Pressurized Water Reactor Pressure Vessels

Material from
"Aging and Life Extension of Major Light Water Reactor Components"
edited by V. N. Shah and P. E. MacDonald
Elsevier, New York, 1993

Introduction

- **"In terms of plant safety, the reactor pressure vessel (RPV) is the most critical pressure boundary component in a PWR"**

- **The RPV ;**

  1.) Vital safety barrier to fission product release
  2.) Supports and guides control rods
  3.) Supports vessel internals
  4.) Provides coolant around the reactor core
  5.) Directs reactor coolant to steam generator

- **2 Major concerns for the RPV.**

  1.) Radiation embrittlement
  2.) Fatigue

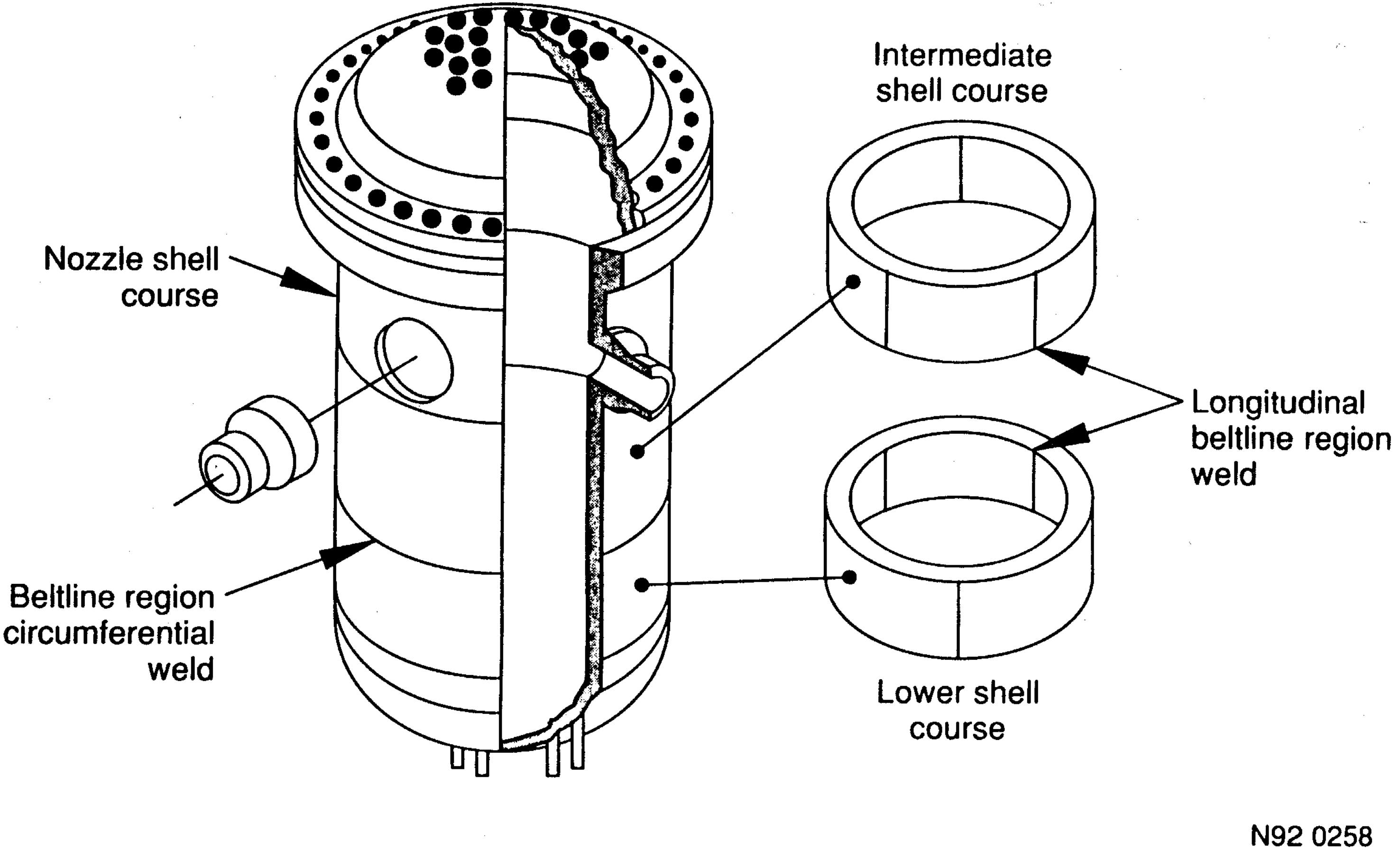


Figure 3-1. Typical PWR pressure vessel with welded plate construction. The beltline region welds are shown.

Design and Materials

- **Major US Vendors for RPV's**

  Combustion Engineering (Now part of a European conglomerate)
  Babcock & Wilcox
  Westinghouse (via CE and B&W, Chicago Bridge & Iron, Rotterdam Dockyard)

- **Different design specifications depending on date of fabrication**

  Before 1963-ASME Boiler & Pressure Vessel Code, Sections I and III.
  After 1963-ASME Boiler & Pressure Vessel Code, Section III.

- **Materials**

  Earliest RPV's used SA302B steel (Table 3-1)
  Most vessels are made from SA533B (Table 3-1)
  Latest RPV's used low Cu/P contents
  Inside RPV is lined with stainless steel (types 304(early), 308 & 309) to reduced corrosion

Table 3-1. Typical PWR pressure vessel steels with their chemical composition and mechanical properties (from the EPRI Reactor Pressure Vessel Materials Database).

		Average chemistry (wt%)				Average mechanical properties (unirradiated)	
Steel	Plant	Cu	Ni	P	S	YS (ksi)	RT_{NDT} (°F)
SA302B	Point Beach-1 (2 heats)	0.16	—	0.013	0.020	62.9	—
SA302B modified	Palisades (4 heats)	0.23	0.49	0.014	0.021	63.8	14
SA533B-1	Diablo Canyon-1 (5 heats)	0.12	0.52	0.011	0.014	65.4	5
SA533B-1 low Cu/P	Vogtle-1 (4 heats)	0.07	0.60	0.007	0.014	66.6	19
SA508-2	North Anna-1 (3 heats)	0.15	0.79	0.013	0.014	73.4	28
SA508-3	Braidwood-1 (3 heats)	0.04	0.72	0.008	0.007	64.1	-15

- **Heat Treatments**

  All vessel welds were post heat treated at ≈ 610 ± 14 °C for 40-50 hr's (early) and ≈ 25 hr's in the newer RPV's.

- **Diameters**

  Westinghouse-3.35 to 4.11 meters
  Babcock & Wilcox-4.34 meters
  Combustion Engineering-3.99 to 4.37 meters
  Combustion Engineering System 80- 4.62 meters

- **See Figure 3-1**

Stressors

- **Primary Stressors**

  Mechanical pressure loads during operation
  Periodic thermal transients
  Dead weight loads
  Pressurized thermal shock

- **Other Important Parameters**

  Temperature
  Water Chemistry
  Mechanical Contact

• **Ductility is an important measure of performance**

> ***Charpy V-notch---(CVN)***
> ***Ductile to brittle transition temperatures (DBTT)***
> ***Upper shelf energies (USE) (see figure 3-2)***

<u>Pressure-Temperature (P-T) Limits</u>

• **PWR vessels typically experience pressures of 15.5 MPa (2250 psi) and temperatures of nearly 288 °C (550 °F) during normal steady state operation.**

• **Perturbations to these conditions are what set the limits to RPV performance.**

• **P-T limits require that plants operate above certain minimum and below certain maximum limits**

> ***Minimum T to be above DBTT***
> ***The reactor coolant pump characteristics govern the maximum T***

• **See Figure 3-3**

> ***Note: if a critical size defect had been present at a critical site and the degree of radiation embrittlement had been severe***

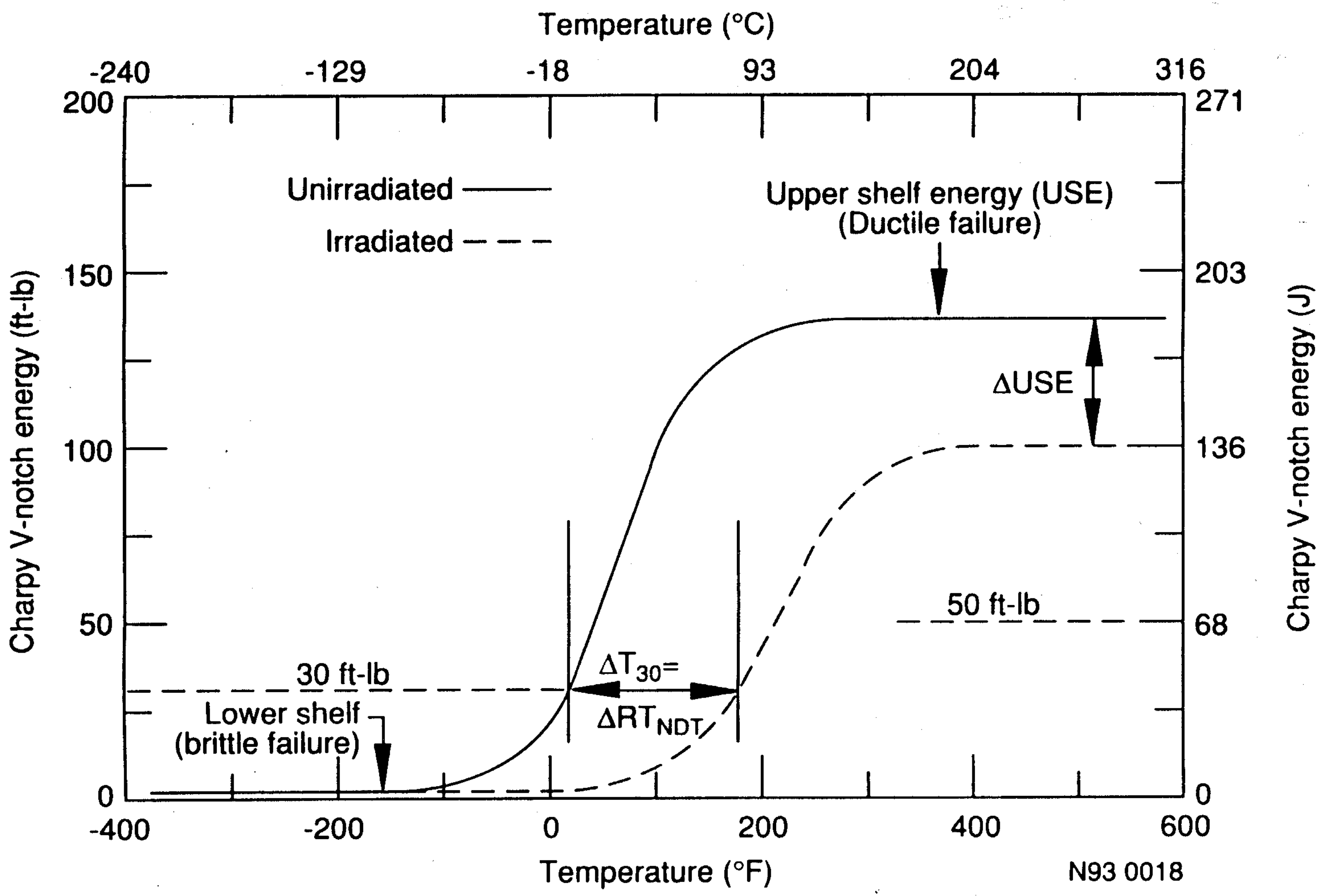


Figure 3-2. Charpy V-notch surveillance data, showing radiation embrittlement effects.

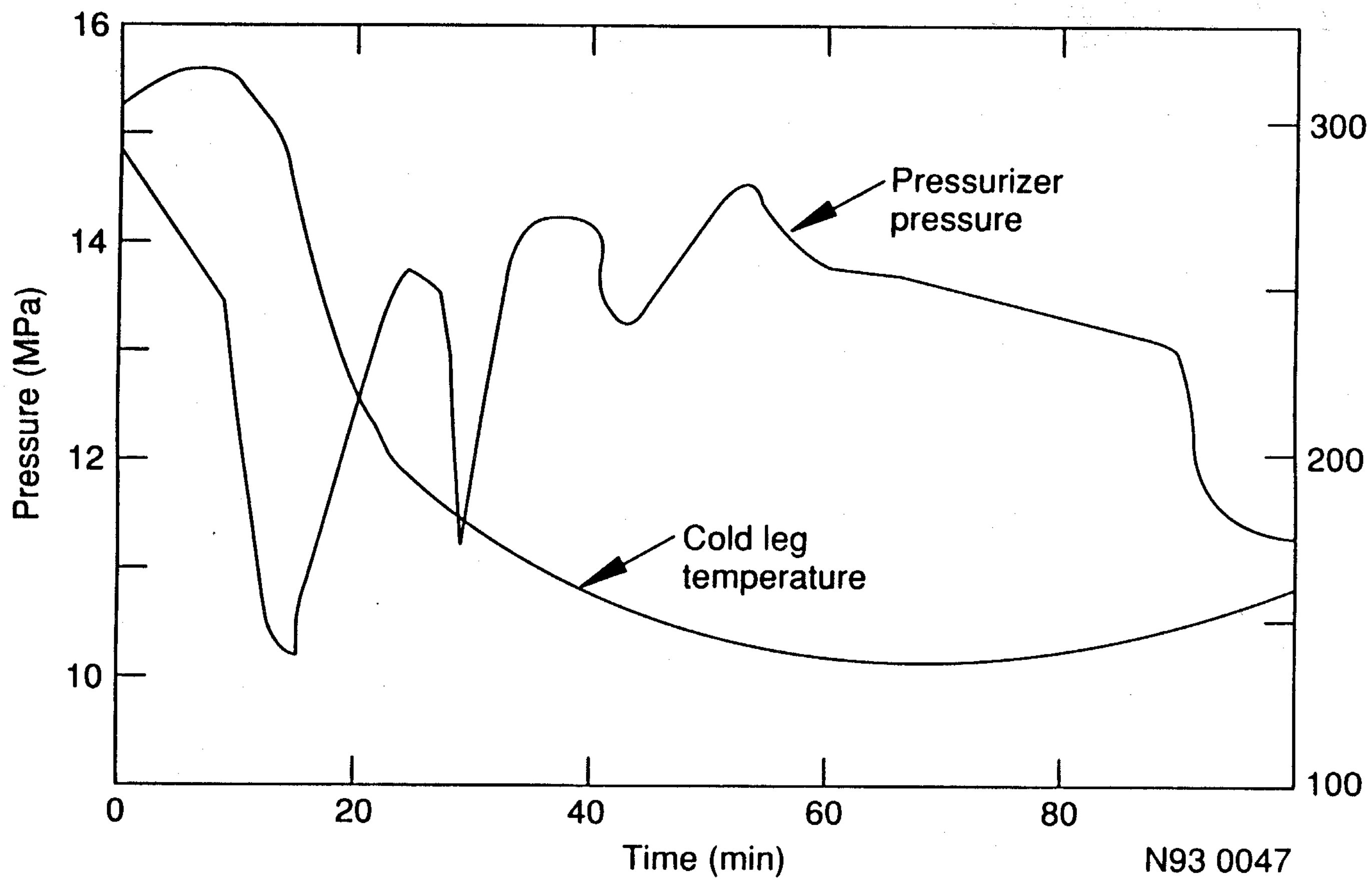


Figure 3-3. Pressure and temperature variations during the 1978 accident at Rancho Seco (Iskander 1986). Copyright American Society for Testing and Materials; reprinted with permission.

enough, this transient might have resulted in the rupture of the pressure vessel.

- **Primary Transients Leading to Fatigue**

  1.) Plant heatup/cooldown
  2.) Plant loading/unloading
  3.) Reactor trips
  4.) Loss of flow
  5.) Abnormal loss of load

  See Table 3-2

Degradation Sites

- **Beltline region (embrittlement)**

  Welds may be weakest link because early welding materials used Cu coated filler rods

- **Geometric discontinuities (fatigue)**

  Closure studs
  Outlet nozzles
  Inlet nozzles
  Instrumentation nozzles
  Control rod drive nozzles

Degradation Mechanisms

- ***Generally corrosion and stress corrosion cracking are not a problem in PWR RPV's because water contains low O_2***

- ***Erosion and cavitation not a problem***
- ***High T creep not a problem***

Radiation Embrittlement

- **Neutron fluence range-**

  10^{18} to 10^{19} n/cm^2 (E> 1 MeV)

- **Result for Charpy V-notch (CVN) specimens:**

  Increase in reference DBTT (RT_{NDT}) (usually measured at 41 J [30 ft-lb] energy, or, T_{30})

  Drop in upper shelf energy (USE)

Design Criteria-Fracture Toughness

• ***Even though one knows σ_{ys}, σ_u, and E, how does one deal with materials of varying thicknesses which contain notches (cracks)?***

Stress Intensity Factors

• **Defined by G. R. Irwin as:**

$$K = \sigma_{ave}\sqrt{\pi c}$$

where σ_{ave} = average stress
c = half length of the crack

• **Mode of Deformation (figure 13-8)**

K_I, K_{II}, K_{III}, ...

• **Fracture toughness, K_c**

K_c represents a critical event similar to yielding in a simple tensile test. The notch, or flaw, suddenly begins to grow, and complete fracture occurs.

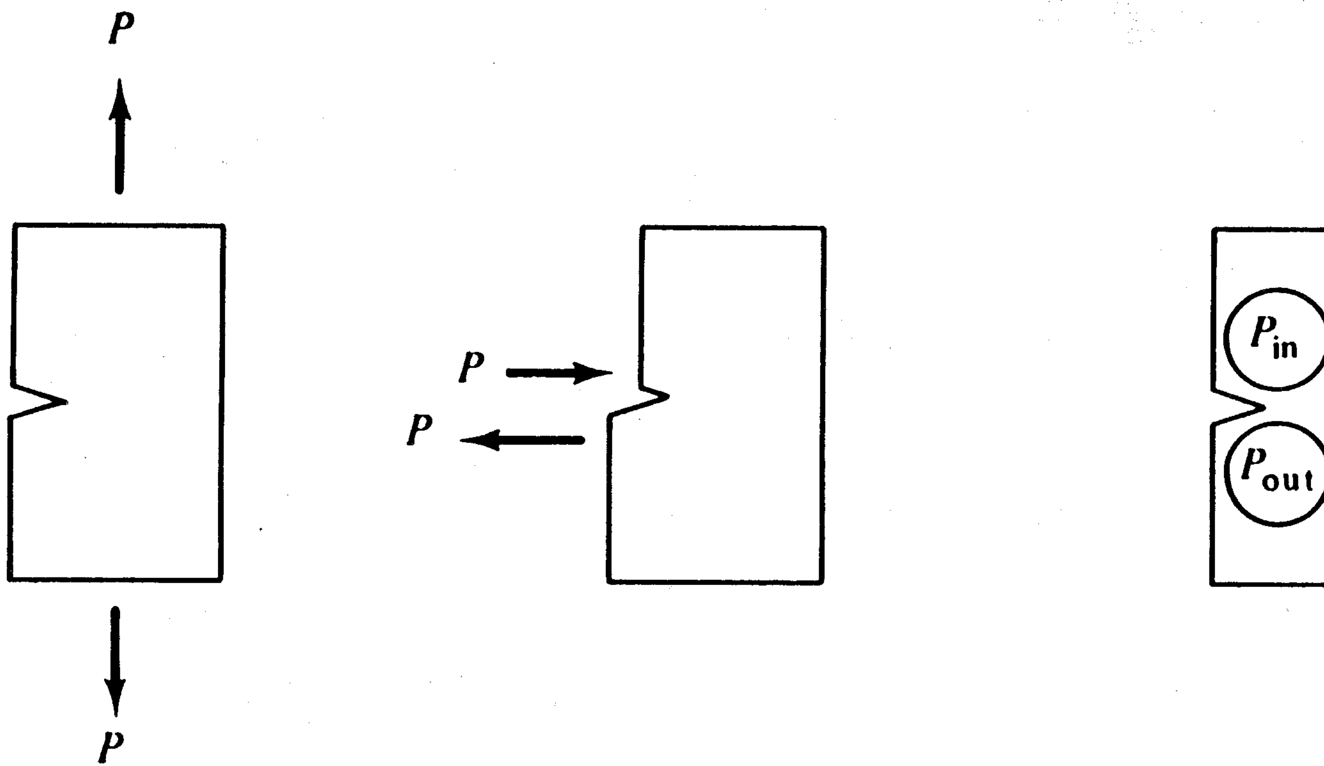
• **Kc depends on thickness of specimen**

(Figure 13-10)

K_{Ic} is the plane strain fracture toughness and the "safe" value

Table 3-2. Typical plant transients and assumed design occurrences (Yahr et al. 1986, Griesbach 1984).

Transient	Number of events
Plant heatup at 55°C (100°F)/h	500
Plant cooldown at 55°C (100°F)/h	500
Plant loading at 5% full power/min	15,000
Plant unloading at 5% full power/min	15,000
Step load increase of 10% full power	2,000
Step load decrease of 10% full power	2,000
Reactor trip from full power	400
Loss of flow and abnormal loss of load	80
Loss-of-secondary pressure	5
Hydrotest to (21.55 MPa) 3125 psig, 204°C (~400°F)	10
Operating-basis earthquake	200
Normal plant variation [100 psi and 12°C (~10°F)]	$>10^6$



(a) mode I (opening mode).

(b) mode II (shear mode) (forces are in plane of plate).

(c) mode III (shear mode) (forces are normal to plane of plate).

Fig. 13-8. Modes of deformation of notched members.

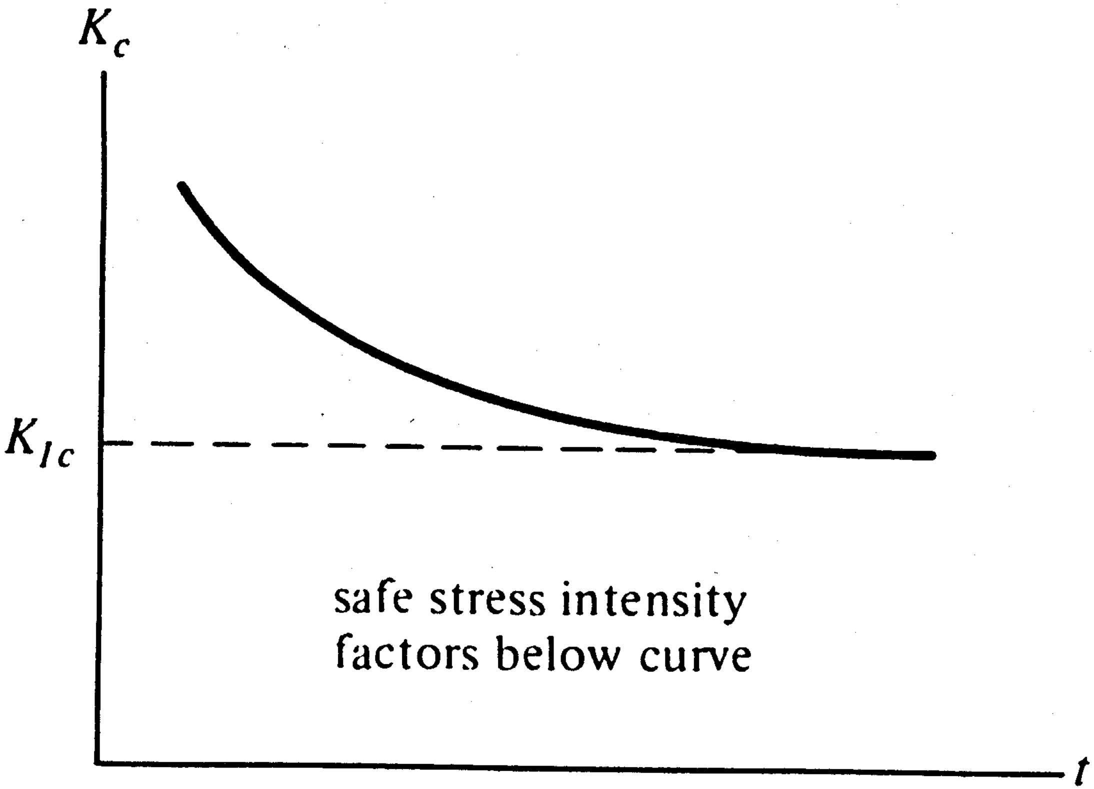


Fig. 13-10

- **Crack arrest toughness, K_{Ia}**

Ability of material to arrest a dynamically propagating crack under plain strain conditions

Problem

The steel 4340 is chosen for a certain structural member. It has the following properties:

$$\sigma_{ys} = 1.5\ \frac{kN}{mm^2}$$

$$\sigma_u = 1.85\ \frac{kN}{mm^2}$$

$$K_{Ic} = 1.5\ \frac{kN}{mm^{3/2}}$$

What is the largest crack that can be tolerated in this steel if the maximum average operating stress is 60% of the ultimate strength?

<u>Answer</u>

The critical stress is:

$$\sigma_c = \frac{K_{Ic}}{\sqrt{\pi c}} = 0.6\sigma_u$$

and the largest allowable crack is:

$$2c = \frac{2K_{Ic}^2}{(o.6\sigma_u)^2\pi} = 1.1mm$$

- **Relationship between allowable fracture toughness, operating temperature, and DBTT (see figure 3-4).**

Note: Data obtained in reactors at 10^{13} n/cm^2-s may give slightly different results than a 10^{10} n/cm^2-s flux at RPV walls

- **What is the final DBTT that should be used?**

Final DBTT = Initial DBTT + ΔDBTT + Margin

- **Below 4×10^{19} n/cm^2:**

$$\Delta DBTT = [470Cu + 350(Cu \times Ni) - 10]\, f^{0.27} \qquad [3\text{-}2]$$

where Cu, Ni = wt%
DBTT in °F
f = fluence in units of 10^{19}

- **Above 4×10^{19} n/cm^2:**

$$\Delta DBTT = 283\, f^{0.194}$$

- **Margin Term (use 2 standard deviations)**

$$2\sigma = 2\sqrt{\sigma_0^2 + \sigma_d^2}$$

where σ_o = 0 if DBTT is measured
= 17 if DBTT not measured

σ_d = 24 if $\Phi t < 4 \times 10^{19}$

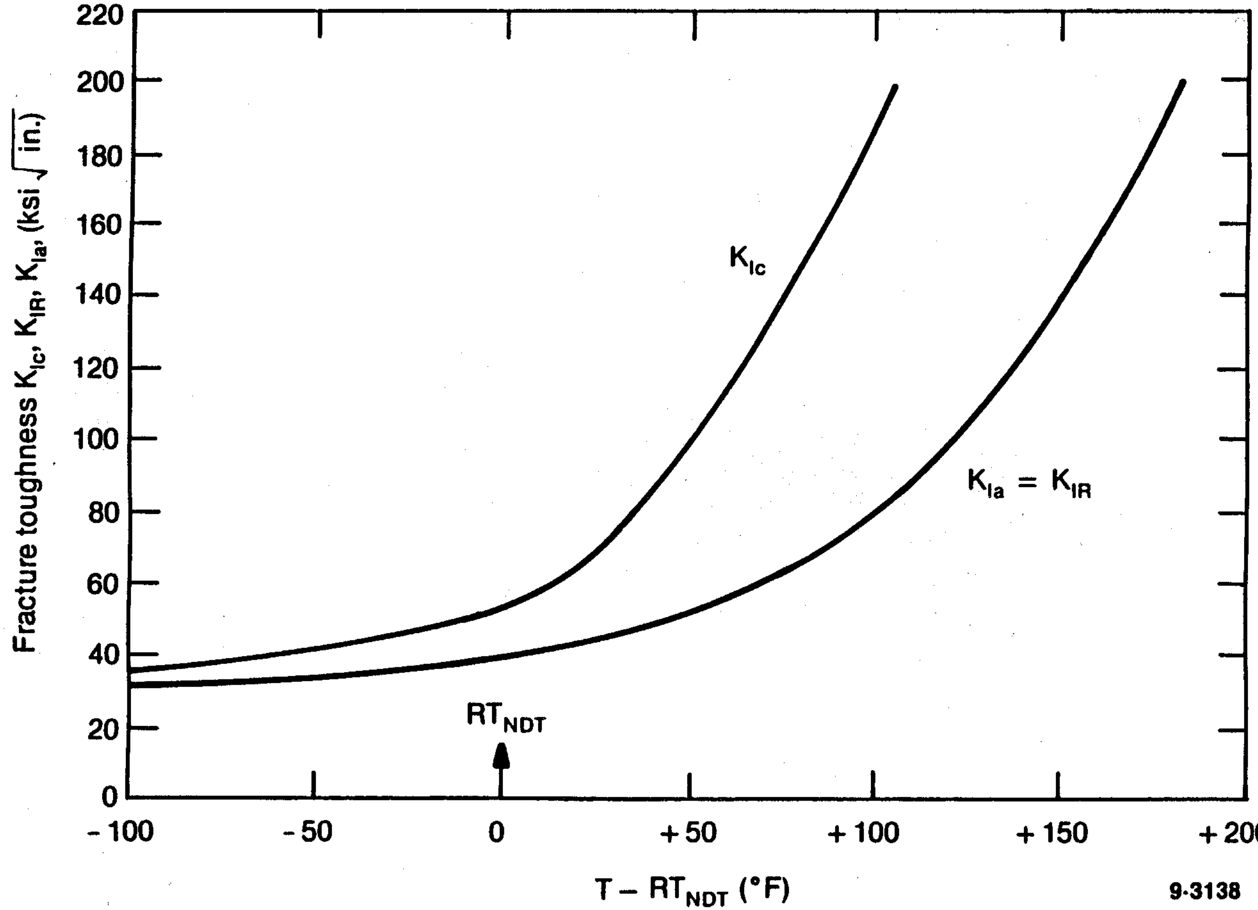


Figure 3-4. Lower bound K_{Ia}, and K_{IR}, and K_{Ic} curves for commercial pressure vessels (ASME 1986b). Copyright American Society of Mechanical Engineers; reprinted with permission.