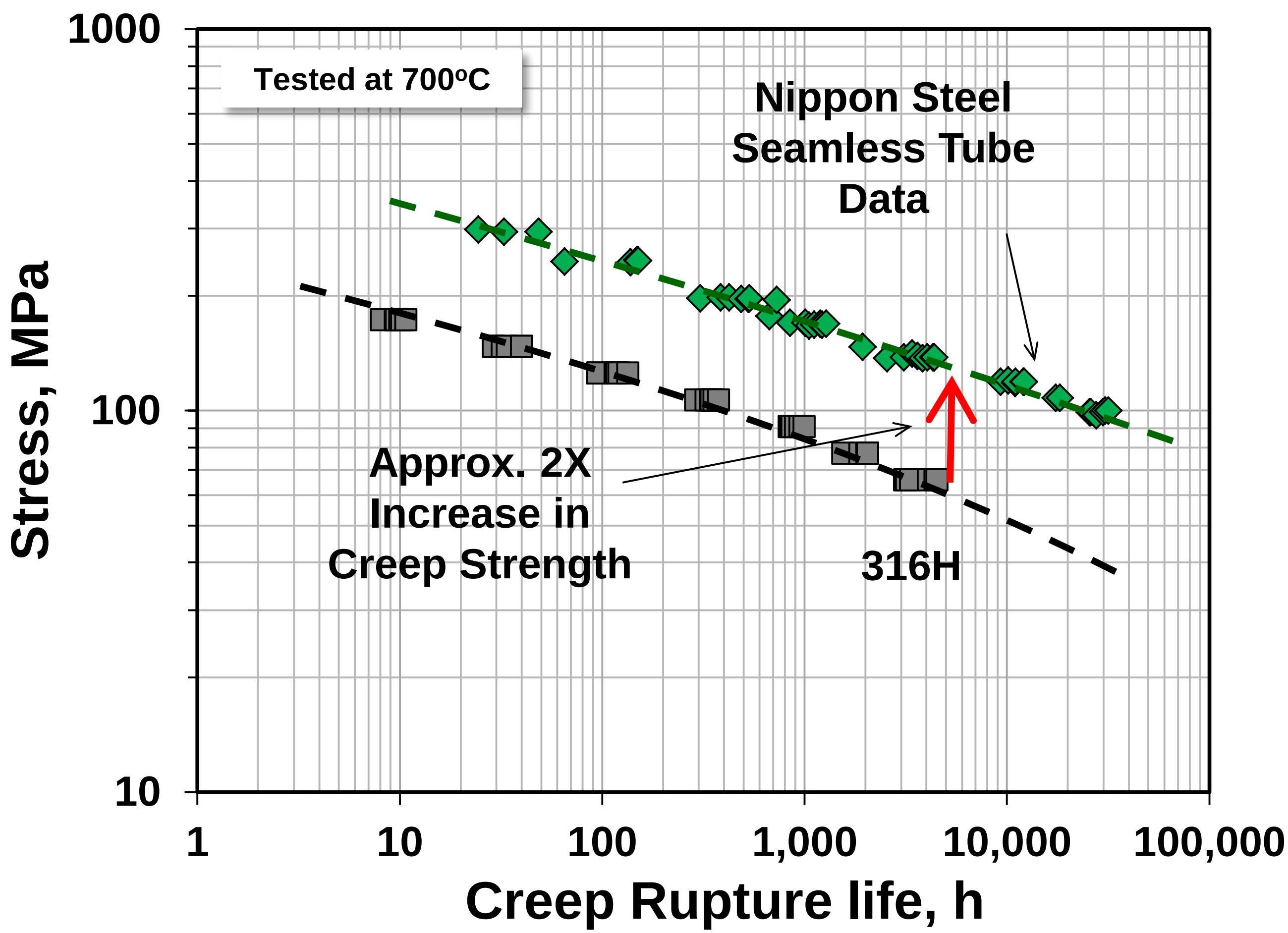
- Lower strain range CF testing and shorter hold time testing ongoing for Heat 8T
- CF cycles to failure degraded relative to continuous cycle fatigue



Thermal Aging and Sodium Compatibility of Opt Grade 92

- Effects of long-term thermal aging on tensile properties of Opt Grade 92 were evaluated.
- The microstructural evolution is evaluated by using ThermoCalc and DICTRA to predict the microstructures over long times.
- Accelerated aging experiments are conducted to validate the calculated morphologies which will enable assessment of long-term performance.
- Thermal aging at 650C resulted in a decrease in the tensile strength
- Sodium exposure at 650C has a much stronger effect on tensile strength than thermal exposure alone at 650C.
- Precipitation of Laves phase was observed in all sodium-exposed specimens of and could be the cause of strength reduction.
- Longer term aging and sodium exposures at 600 and 550C are continuing.
- Microstructural evaluations of the aged/exposed specimens will be made to assess the long-term performance of these materials in SFRs.

Alloy 709 Has Enhanced High Temperature Strengths vs Reference Material (316H)



Capital Cost Reduction and Design Advantages

- **Alloy 709 structural applications identified for AFR-100**
  - Core support structures, reactor vessel, primary and secondary piping
- **Alloy 709's higher strengths provide capital cost reduction and design advantages over reference 316H stainless steel throughout a broad temperature range**
  - Thinner walls
    - Lower material quantities and reduced through-wall thermal gradients
  - Higher allowable thermal gradients
    - Potential for more compact component and plant configurations
  - Smaller piping expansion loops
  - Could open up opportunities for other design innovations

Widening of Design Envelope

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- Alloy 709 allows greater thermal gradients as compared with 316H
- Results in prospect of eliminating costly add-on hardware instituted in past designs to mitigate deficiency of 316H*
  – French Phenix, Super Phenix; German SNR; U.S. FFTF, CRBR; Japanese MONJU

* Dhalla, A.K., (1991), "Recommended Practices in Elevated Temperature Design: A Compendium of Breeder Reactor Experiences (1970-1987) - Volume II Preliminary Design and Simplified Methods," WRC Bulletin 363

Table 1. Examples of past structural design problems for sodium fast reactors

Item	Component	Issue and Resolution
1	French Super Phenix main vessel, bottom head	The initial shallow bottom head did not have adequate buckling capacity for faulted conditions. The revised head design was less shallow and, in addition, the thickness of the shell in the knuckle region was increased.
2	Upper part of Super Phenix main vessel	A fluctuating sodium pool temperature and elevation, combined with dead weight loading, resulted in a potential ratcheting condition. An internal baffle structure was introduced to thermally shield the main vessel wall.
3	Super Phenix main vessel spillway	In order to achieve insignificant creep, a spillway system was used to direct flow into the cold plenum. There was a vibration problem due to hydro-elastic coupling between the spillway head and sodium feed that was resolved by a revised spillway design.
4	Bolted flange connection of the German SNR reactor vessel internals and circuit components	Differential flange temperatures cause radial motions that lead to fatigue straining of the bolts. A prestressed expansion bolt deign is provided when the predicted displacement exceeds experimentally determined limits.
5	U.S. CRBRP reactor vessel	Increases in the specified seismic loading after some large diameter shells were partially fabricated led to extensive reanalysis and increased shell thickness in topmost shell of the reactor vessel.
6	CRBRP reactor vessel outlet nozzle	Because of the temperature difference between the vessel wall and the thermal liner, excessive leakage from the bypass flow annulus could result in "thermal striping" and fatigue failure of some nozzle parts. After over twelve nozzle designs were investigated, a double diaphragm seal design using two flexible disks was developed to resolve the problem.
7	CRBRP reactor vessel makeup nozzle bridge liner	The original design had a 316H extension into the outlet plenum that was subject to "thermal striping" induced fatigue failure. The problem was resolved by adding a pin mounted Inconel 718 sleeve to protect the 316SS bridge liner.
8*	Japanese MONJU upper internal structure - above core structure portion	Thermal striping problem resolved with incorporation of Alloy 718 liner
9	MONJU upper internal structure - cylindrical structure	The thickness required for seismic loading resulted in excessive thermal stresses during start-up and shut-down. Problem resolved by inserting a sodium baffle structure to mitigate the thermal stresses.
10	MONJU reactor vessel - upper part	Sodium level and temperature variations caused excessive thermal stresses in the vessel upper wall. Problem was resolved with addition of sodium thermal baffle, overflow lines and sodium level control protocols.
11	MONJU reactor vessel - inlet nozzle	Hot thermal shock combined with pressure and mechanical loadin required design optimization supported by inelastic analysis.

Item	Component	Issue and Resolution
12	Super Phenix single redan inner vessel	Design iterations were required to find solution to buckling problem.
13	Phenix IHX, secondary sodium outlet header	Observed sodium leakage at the inner shell to closing plate junction was resolved in Super Phenix with addition of a thermal mixing device, thermal insulation and toroidal expansion elements to accommodate large thermal expansion differences.
14	U.S. FFTF shear key forged ring	The calculated strain caused by thermal transient stress due to large differences in thickness exceeded the allowable limit at the weld. The ring was reconfigured to move the peak strain location away from the weld area.
15	CRBRP IHX primary sodium inlet nozzle	Large transient temperature differences between the perforated section of the tubesheet and the surround solid rim and shell resulted in excessive strain and creep-fatigue damage. The tubesheet rim was scalloped to reduce its mass and the knuckle radius was revised to move the weld out of the highest stress region.
16*	CRBRP primary sodium pump	Thermal shields, convection-cell barriers, minimized thickness variations and substituting higher strength 316SS for 304SS were used to ensure that operational clearances were maintained.
17	FFTF IHX primary closure seal	This seal is required to accommodate differential thermal motion between two massive components. A series of design optimization studies using inelastic analysis were required to finalize an acceptable configuration.
18	FFTF IHX shear key forged ring	The difference in thermal response between the relatively thin shell and much thicker forged ring resulted in excessive creep-fatigue damage and excessive strain even when evaluated with inelastic analysis. Since this was evaluated after the ring was fabricated, changes in the geometry were not feasible. Heaters were applied to the outside surface of the ring during heat-up to minimize the difference in thermal response. The satisfaction of strain and creep-fatigue limits was demonstrated with inelastic analysis.
19	CRBRP steam generator (superheater) tubesheet.	The original bolted head design to facilitate tube to tubesheet joint inspection and tube plugging was revised to a welded head design out of concern for the O-ring seal and creep-fatigue damage at the stud hole. The welded head design was optimized by contouring the rim and shell to minimize the difference in thermal response.
20	MONJU IHX upper tubesheet	Transient temperature differences between the thin conical shroud and the tubesheet caused excessive thermal stresses and ratcheting when analyzed. A half scale test was performed in a tubesheet thermal shock test facility which demonstrated non-ratcheting.
21*	CRBRP helical coil steam generator helical coil clamp assembly	A preload must be maintained on the clamp assembly to prevent slippage between the tube and clamp under all operating conditions. The original continuous clamping had too much interface load variation; it was resolved by changing to a segmented clamp.

Component	Issue and Resolution
SNR-300 inlet nozzle of the IHX ...glazing line to the heat ...sfer piping system	A single thermal sleeve did not provide adequate thermal protection from the differences in sodium temperature as the two fluid streams mixed. A double baffle design which separated the load carrying pipe support function from the thermal shielding function resolved the problem.
... of the SNR 300 main ...stem	Large wall thickness required for pressure and piping reaction loads causes high thermal stresses when joining sodium flows with different temperatures. Introduction of a thermal shield and stagnant sodium buffer heat capacity provided sufficient isolation from thermal transients.
...heads at ...lls	Excessive strain accumulation due to the temperature difference between the flued head cone and the pipe was resolved by removing the insulation to permit radiation heat transfer and reduce the steady-state stress intensity.

...e boundary related, but are of significance in terms of structural design.

Alloy 709 Code Qualification Plan

- Environmental effects (sodium, irradiation)

Elevated Temperature Structural Design (Construction)

Support NRC Licensing & Plant Operation

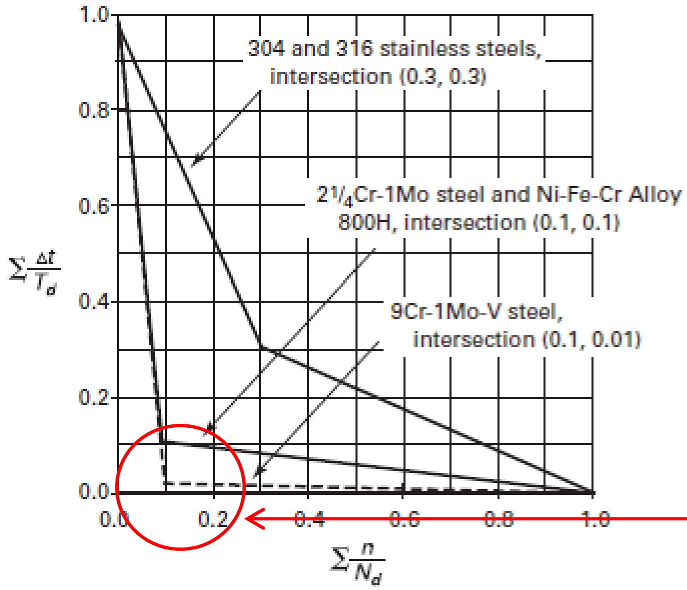
- ASME Section III Division 5, Subsection HBB (Class A) and HCB (Class B)

- Verification and validation of ASME code rules
- SFR structural issues
- High temperature flaw evaluations

Effort will be initiated in FY16 to implement part of the Code qual. plan

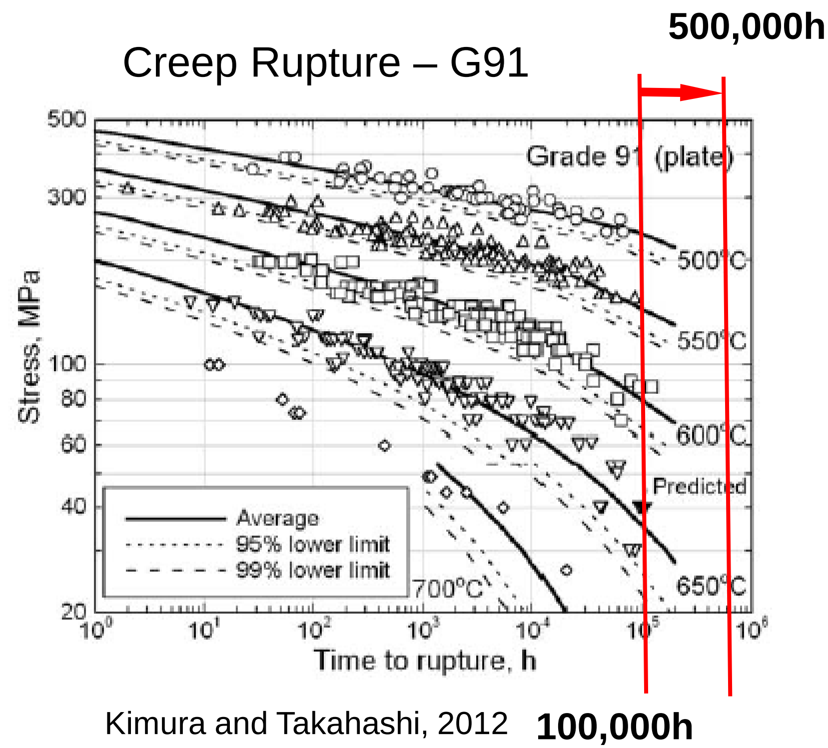
Other Ongoing Activities for Fast Reactor: Materials Design Technology

Creep-Fatigue Interaction



Removal of unnecessary conservatism in design methodology could lead to more flexibility in construction and operation of AFR

Creep Rupture – G91



Kimura and Takahashi, 2012

Gaining mechanistic understanding of long term degradation mechanisms such as creep and creep-fatigue damage and thermal aging could provide guidance on the extrapolation of accelerated time-at-temperature design data to 60-year design life, and beyond, with higher confidence

THANK YOU