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Fracture Toughness Vis-a-Vis the Master Curve for Some Advanced Reactor Pressure Vessel and Structural Steels

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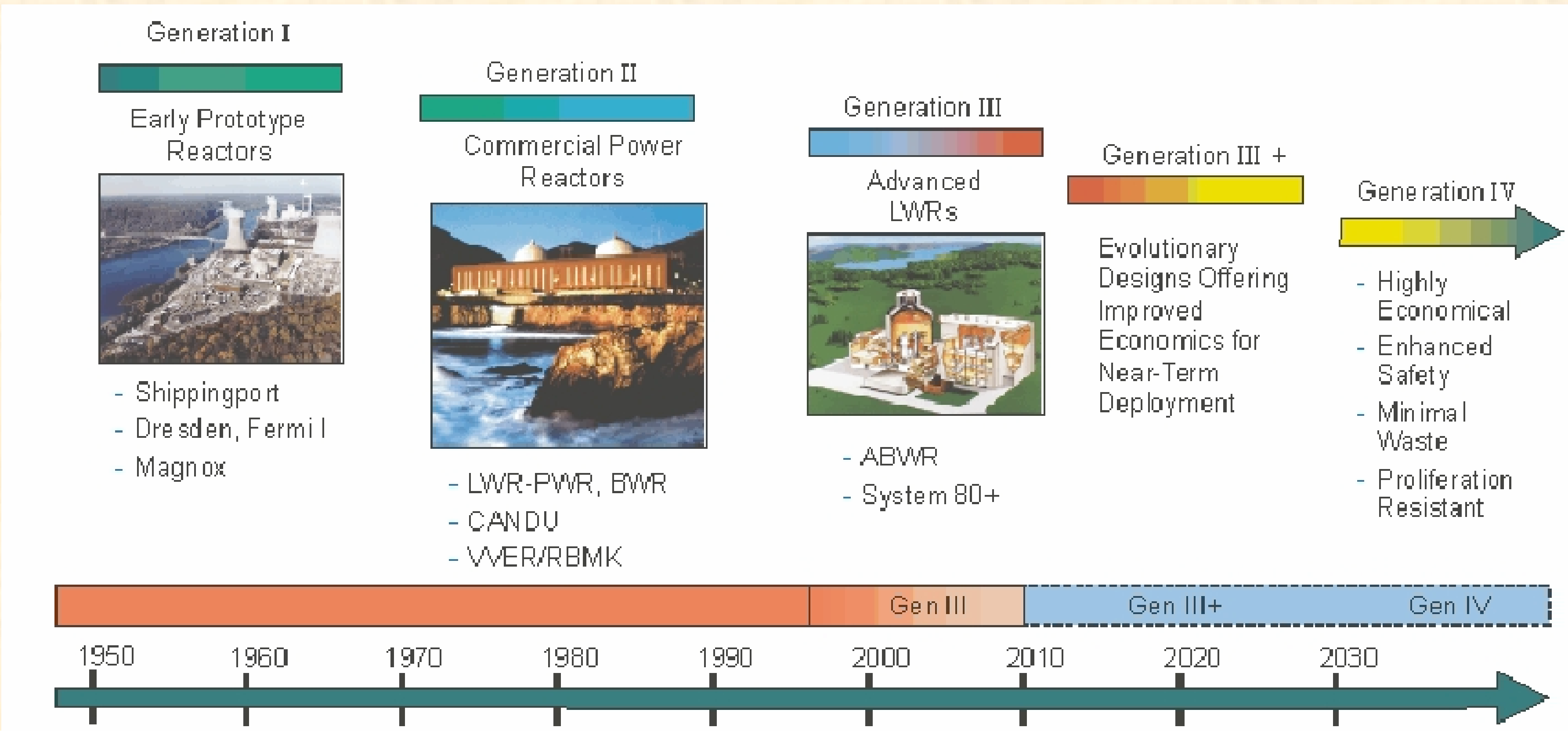
Generation IV Initiative

- **Generation IV Goal-**to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation and physical protection, and public perception concerns.



U.S. Materials Program Is Focused on Four Generation IV Reactor Systems

	Acronym	Coolant	Neutron
Gas-Cooled Fast Reactor (Medium, Long Term)	*GFR*	*Gas*	*Fast*
Lead-Cooled Reactor (Medium, Long Term)	*LFR*	*Liquid Metal*	*Fast*
Molten Salt Reactor (Low, Long Term)	*MSR*	*Molten Salt*	*Thermal*
Sodium-Cooled Reactor (Low, Long Term)	*SFR*	*Liquid Metal*	*Fast*
Supercritical Water-Cooled Reactor (Medium, Long Term)	*SCWR*	*Water*	*Thermal – (Fast)*
Very High Temperature Reactor (High, Mid Term)	*VHTR –NGNP–*	*Gas*	*Thermal*

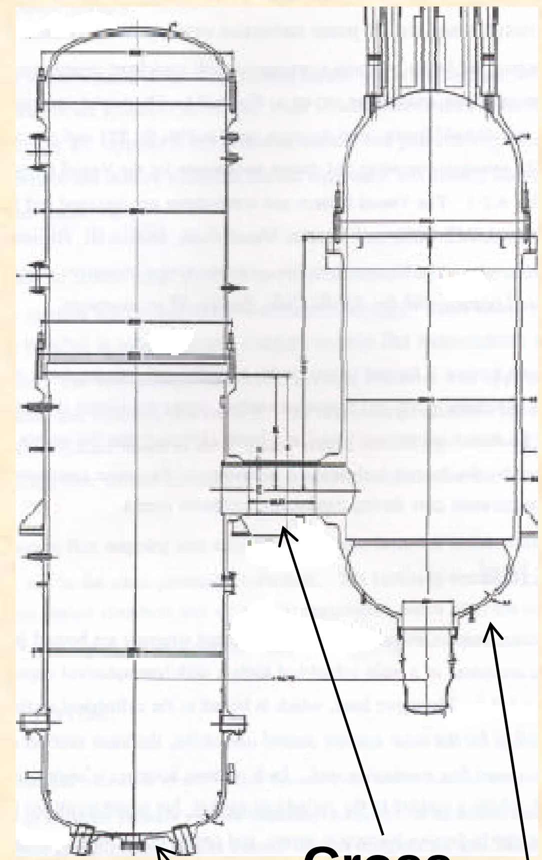
(Current U.S. Priorities)

9Cr-1MoV Is Primary Choice for the New Generation Nuclear Plant (NGNP/VHTR) Reactor Pressure Vessel System

- **Up to 490ºC (previous = 650°C) operating temp and 3 x 10^{19} n/cm^2 fluence (>0.1 MeV), 0.077dpa**
- **Issues include irradiation effects in creep range, and long-term strength**
- **High-temperature design methodology needs updating for nuclear service**
- **Very large vessel sizes will require scale-up of ring forging and joining technologies and ensuring thick-section properties**

7 to 9Cr-2WV, 3Cr-3WV, 2 1/4Cr-1Mo, and 12Cr-1MoWV also being evaluated

Correctly scaled size of typical PWR RPV



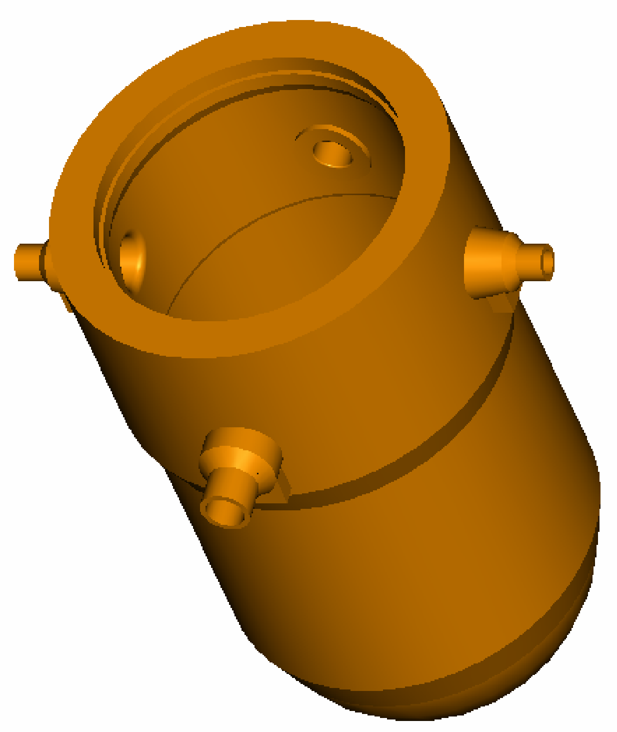
PCV

Cross vessel

RPV

Manufacturing Requirements for Super Critical Water Reactor (SCWR) Vessel Ring Forgings Stretch Infrastructure

- **Maintaining through-thickness mechanical and chemical properties during fabrication is primary challenge**
- **Inspectability for very heavy sections must be ensured**
- **Primary candidate material**
  - **A508 Grade 3 Class 1**
- **Alternate high-strength materials**
  - **A508 Grade 4N Class 1**
  - **3Cr-3WV**
- **Uninsulated hot nozzle may require 2 1/4 Cr-1Mo**



- **280°C wall temperature**
- **<5x10^{19} n/cm^2 (E>1 MeV)**
- **27.5 MPa nominal pressure**
- **Thickness *46 cm (18") in the beltline region, ~61 cm (24") in the nozzle region***

Design of hot nozzle thermal sleeve not completed

Japan Steel Works has Fabricated Very Large Forgings, Yet Not as Large and Thick as that Projected for the SCWR and Not of the Material Projected for the NGNP

Conceptual Design Operating Conditions for Gen IV RPVs Differ Widely

Reactor	Coolant	Operating Temp. (°C)	Off-Normal Temp/Time. (°C)	Operating Pressure (MPa)	Neutron Spectrum	Dose (dpa)
NGNP	Helium	400-500	520-610/50h	7.4-8.0	Thermal	0.08
SCWR	Supercritical Water	280	280	25	Thermal	0.05
GFR[a]	Helium	300-850	800-1100	7-21	Fast	<40
LFR[b]	Lead or Pb/Bi	Near:<550 Long:<800	??	Atmos-pheric	Fast	15-40

[a] Temperatures and pressures dependent on specific design: He direct, He/Supercritical CO_2, Supercritical CO_2

[b] Temperatures dependent on specific design: Near-term w/ outlet temp of 550°C and long-term w/outlet temp of 800°C. RPV temperature equal to inlet temperature.

Four Different Structural Materials of Varying Experience Will Be Discussed

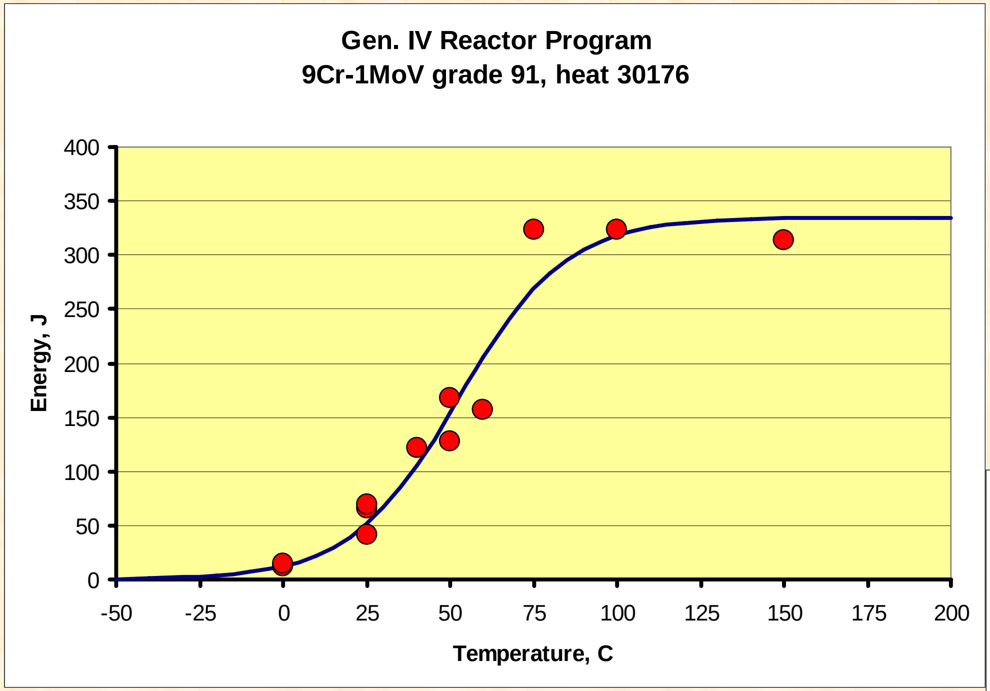
- **Modified 9Cr-1Mo alloy, 9Cr-1MoV (Grade 91) ferritic-martensitic alloy used for high temperature structures, including pressure vessels – industrially mature high temperature database and experience, approved in ASME Code to 649ºC.**

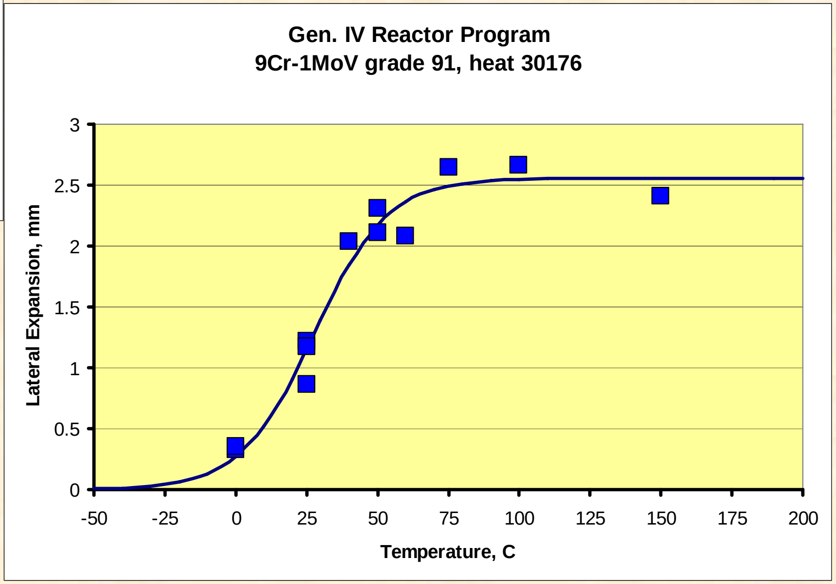
- **F82H, an 7.5Cr2WV ferritic-martensitic alloy. Smaller database than above, but good potential for higher strength. One of alloys developed to have reduced activation under neutron irradiation with resultant advantages for decommissioning**

- **12YWT, an oxide-dispersion-strengthened (ODS) alloy with very good high temperature strength, very new alloy, no database, no experience, under development.**

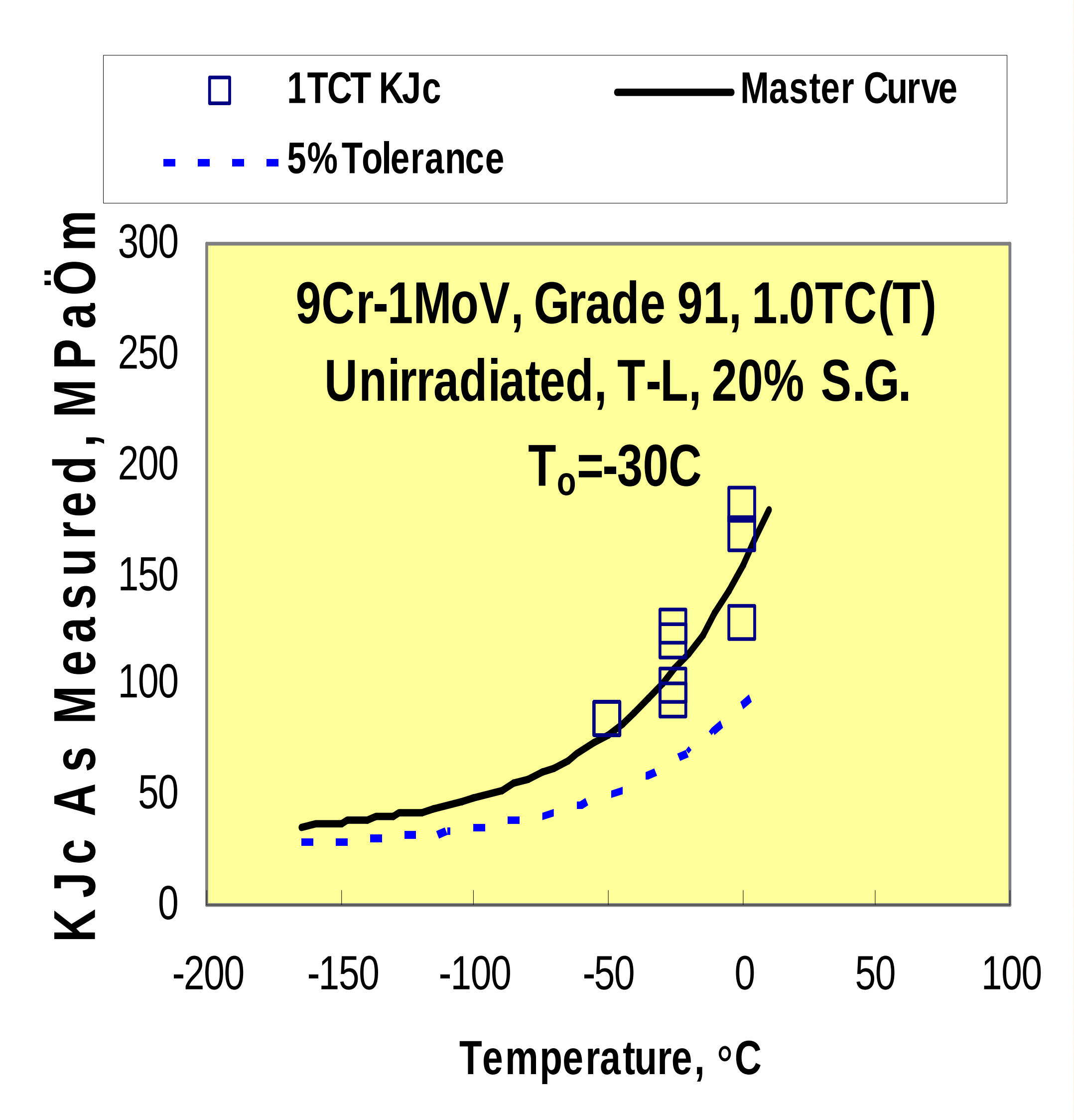
- **HSLA-100, a high strength (~710 MPa @ R.T.) low alloy structural steel with improved toughness relative to HY-100 steel, often used for naval structures.**

Charpy Impact T_{41J} for 9Cr-1MoV (Heat 30176) is 20ºC





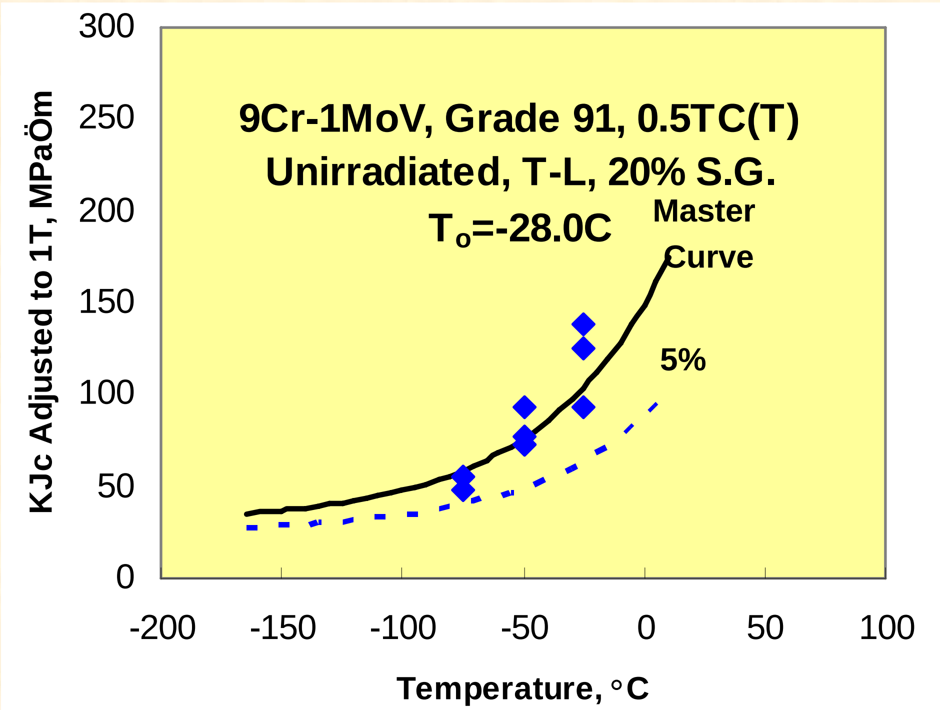
Ten 1T Compact Specimens of 9Cr-1MoV at Three Temperatures Gave Master Curve T_0=-30°C



Thus,

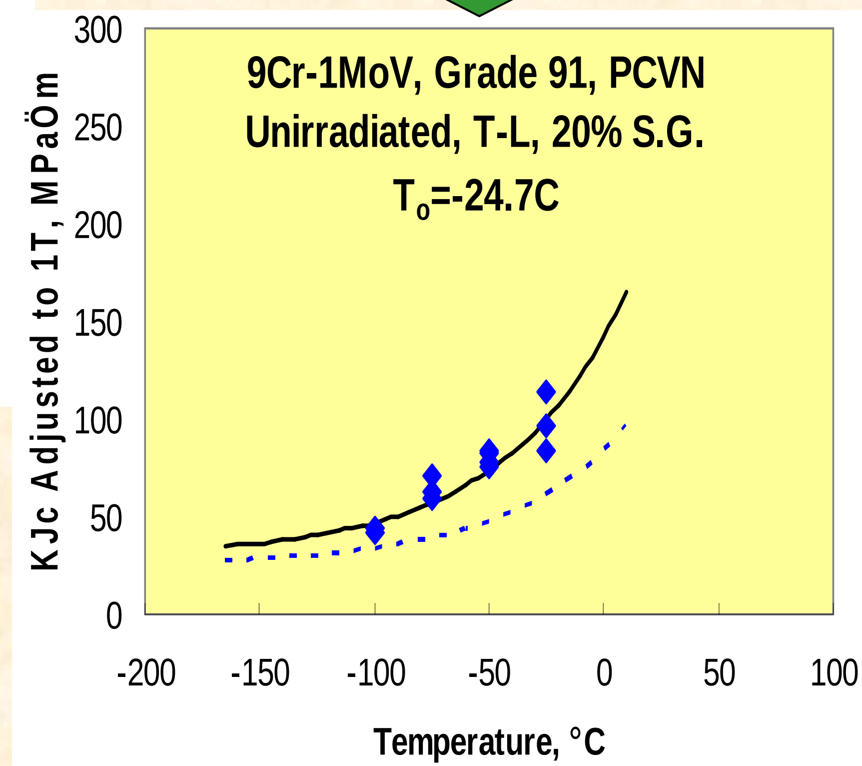
$$T_0 = T_{41J} - 50°C$$

Fracture Testing of 9Cr-1MoV (Heat 30176) with 0.5T Compact and PCVN Specimens Gave T_o Values of –28ºC and –25ºC with Reasonable Relationships to Master Curve

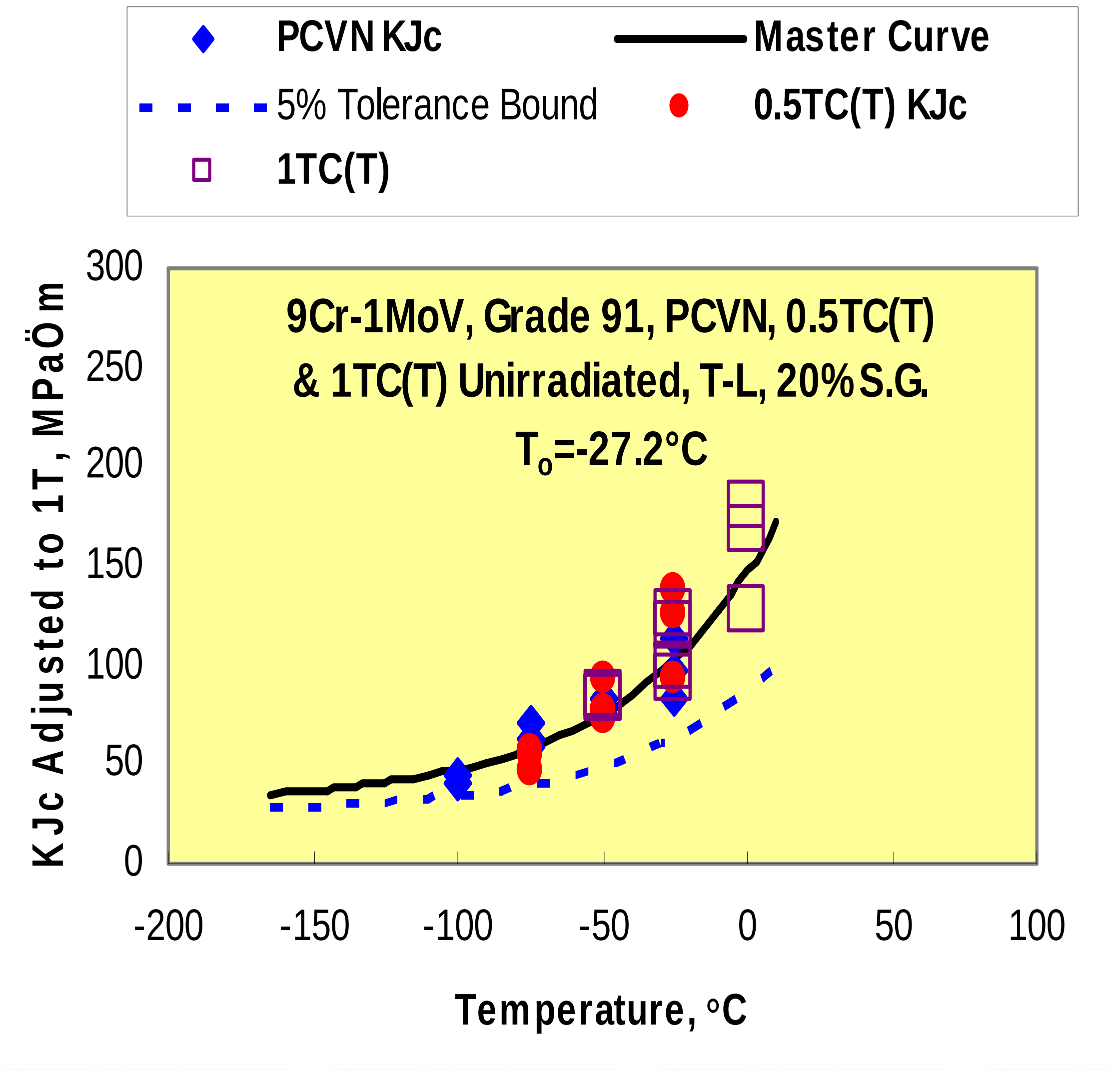


0.5TC(T)

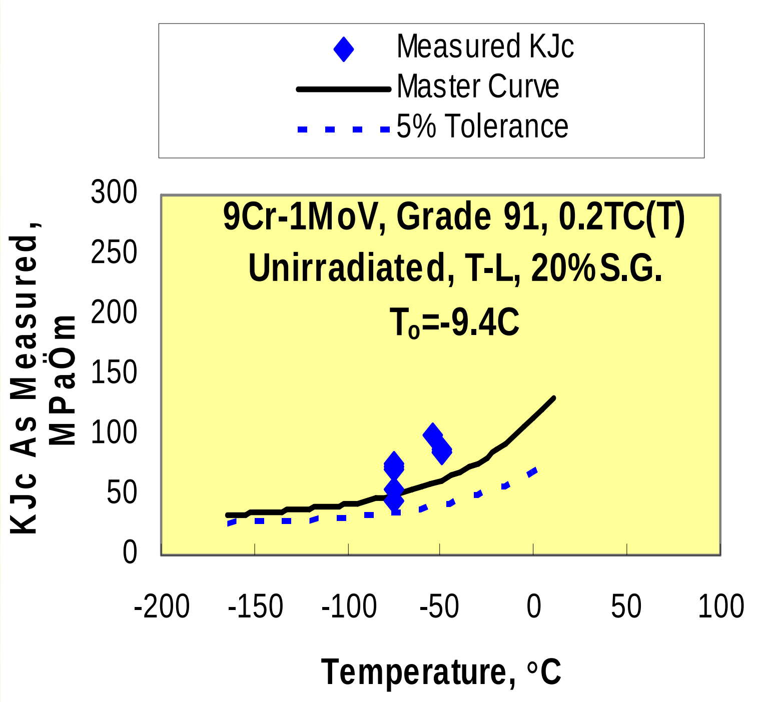
PCVN

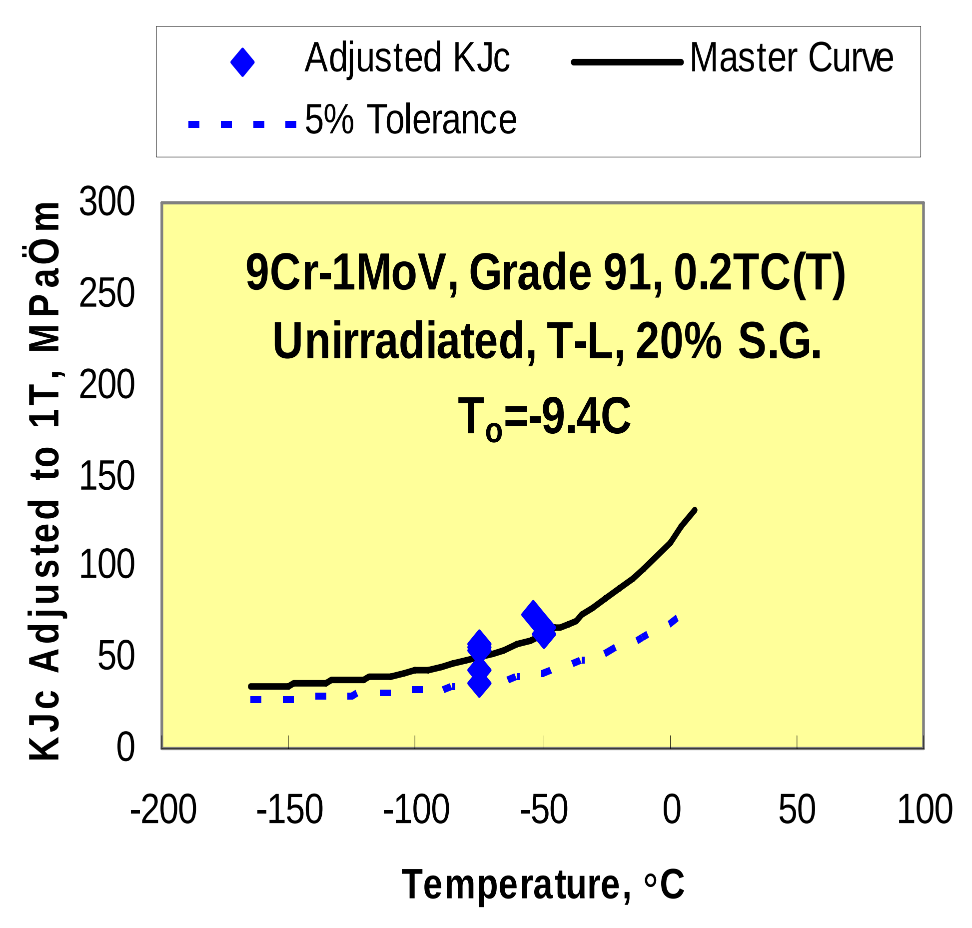


Weakest Link Size Effects For Three Specimen Sizes Show Excellent Relationships With Master Curve for 9Cr-1MoV Steel (Grade 91) With T_0 of -27°C

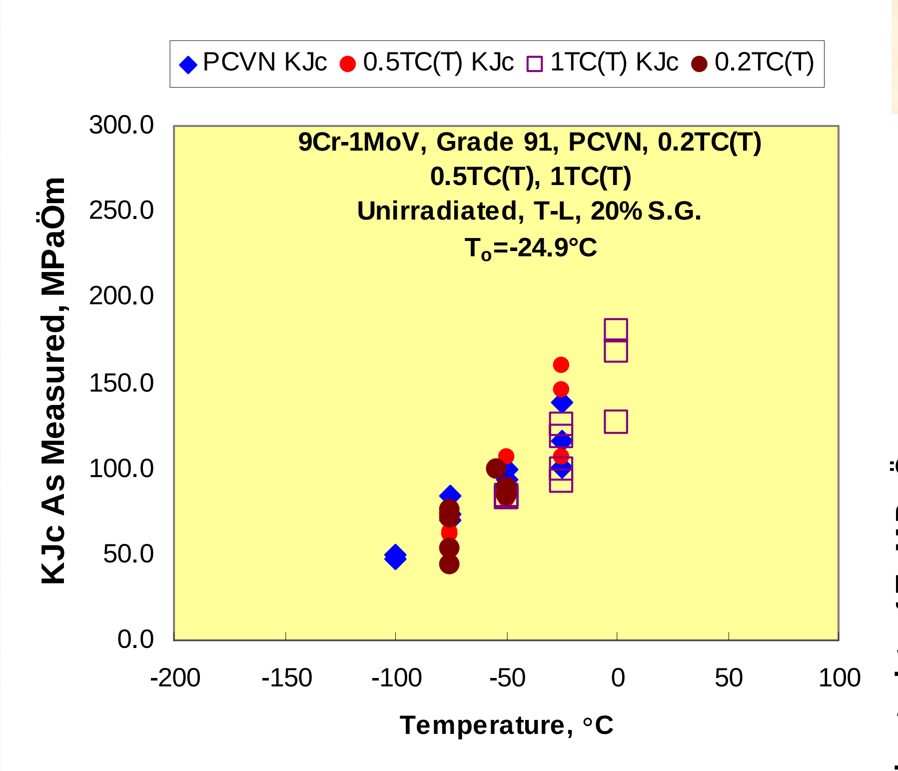


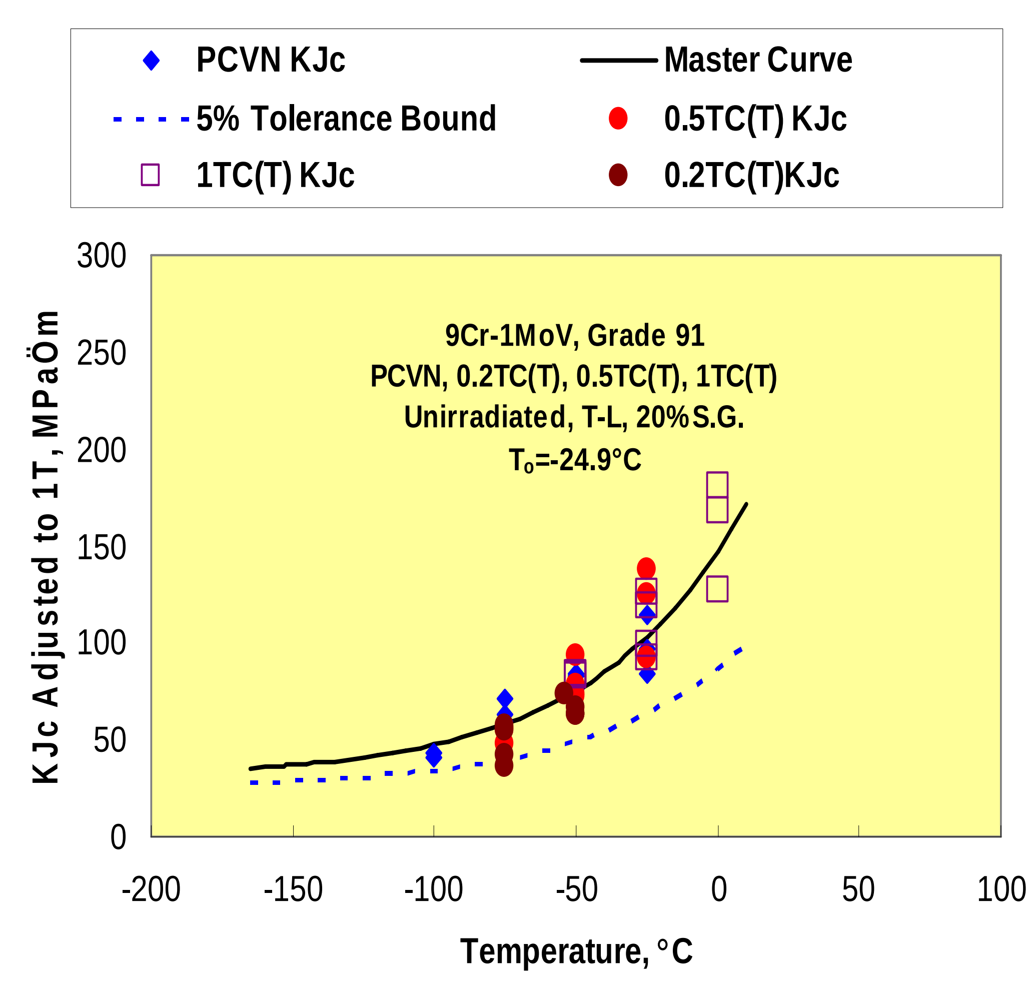
Tests of 0.2TC(T) Specimens Resulted in T_0 of -9°C, Which is 18°C Higher Than That from the Combined PCVN, 0.5TC(T), and 1TC(T)



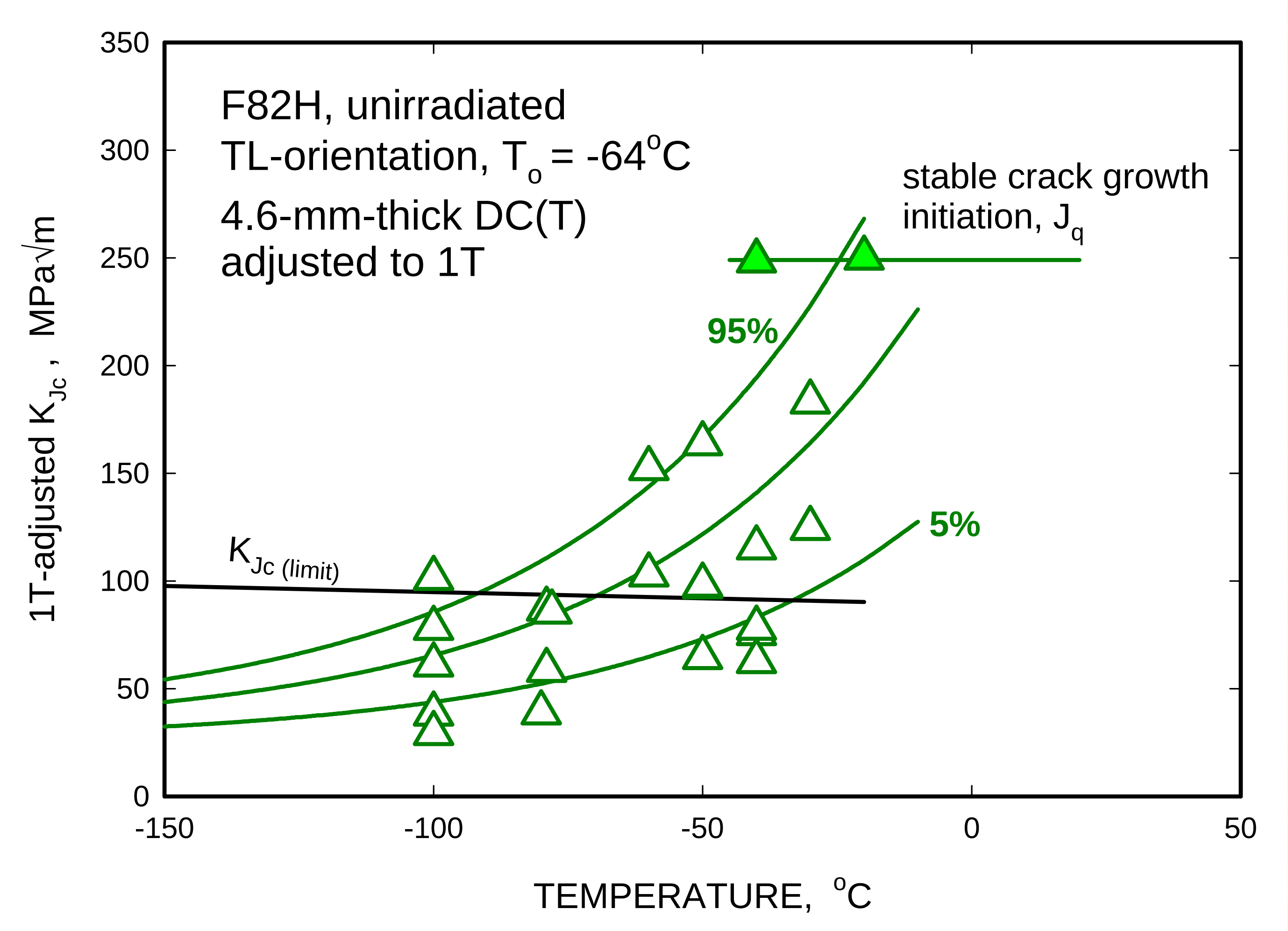


Addition of 0.2TC(T) Specimens to Combined Data Changed T_0 from -27 to -25°C





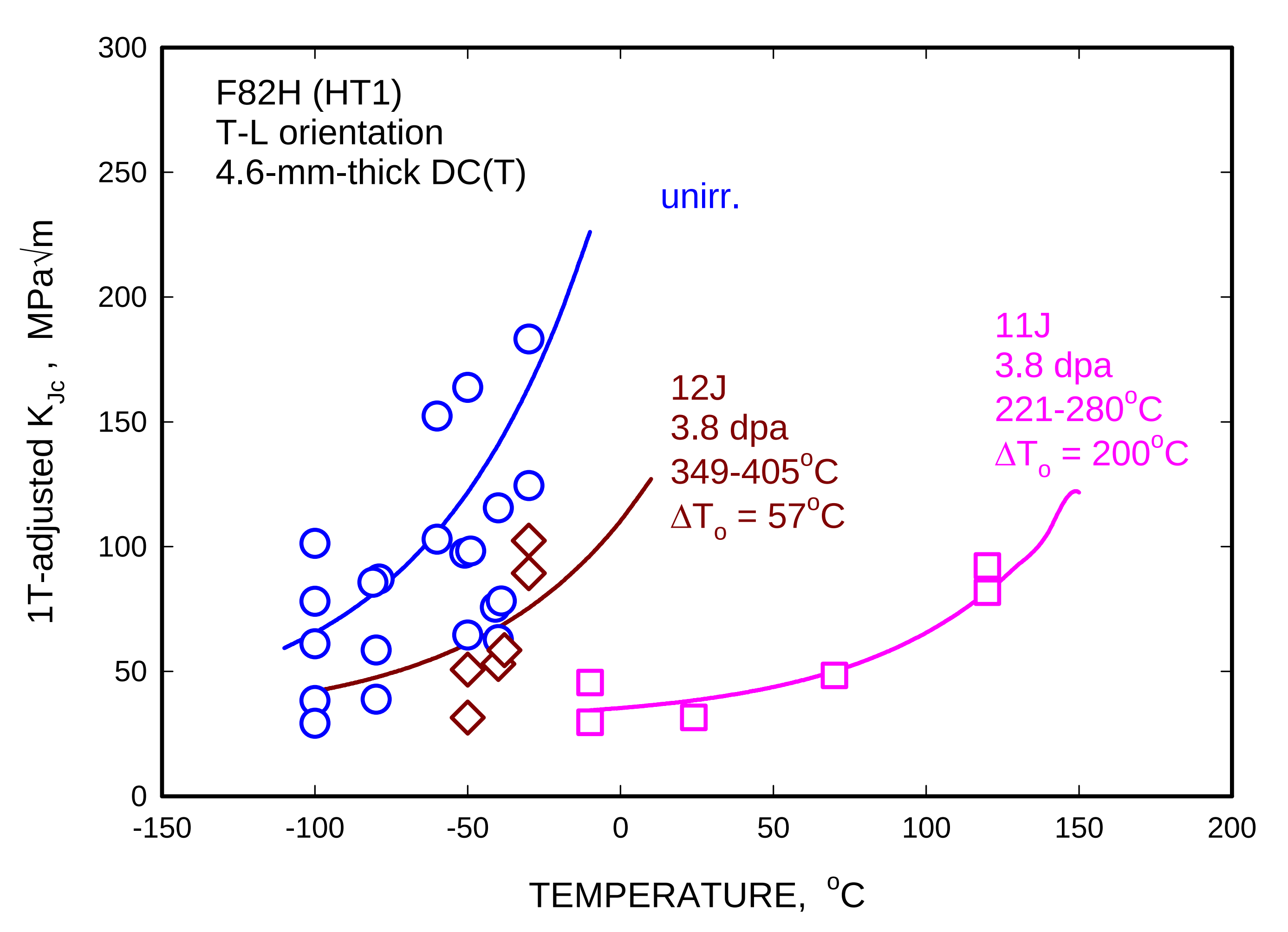
Master Curve Concept Was Used To Evaluate Transition Fracture Toughness of F82H, an 8Cr-2WTa Ferritic-Martensitic Steel



- **US-JAERI Cooperative Program for Fusion Materials Program**
- **Small specimen size leaves a very narrow test temperature window.**

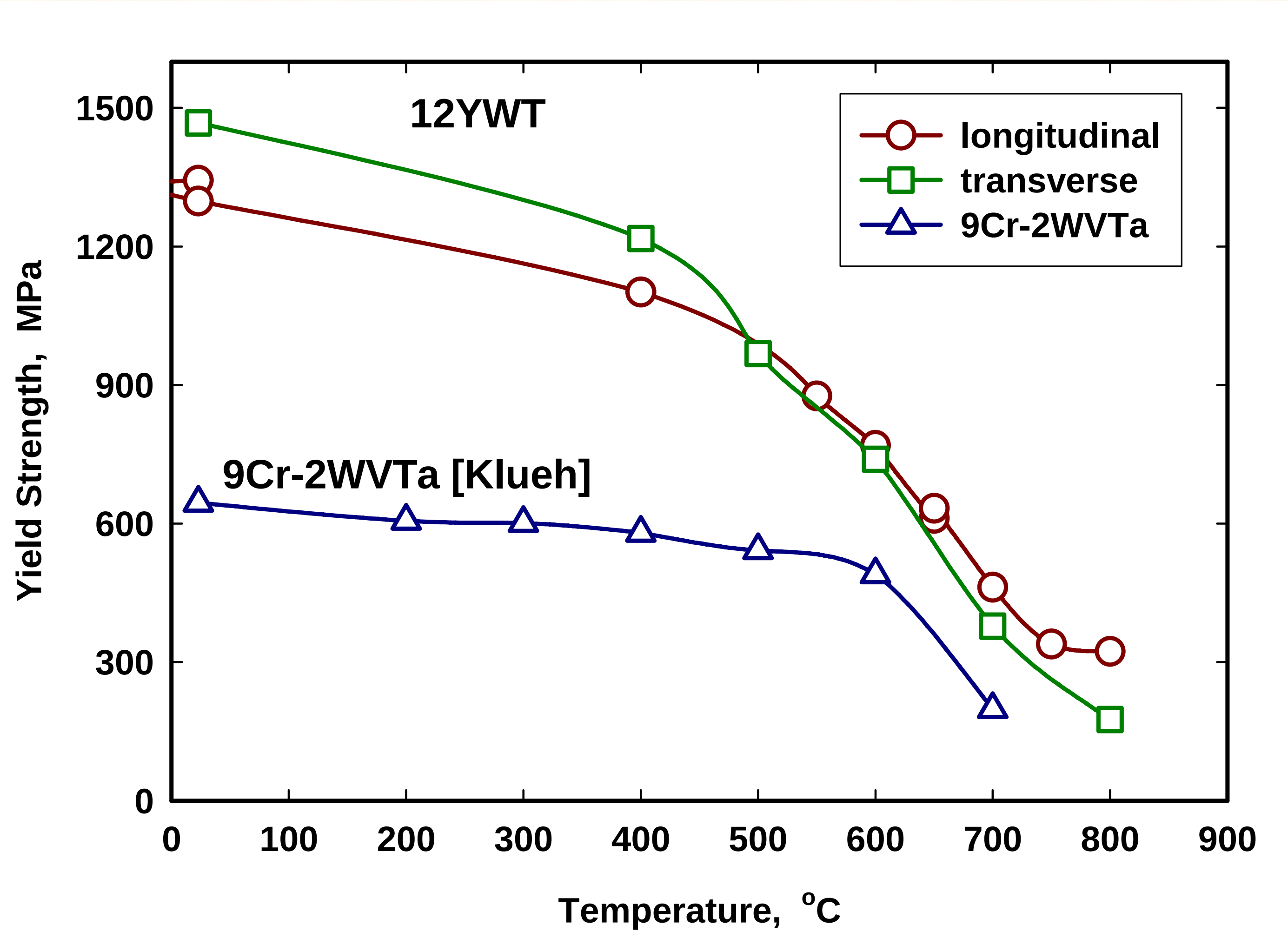
After Sokolov, et. al.

F82H Exhibited a Relatively Modest Shift (57ºC) Of Fracture Toughness After Irradiation at 377ºC But Much Larger Shift (200ºC) After Irradiation at 250ºC



After Sokolov, et. al.

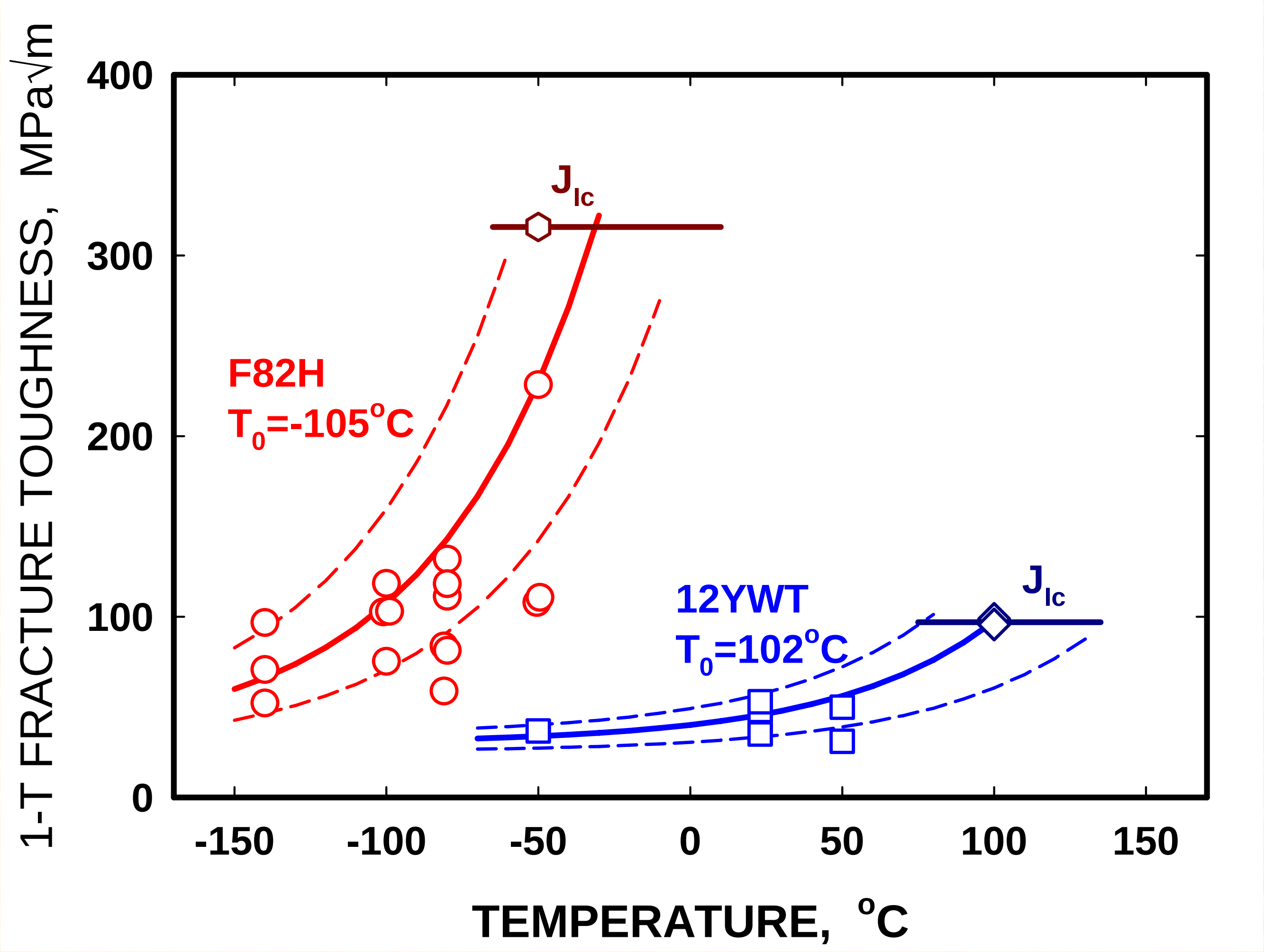
12YWT ODS Alloy Exhibited Excellent Elevated-Temperature Yield Strength Relative to Conventional Steels



Mechanical-alloying (MA) of fine pre-alloyed metal and Y_2O_3 powders used to produce very high strength oxide-dispersion-strengthened alloy of 12Cr-3W-0.4Ti + 0.25Y_2O_3

After Sokolov, et. al.

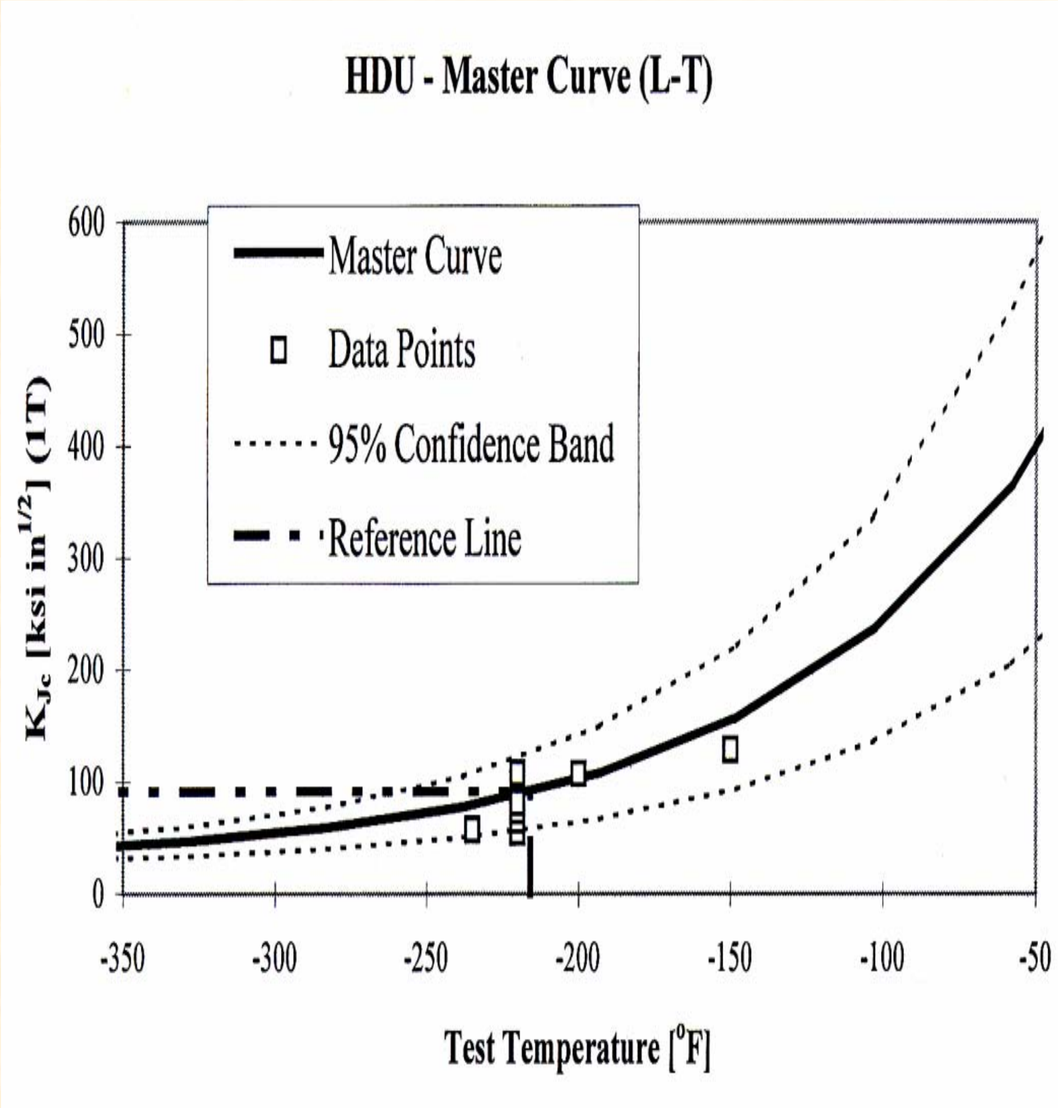
12YWT ODS Alloy Has Significantly Higher Transition Temperature and Lower Upper-Shelf Compared To Typical F/M Steel (F82H)



Three-point bend specimens 1.6×3.2×25 mm were used for fracture toughness testing of 12YWT

After Sokolov, et. al.

HSLA-100 Exhibits Yield Strength of ~ 730 MPa and Very Low T_o (-140ºC) in Plate of 33-mm Thickness



After Mercier, et. al.

HSLA-100 with ~2.6% Ni and YS~730 MPa

- **HSLA-100 with ~3.5% Ni and YS~710 MPa exhibited T_0 = –179°C in 41-mm plate.**

A Number of Advanced Materials Offer Potential for Future Use in Nuclear Reactor Pressure Vessels

- Ferritic-martensitic steels (e.g., 9Cr-1MoV) offer excellent high temperature strength, but are sensitive to irradiation-induced embrittlement at irradiation temperatures less than ~ 450ºC.

- In general, master curve appears to adequately describe fracture toughness vs temperature behavior for ferritic-martensitic steels - although some F-M steels tend to exhibit relatively high scatter outside of tolerance bounds. Fracture toughness data in ductile-brittle transition region are sparse, especially in thick sections.

- Developmental materials such as the oxide dispersion strengthened (ODS) 12YWT alloy (12Cr-3W-0.4Ti + 0.25Y2O3) appear to follow master curve behavior using very small specimens, but such materials have not been made in heavy sections.

- High-strength low-alloy (HSLA) steels (e.g., HSLA-100) offer good fracture toughness with significantly higher strength than current LWR steels, fracture toughness data in ductile-brittle transition region are sparse, especially in thick sections.

- Data in the irradiated condition for all materials discussed are sparse or nonexistent.