Structural Integrity Assessment of VVER-1000 RPV under Accidental Cool down Transients

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Scope of the Presentation

- → Overview of Indian Nuclear Power Program
- → Upcoming Reactors in India
- → Safety Assessment of VVER-1000 RPV: Specific Results
- → Parametric Studies

THE THREE-STAGE INDIAN NUCLEAR POWER PROGRAM

→ The importance of nuclear energy, as a sustainable energy resource for our country, was recognized at the very inception of our atomic energy program more than four decades ago.

→ A three-stage nuclear power program, based on a closed nuclear fuel cycle, was then chalked out. The three stages are:

1. Natural uranium fuelled Pressurised Heavy Water Reactors (PHWRs),

2. Fast Breeder Reactors (FBRs) utilising plutonium based fuel (available from spent fuel reprocessing of PHWRs), and,

3. Advanced nuclear power systems (AHWR) for utilization of our abundant thorium reserves

Advantages of the 3 stage Program:

(1) Closed Fuel Cycle

(2) Utilization of Country's Thorium Reserves for LT Energy Security

Nuclear Reactor Technologies in India

PHWR

- Uses Nat. U Fuel
- Heavy Water Cooled
- Heavy Water Moderated
- Light Water Steam
- Pressure Tube type

LWR

- Uses Enriched U Fuel
- Light Water Cooled
- Light Water Moderated
- Light Water Steam
- Pressure Vessel type

FBR

- Uses Pu-239 (+ U- 235 + Th-232) Fuel
- Generation of U-233 from Th-232
- Liq. Sodium Cooled

AHWR

- Uses U-233 (+ Th-232 Fuel)
- Self sustaining Th-U Fuel cycle

BWR → Boiling & Steam generation takes place inside the Reactor Vessel

PWR → Steam generation takes place outside the Reactor Vessel- In Steam Generator

Operating Nuclear Power Reactors in India

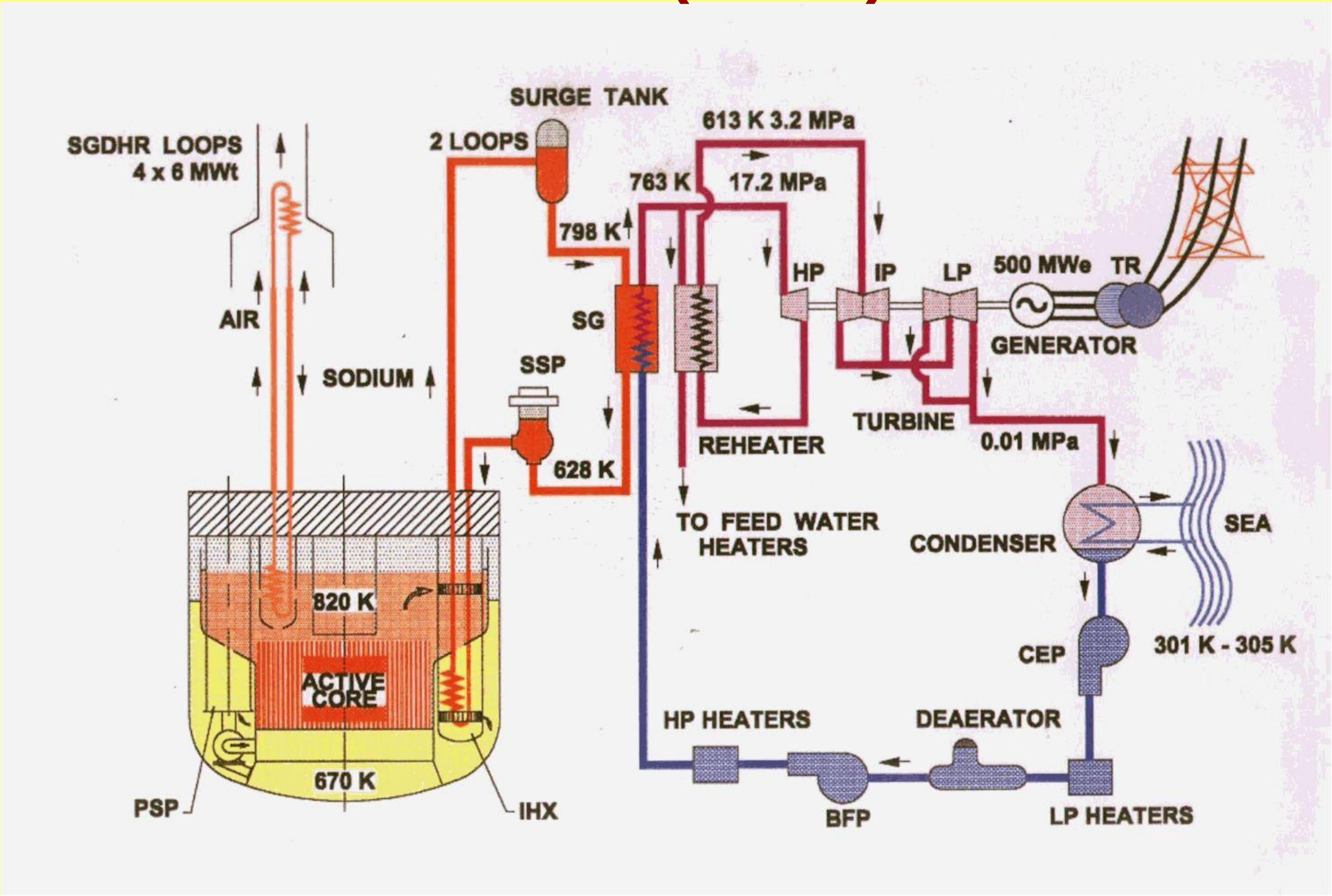
Location of NPP	Type/Capacity
Tarapur	**BWR/**2x160 MWe **PHWR**/2x540
Rajasthan	**PHWR**/1x100 MWe, 1x200 MWe and 4x220 MWe
Kalpakkam	**PHWR**/Unit 1-170 MWe Unit 2-220 MWe
Narora	**PHWR**/2x220 MWe
Kakrapara	**PHWR**/2x220 MWe
Kaiga	**PHWR**/4x220 MWe

- ✓ In all 20 Units are Operational As On Date.
- ✓ Total Nuclear Power Plant Capacity As On Date: **4780 MWe**
- ✓ Tarapur BWR is the Oldest Unit with 40 yrs of Service

Upcoming Nuclear Power Reactors in India

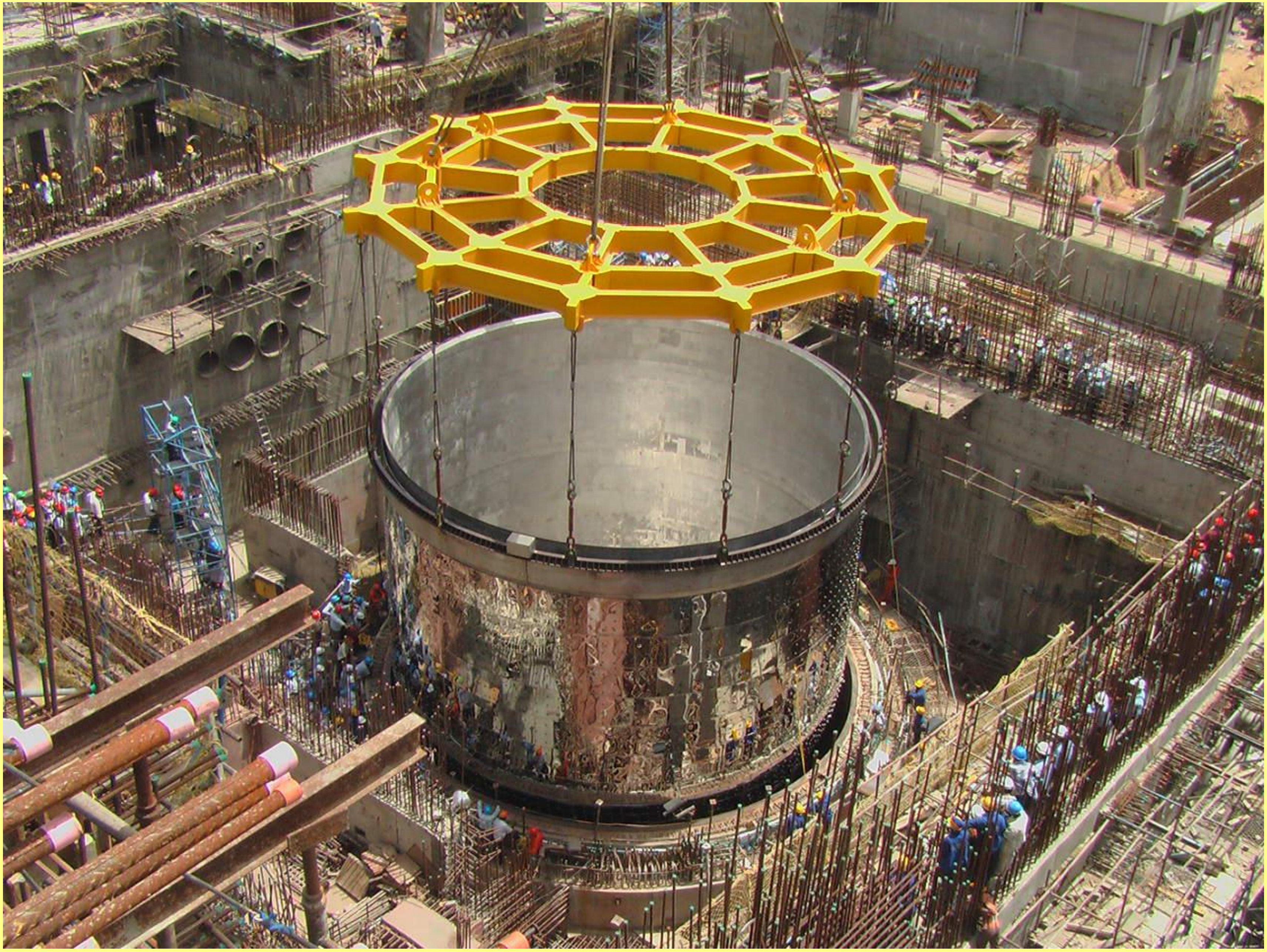
Location of NPP	Type/Capacity	Expected Commercial Operation
Kudankulam	**PWR (VVER)/** 2 x 1000 Mwe	-----
Jaitapur	**PWR**	-----
Kakrapar (3 & 4)	**PHWR**/2x700 MWe	2015
Rajasthan (7 & 8)	**PHWR**/2x700 MWe	2016
Kalpakkam	**PFBR/**1x500 MWe	2015
--	**AHWR/**1x300 MWe	--

Upcoming Prototype Fast Breeder Reactor (PFBR)



NUCLEAR ISLAND BHAVINI :
PROTOTYPE FAST BREEDER REACTOR UNDER CONSTRUCTION AT KALPAKKAM SITE IN THE EASTERN COASTAL REGION OF INDIA

Erection of the Main Vessel of PFBR



Erection of the Inner Vessel of PFBR

Conceptual Layout of AHWR

Upcoming VVER type NPP

NPP with 2 units of VVER-1000 under Construction

Safety Assessment of VVER-1000

- Safety Assessment carried out for initial Licensing

- Structural Integrity Evaluation of RPV a mandatory task for Safety Assessment

  → At the beginning of Life for Licensing

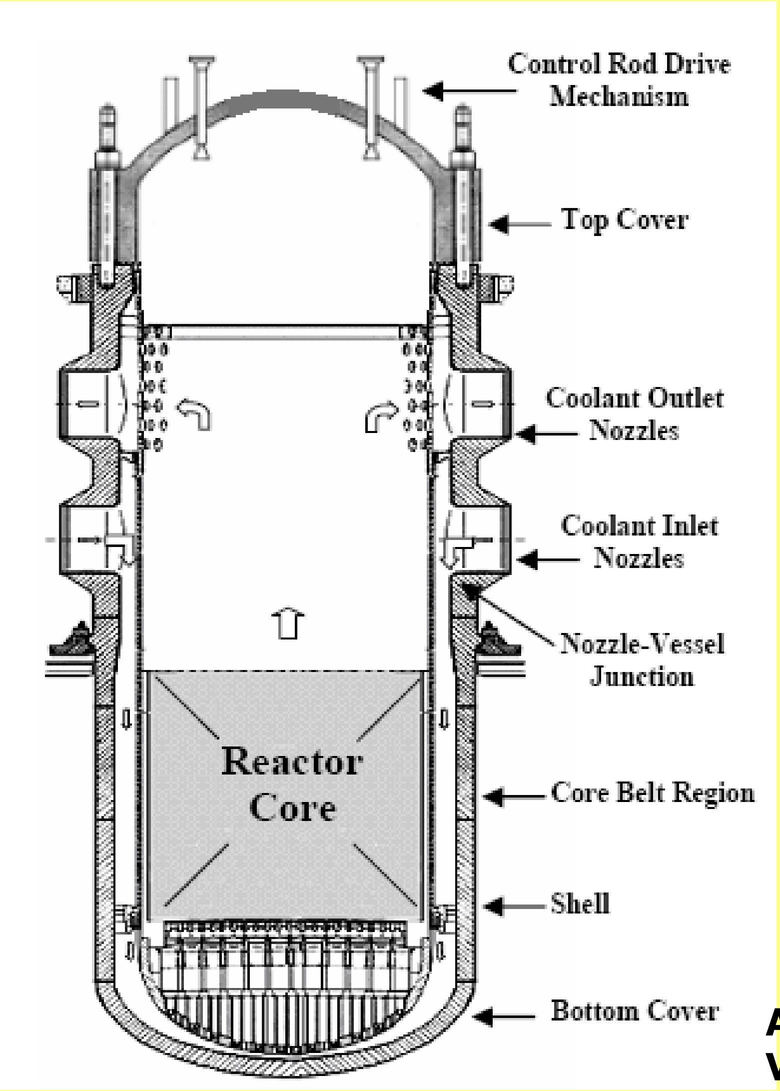
  → At Regular intervals during operation for PLiM to ensure safety & LTO

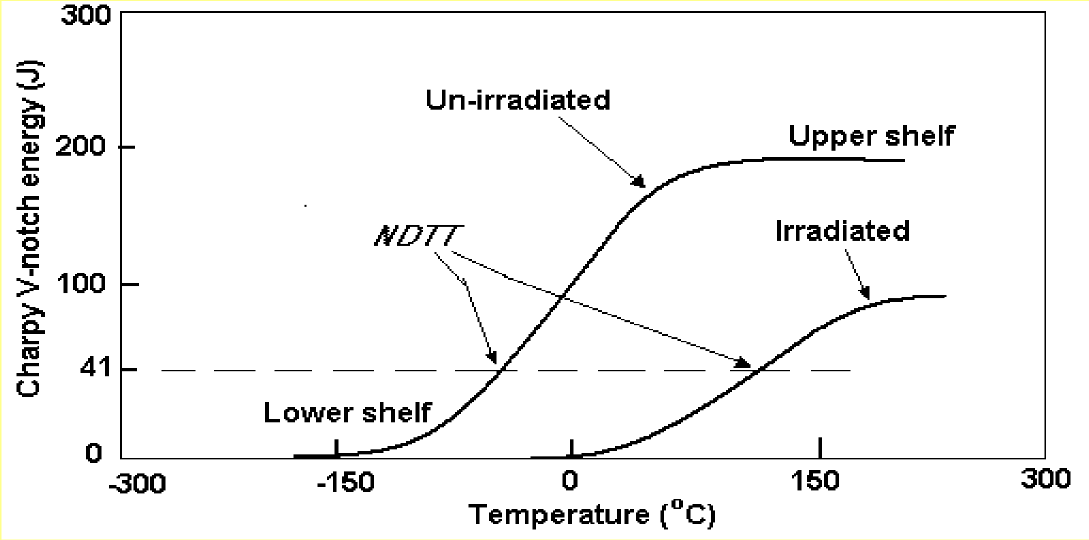
- This Paper Presents the Specific Results of Integrity Assessment of RPV under Postulated LOCA conditions

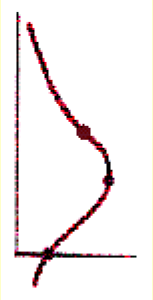
- Guidelines from both PNAE & ASME are considered & justifiable Best Practice adopted.

- Based on the results, Inferences are drawn for successful LTO

RPV of a LWR: Critical to Safety & LTO





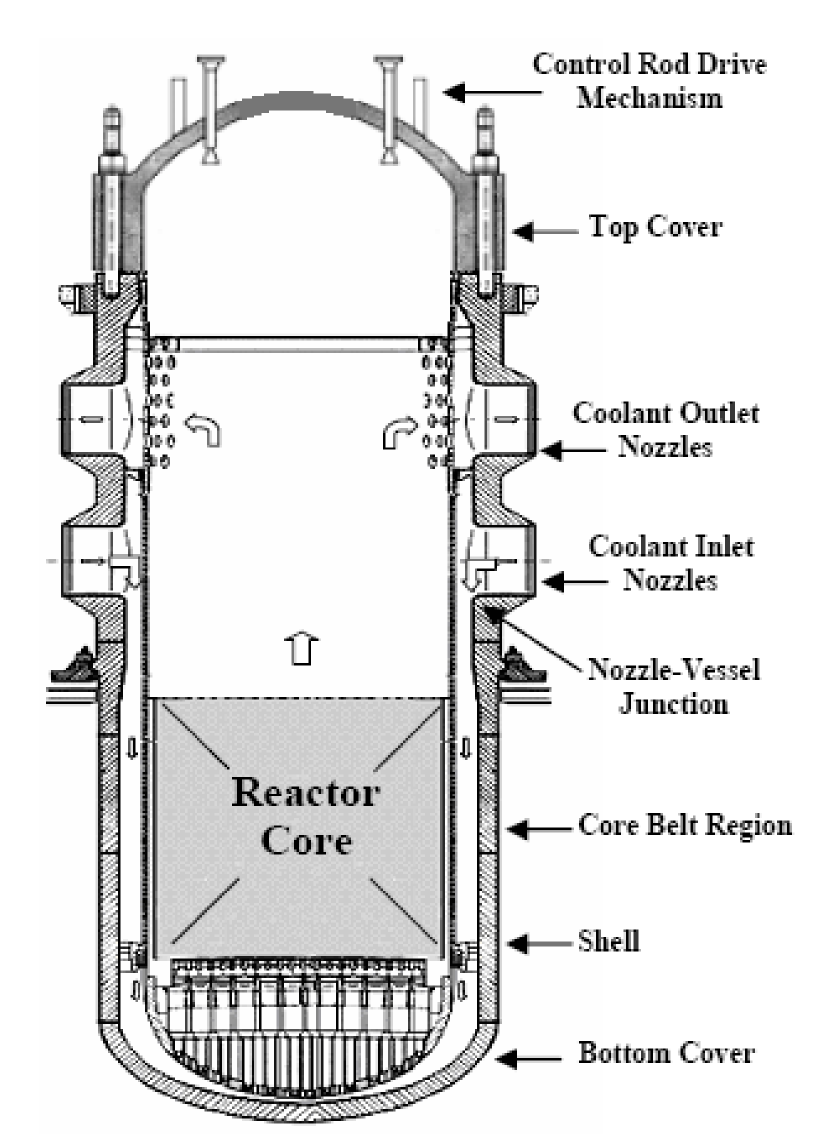


Axial Flux Variation

GENERAL HIGHLIGHTS:

- **KEY COMPONENT OF NPP**
- **CRITICAL TO SAFE OPERATION**
- **EXPERIENCES SEVERE LOADS UNDER NORMAL & EMERGENCY CONDNS.**
- **INSERVICE DEGRADATION OF MATERIAL PROPERTIES DUE TO IRRADIATION DAMAGE**

Reactor Pressure Vessel: VVER-1000



Dimensions of Core Belt Region:

Inner diameter (excluding clad) = 4150 mm,
Base metal wall thickness = 192.5 mm,
Length = 3500 mm

Material of Construction:

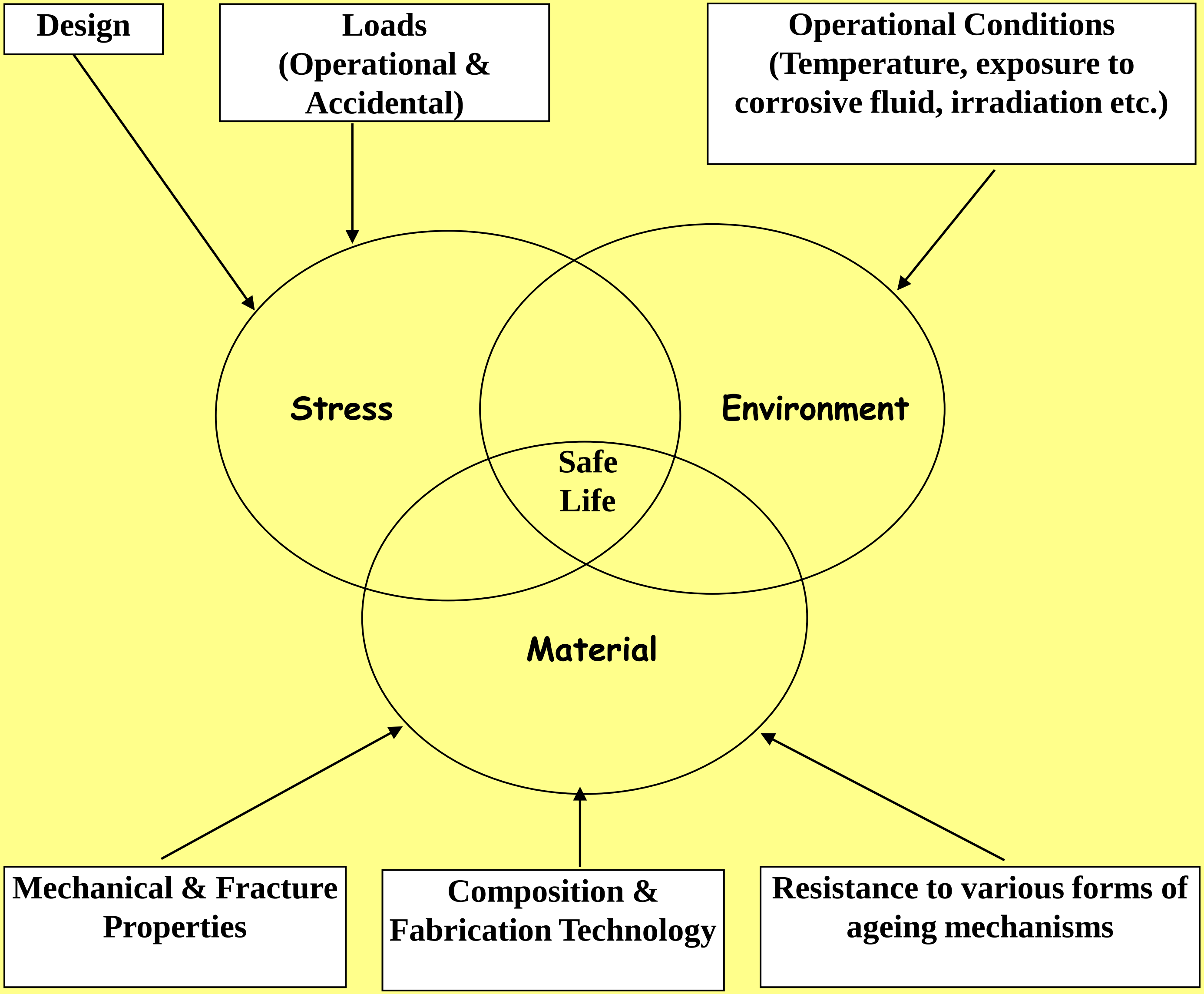
Cr-Mo-Ni-V **Low Alloy Steel**

Normal Operating Conditions:

(a) Pressure = 16 MPa
(b) Temperature = 280 ^{0}C

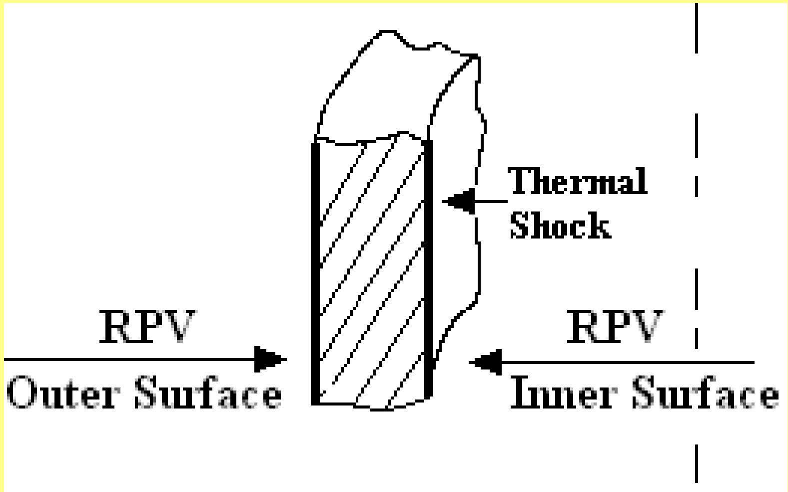
4 Loops of Main Coolant System
3 Loops of HP ECCS
3 Loops of LP ECCS

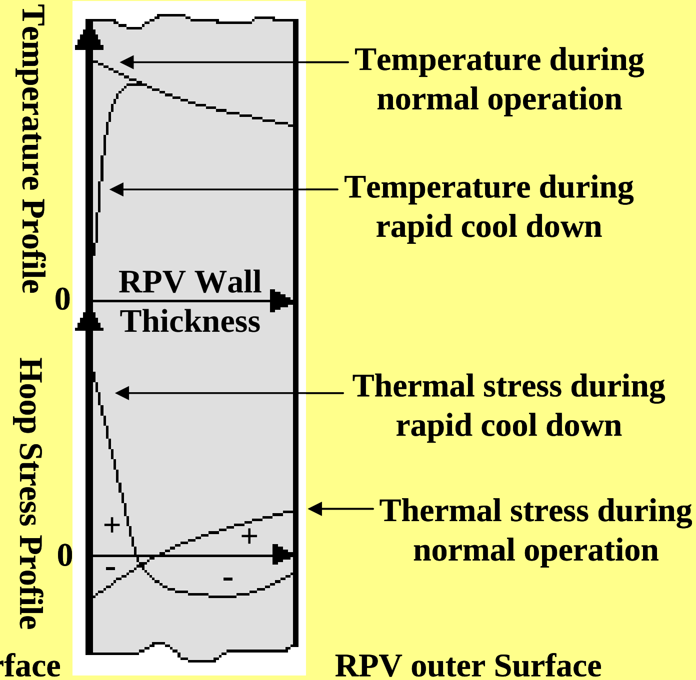
Factors Governing Safe Operating Life



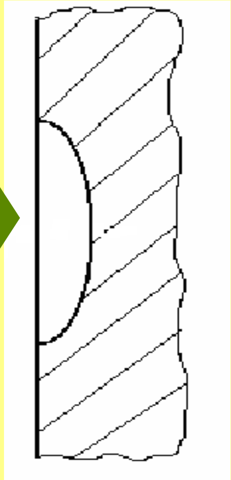
LOCA: A Low Probability but High Severity Event

ECCS Injection during LOCA leads to severe cool down transients on RPV



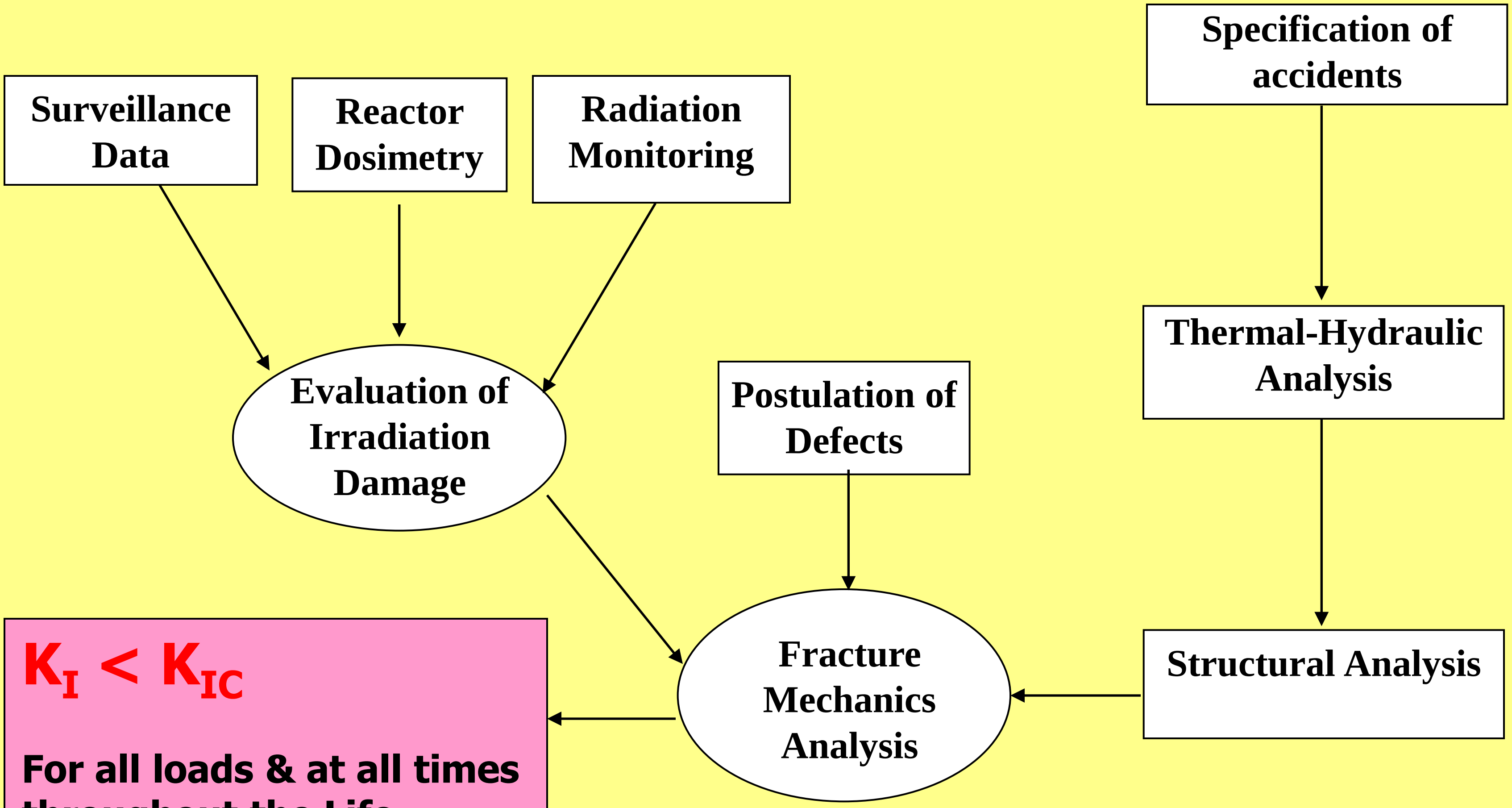


A Crack-like Flaw near the Inner Surface of RPV wall will be subjected to Large Opening Stresses during a Thermal Shock



Can lead to Brittle Fracture under reduced toughness due to irradiation damage

Structural Integrity Assessment of RPV important to PLiM for LTO



$K_I < K_{IC}$

For all loads & at all times throughout the Life

The Approach

Assessment of Irradiation Damage of *Cr-Mo-Ni-V* Steel: *As per Russian Code PNAE*

Toughness Transition Curve for Emergency Conditions:

$$K_{IC} = 65.2 + 18.12.\exp\{0.0293\,(T - T_K)\}$$

T_K (after irradiation damage) = T_{K0} (initial) + δT_F (shift due to fluence)

where,

$\delta T_F = (AF)\,(\Phi/\Phi_0)^{1/3}$,
A_F = a constant depending on the composition of RPV steel,
Φ = fluence ($E \geq 0.5$ MeV) in n/m^2, and
Φ_0 = reference fluence = 10^{22} n/m^2

Structural Integrity Assessment of VVER-1000 RPV

Results of Integrity Evaluation of RPV under following Postulated LOCA Events are Presented in this Paper:

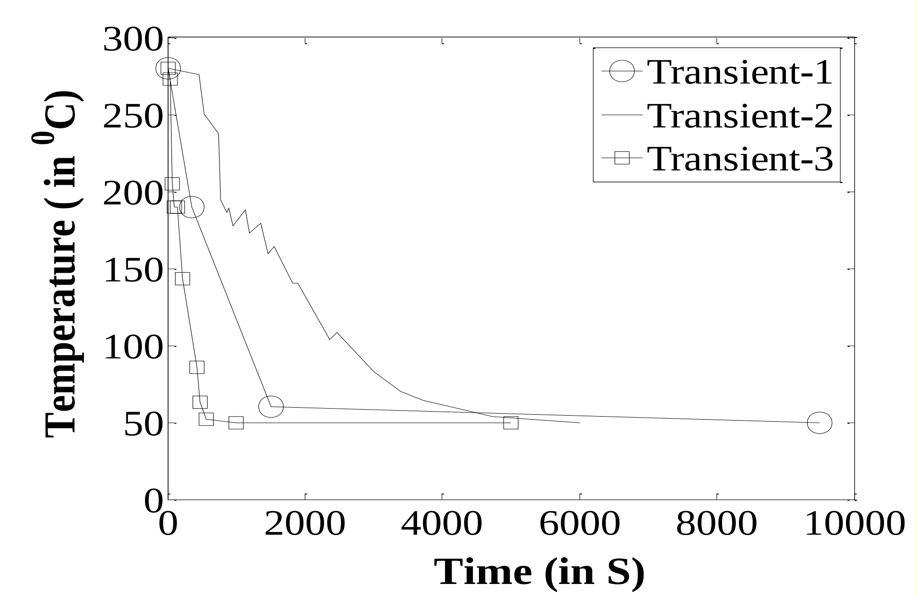
→**Inadvertent opening of the pressurizer safety valve** followed by its failure to seat.

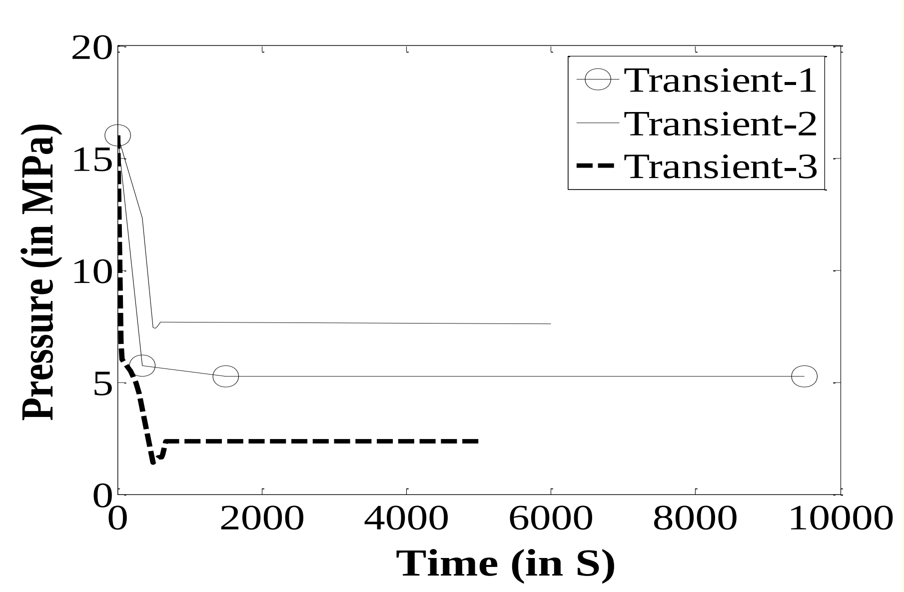
→**Medium Break LOCA:** Leak of diameter 25 mm from the main coolant pipe with four ECCS channels under operation.

→**Large Break LOCA:** Leak of diameter 100 mm from the main coolant pipe with four ECCS channels under operation.

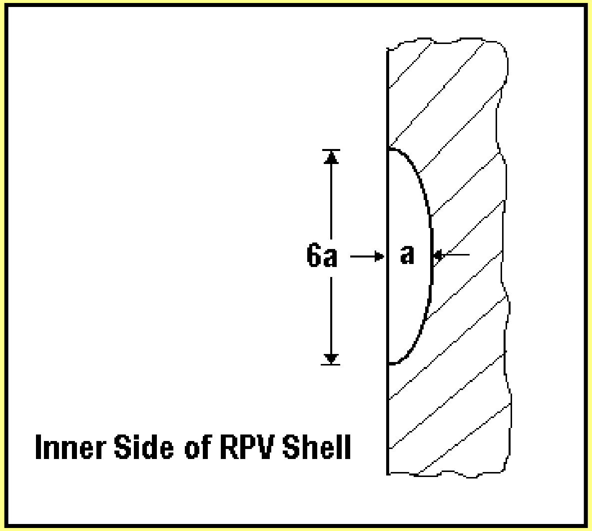
Temperature & Pressure time histories on RPV Core Belt Region due to the Postulated events

Obtained From Thermal Hydraulics Calculations:





Structural Integrity Evaluation: Procedure



Crack Size, a =

10 mm to 48 mm

Where,

48 = 25% of RPV Wall thickness (as per ASME Code guidelines)

Postulation of Cracks of various sizes

(As per ASME)

↓

Thermal Analysis (FEM)

↓

Stress Analysis (FEM)

↓

Fracture Analysis (1st Principles)

↓

Severity Assessment of Postulated Cracks

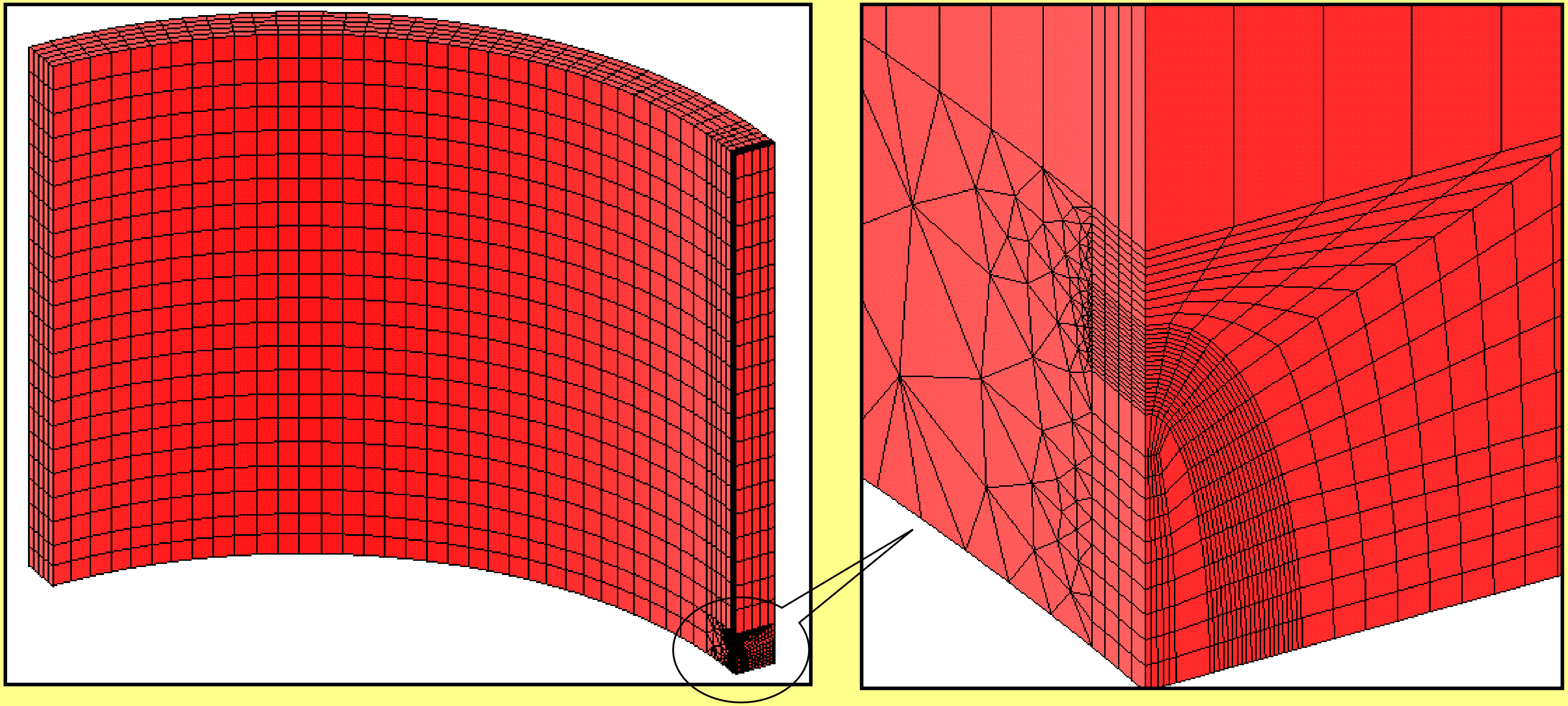
↓

Safety Margins Available

Procedure Highlights

- ✓ **Fracture Toughness & Irradiation damage assessment as per PNAE (As Material is Russian Grade)**
- ✓ **Postulation of Cracks as per ASME (More Justifiable, as will be shown later)**
- ✓ **Thermal & Stress Analysis using FEM (Code methods approx. for complex loads)**
- ✓ **Fracture Mechanics Evaluation directly from 1st Principles (Code formulae approx. for complex loads)**
- ✓ **Austenitic SS Clad not considered for Conservatism**
- ✓ **Procedure Validated through a no. of Benchmarking Exercises**

FE Modeling of Core Belt Region with Postulated Crack



- Symmetrical 1/8th geometry is modeled
- SS Clad not modeled for conservatism
- Crack is included in the FE model for direct evaluation of K_I
- 20 noded iso-parametric 3D elements are used
- Quarter point singularity elements near the crack
- ~ 14000 elements (typical)
- ~ 45000 nodes (typical)

Material Properties for Thermal & Stress Analyses

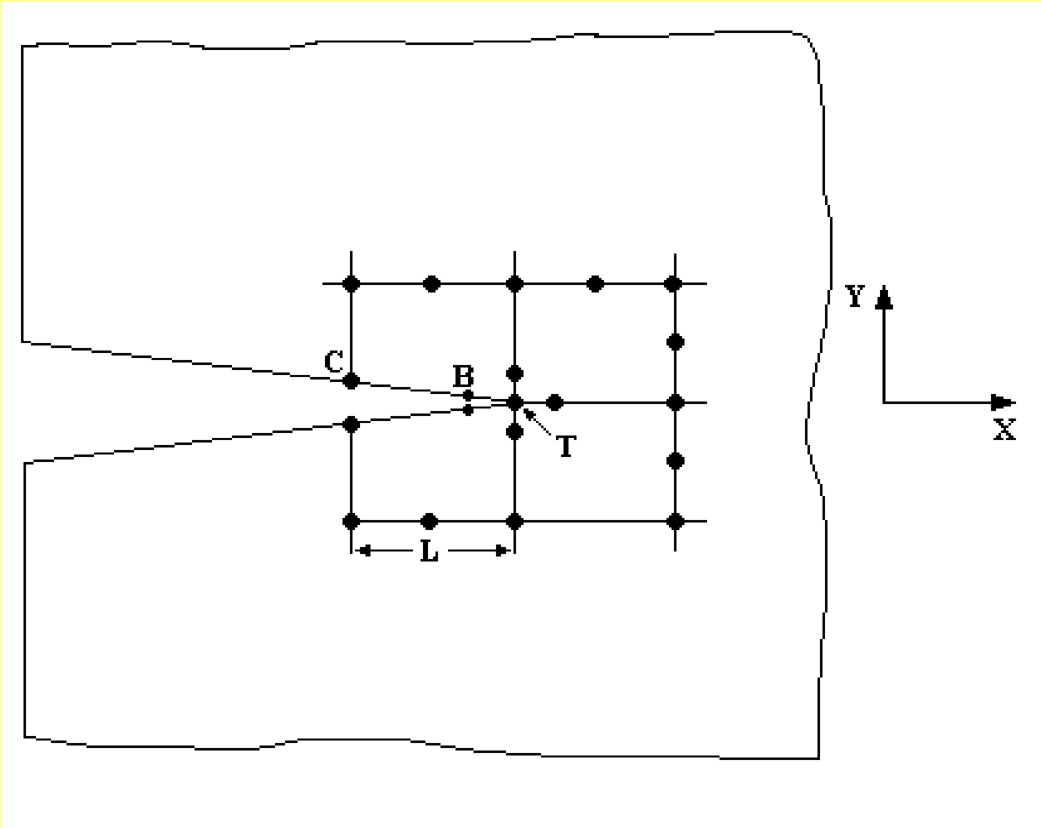
T	E	α	α_0	ν	λ	c_p	ρ
[°C]	[10^3 MPa]	[10^{-6} K^{-1}]	[10^{-6} K^{-1}]	[1]	[Wm^{-1}K^{-1}]	[Jkg^{-1}K^{-1}]	[kgm^{-3}]
20	208		12.5	0.3	35.0	446.9	7830
50				0.3	35.5	458.9	7822
100	201	11.6	12.9	0.3	36.1	478.8	7809
150				0.3	36.6	499.7	7795
200	193	12.0	13.6	0.3	36.8	520.4	7780
250				0.3	36.6	541.2	7765
300	183	12.6	14.2	0.3	36.2	562.0	7750
350	177.5			0.3	35.6	584.6	7733

Ref: IAEA-EBP-WWER-08 (Rev.1) January 2006

Convective Heat Transfer Coefficient at the outer surface of RPV calculated from empirical Handbook solutions

$$h_{Laminar} = \begin{cases} 0.65\ \mathrm{W/m^2{\cdot}K} - \text{for steady state analysis} \\ 0.325\ \mathrm{W/m^2{\cdot}K} - \text{for transient analysis} \end{cases}$$

Fracture Mechanics Analysis: Numerical Evaluation of K_I from 1st Principles



$$K_I = \frac{E}{(1+\upsilon).(1+\kappa)} . \sqrt{2\pi} \left(\frac{4v_B - v_c}{\sqrt{L}} \right)$$

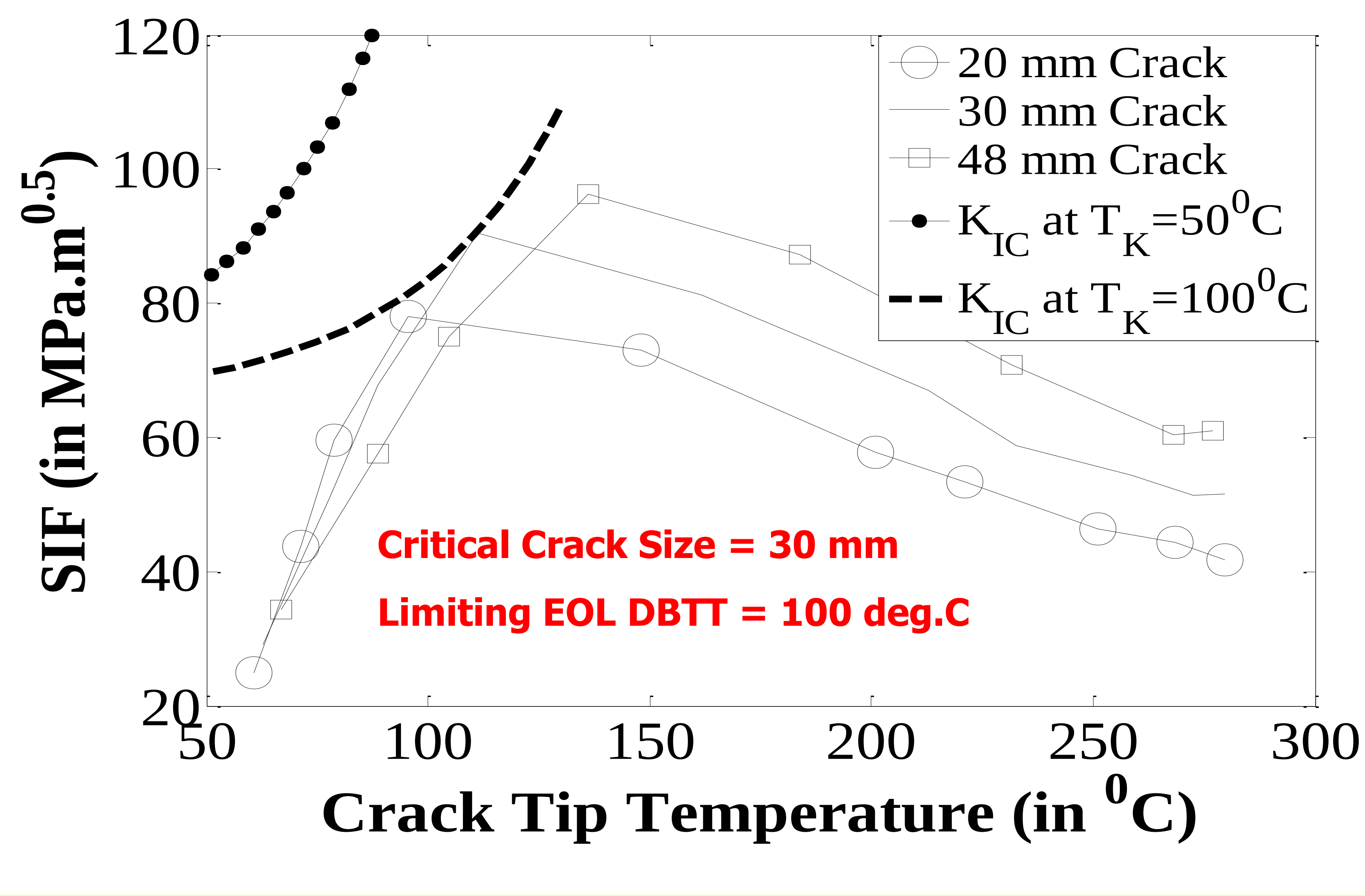
where,

V_B & V_C are the displacements of the nodes B and C respectively in the Y-direction (crack opening direction)

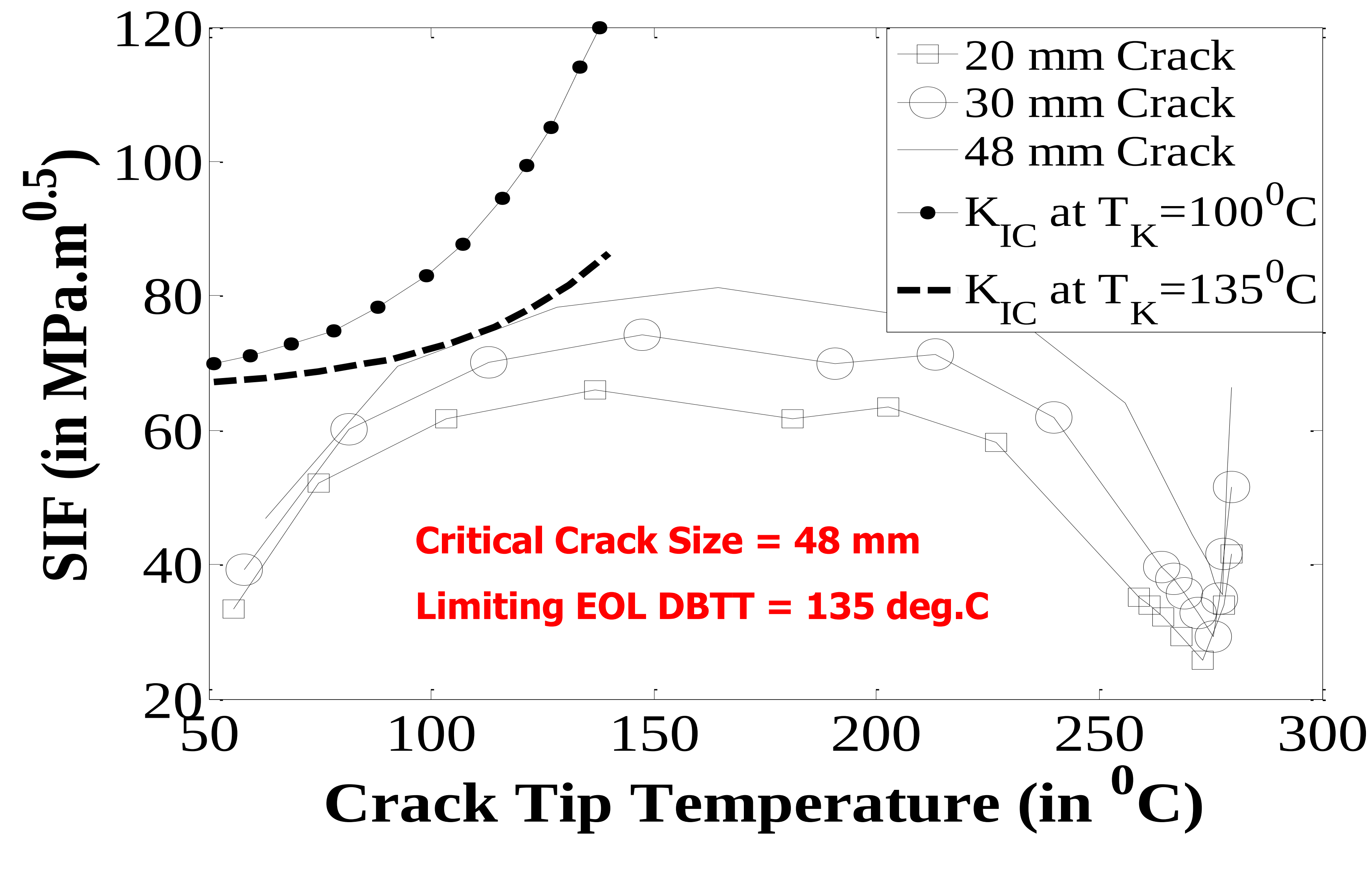
$k = 3 - 4v$ for Plane-Strain

Results of Integrity Evaluation of VVER-1000 RPV

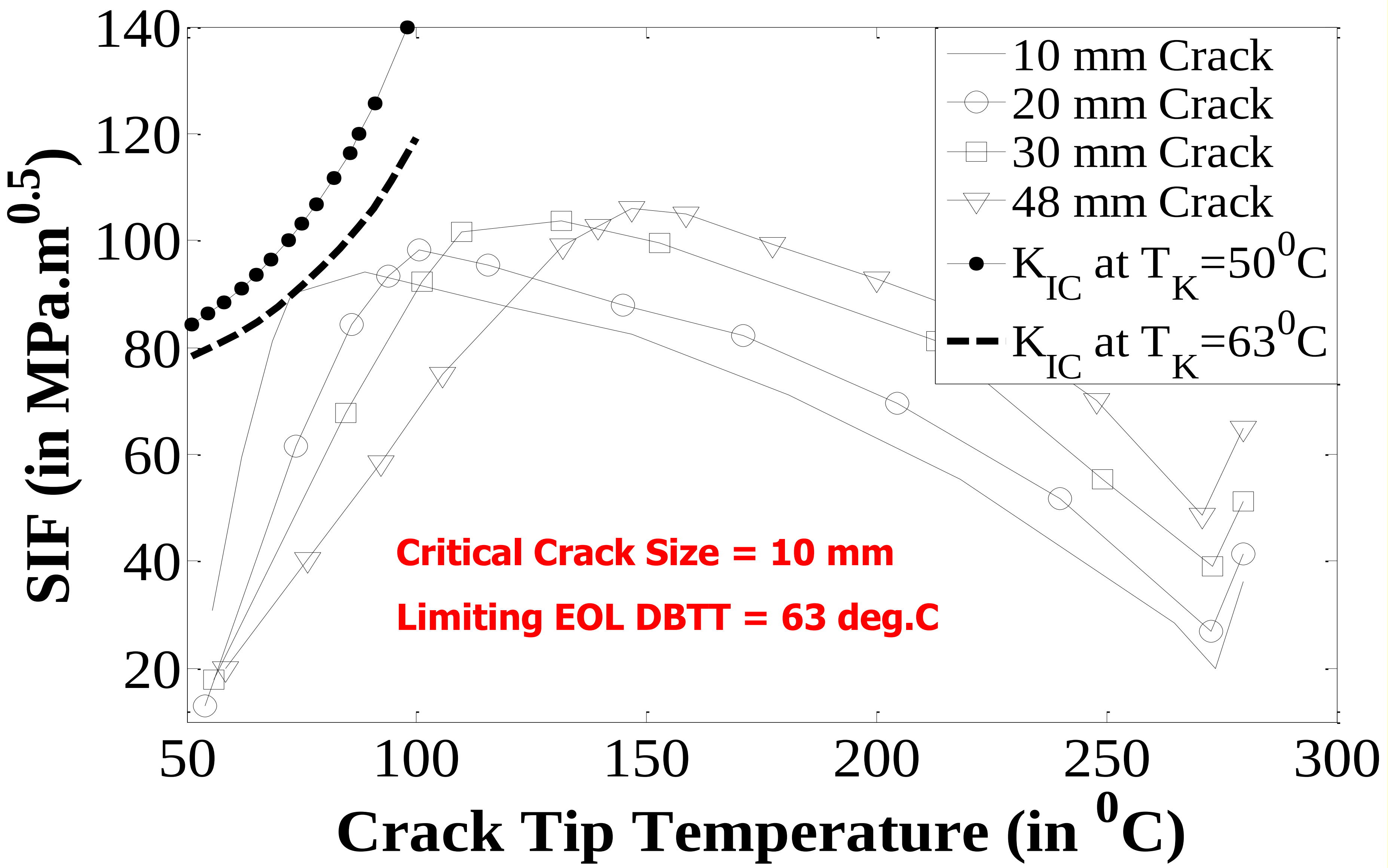
Fracture Assessment Diagram for Transient-1



Fracture Assessment Diagram for Transient-2



Fracture Assessment Diagram for Transient-3



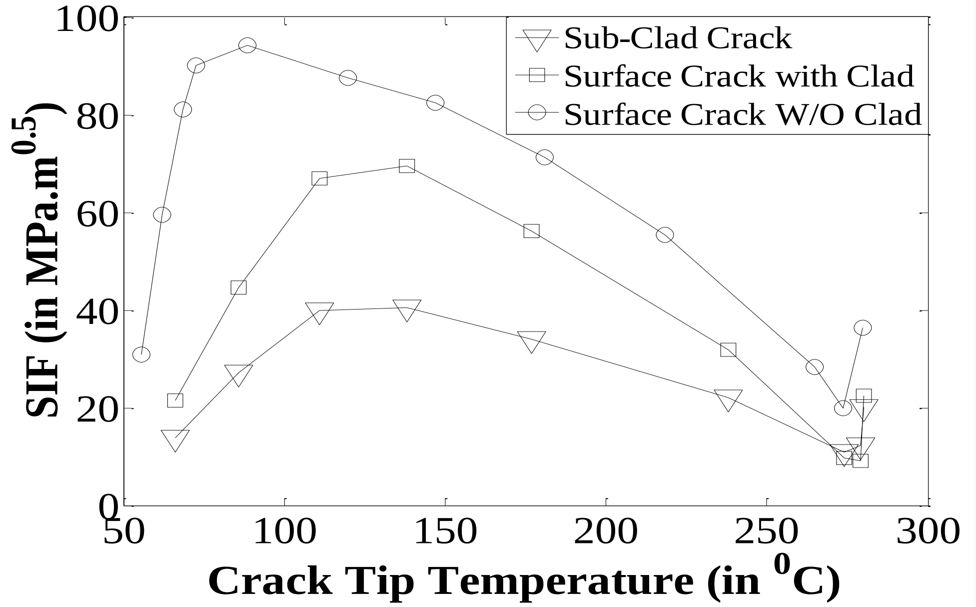
Parametric Studies

→ Effect of Austenitic SS Clad

→ Effect of Crack Aspect Ratio

→ Effect of Uncertainty in heat transfer coefficient, h

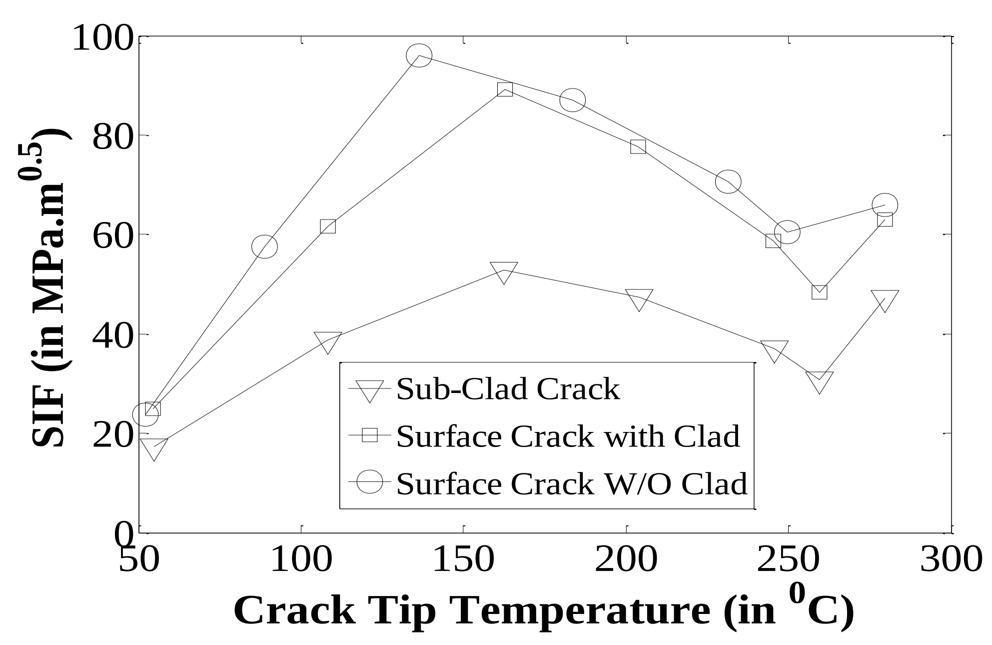
Effect of Austenitic Stainless Steel Clad



9 mm thick SS Clad included in the FE Model

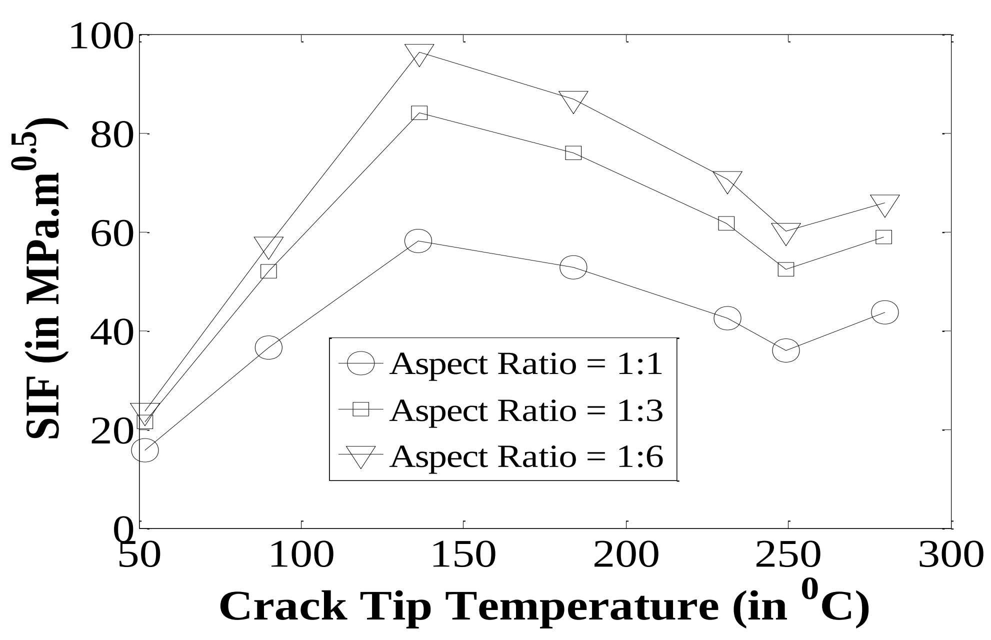
For 10 mm Crack under Transient-3

Clad Plays a Significant Beneficial Role in reducing the Severity of Surface Cracks



For 48 mm Crack under Transient-1

Effect of Crack Aspect Ratio (48 mm Crack Under Transient-1)

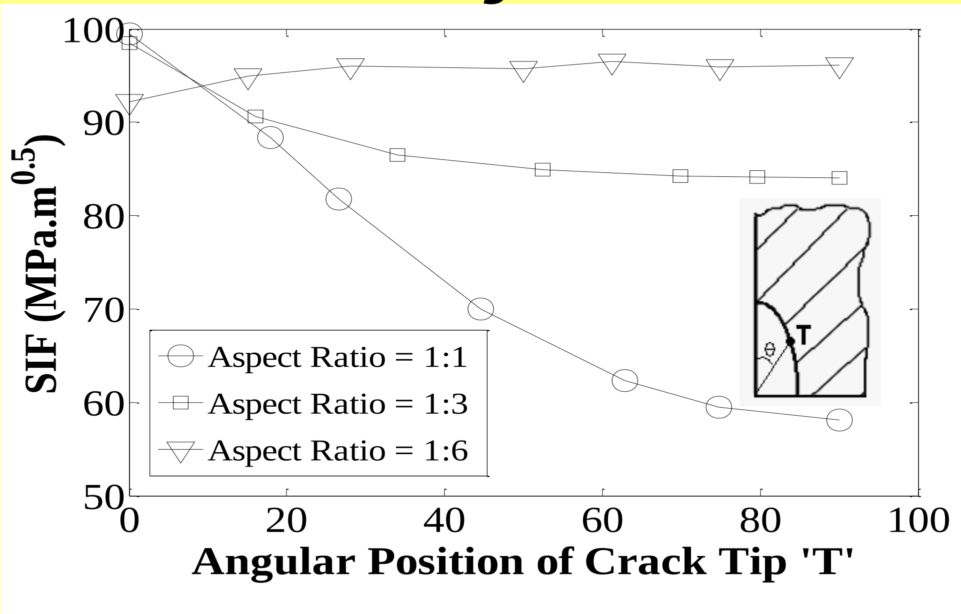


SIF at the Deepest Crack Tip

ASME Recommends 1:6

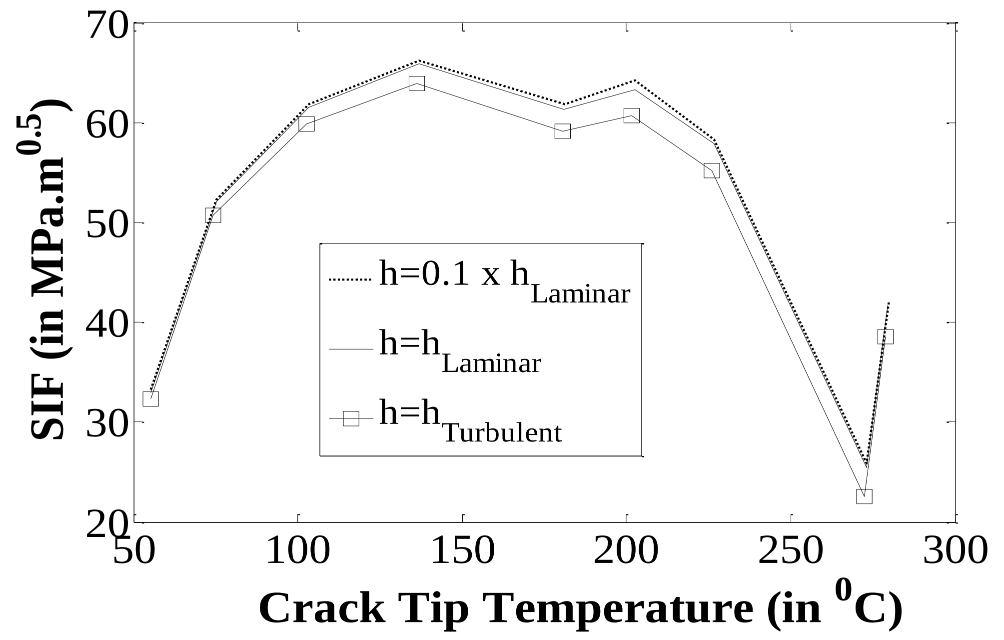
PNAE Recommends 1:3

SIF variation along the Crack at 1500 S



Aspect Ratio 1:6 as recommended by ASME most resembles natural defects & produces most severe crack driving force

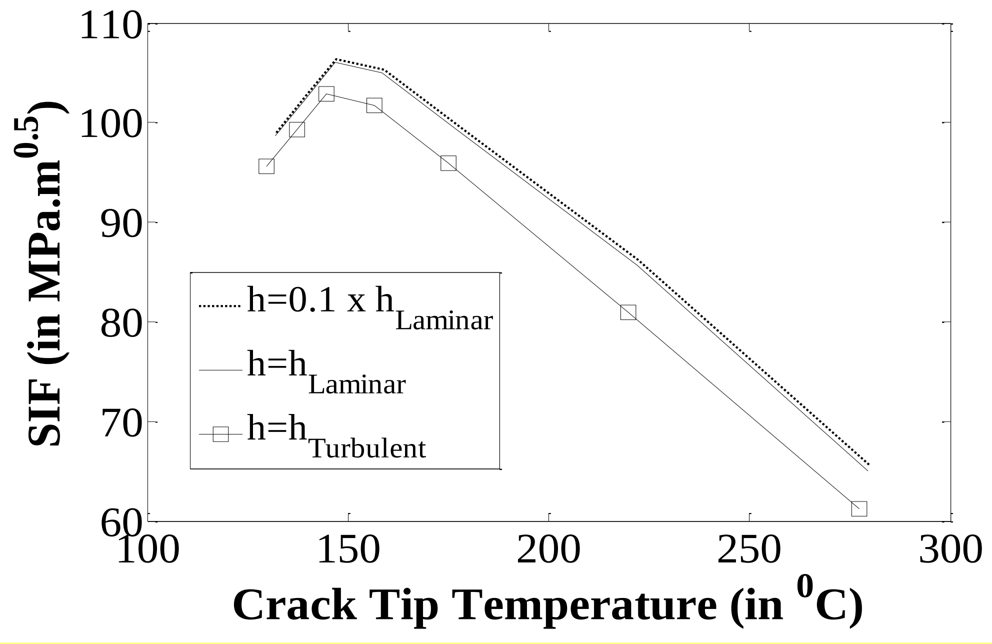
Sensitivity of Integrity Evaluation to heat transfer coefficient, h



For 20 mm Crack under Transient-2

$$h_{Turbulent} = \begin{cases} 8.0\ W/m^2{\cdot}K \text{ – for steady state} \\ 4.0\ W/m^2{\cdot}K \text{ – for transient} \end{cases}$$

For 48 mm Crack under Transient-3



Max. error = 3.5% only

Accuracy of integrity evaluation is insensitive to h

Conclusion

✓ Structural integrity evaluations for VVER-1000 RPV core belt region are presented for severe accidental cool down transients.

✓ A large break LOCA from the main coolant pipe found to be the most severe scenario with the limiting end-of-life DBTT of only 63 ^{0}C with a critical crack size of only 10 mm. This corresponds to 12.5 FPYs of operation, assuming initial DBTT of 0 ^{0}C as per the Material Specification of the RPV.

Inferences drawn for ensuring LTO:

✓ The *emergency operation procedures,* must be timely and correctly followed to cope with a Large Break LOCA Scenario safely.

✓ Initial DBTT of virgin RPV material should be achieved as low as possible.

✓ ISI Program is very important for VVER-1000 RPV and should be reviewed from time to time based on actual operational experiences.

✓ Proper Surveillance Program must be in place as the actual surveillance data will be very helpful in reducing the conservatism in the initial safety assessment and hence extending the life.

✓ Based on operational experience, it may be decided to keep the ECCS storage water at a higher temperature during the extended life of the NPP.

Conclusion (Contd..)

- ✓ The austenitic clad plays a significant beneficial role in reducing the severity of cracks for brittle fracture.

- ✓ Therefore, while ignoring the clad is satisfactory for initial conservative safety assessment of NPPs, it must be considered in integrity evaluations carried out at later stages for residual life estimation/extension.

- ✓ Semi-elliptical crack with aspect ratio 1:6 as recommended by the ASME Code most resembles natural cracks and also produces more severe crack driving force, and hence is more justified for integrity evaluations.

- ✓ The accuracy of the RPV integrity evaluations is practically independent of that of the convective heat transfer coefficient.

THANK YOU