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Design Criteria for Fracture Assessment of Pressurized Nuclear Components

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ABSTRACT

This paper presents the design criteria adopted for fracture assessment of pressurized components of nuclear power plants. Although there are wide variety of components in a typical nuclear power plant, the thrust in this paper is on components which are part of the primary heat transport system. The paper presents an overview of design rules, practices adopted and experimental verification needed for ensuring the structural integrity of nuclear pressure vessels and piping.

INTRODUCTION

Prevention of failure of any structure has always been the main aim of the designer. The common causes of failure are inappropriate material selection, deficiency in the design, inadequate non-destructive examination, manufacturing deficiencies *etc*. Even if these causes are prevented then failures can still occur during service due to improper and insufficient maintenance, operational overloading, time-dependent failure mechanisms such as corrosion, fatigue *etc*. All these factors directly or indirectly lead to formation of defects in the component leading to fatigue or fracture failure. Adequate margins should be built into the design to allow for the effects of time-dependent degradation.

Fracture basically occurs in the presence of defects under tensile stresses, if the material fracture resistance is poor. Crack may also initiate during the operation due to fatigue, stress corrosion *etc*. The fracture resistance reduces significantly due to ageing under certain environmental conditions. Certain grades of cast stainless steels such as, grades CF-8, CF-8M and CF-8A, have austenitic-ferritic microstructures which, degrade by thermal embrittlement. Thermal embrittlement of the base metal results in a slow loss of fracture toughness over extended periods of time and is influenced by coolant temperature and corresponding exposure time. The loss of material toughness is caused by the formation of an alpha-prime phase in the ferrite.

Extensive research and development effort has been made to understand the potential failure modes of pressure vessels and piping components. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B and PV) design code puts appropriate limits to prevent these failure modes. However, this requires detailed stress analysis for all the possible loads and load combinations. As far as components of nuclear power plants are concerned, there are additional requirements to be satisfied since their failure may lead to radiation hazard. The most important components are the ones that form the boundary of the primary heat transport (PHT) system. These components operate at high temperature but below creep range and they contain very high enthalpy fluid. The breach of primary boundary may lead to over-pressurization of the containment, which is the ultimate barrier between the reactor and the environment.

The essence of providing fracture resistance is to minimize the chances of defects getting initiated, and if they initiate, then minimizing their growth, and even if they grow, then adequate margin should be made available against unstable fracture failure. The basic inputs required are:

a) accurate pre-service non destructive examination (NDE) and in-service inspection (ISI) data. On-line fatigue monitoring systems, may minimise ISI data collection.

b) material data related to

1. impact toughness such as charpy V-notch (CVN) values.
2. reference temperature for nil ductility transition (RT_{NDT}) and its variation with high energy neutron fluence.
3. fatigue such as low cycle fatigue (LCF), high cycle fatigue (HCF) and fatigue crack growth rate (FCGR) curves.
4. fracture toughness, such as K_{IC}, J_{IC}, J_i, *J-R*esistance curves under static, cyclic and dynamic conditions.

c) effect of component geometry, stress analysis and categorization of stresses.

d) magnitude of different loads and their cyclic variation.

The first part of the paper would deal with the design rules for fracture prevention, as per standard design codes such as ASME B and PV design code. The later part would cover the recent work being done at Bhabha Atomic Research Centre (BARC) in the area of fracture assessment and leak-before-break verification.

Design Rules for Fracture Prevention

The rules which are followed at the design stage are covered in section III of ASME B and PV code. During operation, if cracks are detected due to fatigue or other damage mechanisms, then Section XI rules are followed. Section XI has provided a set of flaw evaluation criteria, which permits the plant utilities to evaluate the defect with regards to its effect on component acceptability for continued operation in service.

Criteria for Material Selection and Examination

Before applying the ASME design criteria, the material, fabrication and operating conditions should satisfy certain requirements. These are briefly listed below:

a) The materials used in nuclear components, are generally very ductile and have adequate impact toughness. All the ferritic materials shall possess 50 ft-lb (69

Joules) CVN value and 35 mils lateral expansion at a RT_{NDT} + 60 °F for pressure vessels and RT_{NDT} + 100 °F for piping material.

b) Material must remain in the upper shelf regime during all operating conditions.

c) In nuclear reactors, the neutron fluence tends to increase the reference temperature for nil ductility transition (RT_{NDT}). Due consideration should be given to limit certain trace elements in ferritic steel responsible for higher shift in RT_{NDT}.

d) It should be established that austenitic stainless steels do not undergo stress corrosion cracking. Stress corrosion cracking is slow growth of cracks in materials, which experience, simultaneously, tensile stresses and an aggressive environment. Stress corrosion cracking is a problem in many areas of modern technology and has often been the reason for major failures. Thus, it is extremely important to recognise the problem of stress corrosion cracking and to investigate all relevant parameters which may affect such cracking.

e) 100% volumetric examination of starting stock for fabrication and rigorous non-destructive examination during manufacturing stage. The typical target of allowable flaw depth is about 5% of nominal thickness. Sometimes designer may enforce even tighter limits.

f) Adequate weld procedure specifications and qualification.

Design Criteria for Prevention of Fatigue Crack Initiation

Metal fatigue is one of the important degradation mechanisms addressed in the design and operational maintenance of nuclear power plant equipment. For reactor coolant systems comprising of pressure vessels and piping, the section III of ASME B and PV code prescribes a detailed fatigue evaluation procedure. The ASME code committee is implementing a program to develop improved cumulative fatigue design curves and criteria which would include environmental and ageing effects. Following activities are in progress [1]:

- revising the design fatigue curves to include the updated fatigue data.
- extension of carbon, low alloy and high strength steel fatigue curves beyond 10^6 cycles.
- inclusion of reactor water environment effects.
- further development of fatigue design curve for ferritic and austenitic weldments.
- development of fatigue life evaluation curve which includes ageing.
- development of improved analytical procedures for performing fatigue analysis.

However, as per the existing procedure the basic steps are:

(i) *Prevention of gross fatigue damage*: The stresses in nominal regions (away from discontinuities like openings/nozzles, thickness change, welds profile *etc.*) and nominal stresses at discontinuities shall be limited to prevent gross fatigue damage. The limit is as follows :

$$P_m + P_b + Q \leq 3\, S_m \qquad \dots (1)$$

where P_m = primary membrane stress
P_b = primary bending stress
Q = nominal secondary stress
S_m = design stress intensity
= 2/3 of S_y or 1/3 of S_{ult}, whichever is less
S_y = yield strength of the material
S_{ult} = ultimate tensile strength of the material.

The primary stresses are produced due to mechanical loads and secondary stresses are generated due to thermal loads. P_m and P_b are the nominal stresses. The sum (P_m+P_b+Q) at any point represents the total nominal stress at that point. The limit $3S_m$ ($\equiv 2S_y$) ensures *shake down* to elastic action. If this limit is exceeded then each load cycle will lead to plastic cycling.

The thickness of components shall be such that, under the design loads, the mechanical stresses shall be limited to prevent failure by plastic collapse. The stress limits are

$$P_m \leq 1.0\, S_m \qquad \ldots (2a)$$

and

$$P_m + P_b \leq 1.5\, S_m \qquad \ldots (2b)$$

Also, the possibility of non-ductile fracture, at the end of life, in presence of a postulated crack, shall be ruled out. This is assured by postulating a part-through-thickness crack in the direction perpendicular to the direction of maximum stress. The crack depth is taken at least equal to one-fourth of component thickness and its length as six times the crack depth. It has to be shown that

$$2\, K_{IM} + K_{IT} \leq K_{Ia} \qquad \ldots (3)$$

where, K_{Ia} is material fracture toughness which is lower of static, dynamic or arrest condition, K_{IM} and K_{IT} are stress intensity factors corresponding to mechanical and thermal loads respectively. The K_{Ia}, should account for temperature effect and the end-of-life neutron fluence.

(ii) *Prevention of plastic cyclic ratcheting*: Ratcheting is a progressive incremental inelastic deformation due to variation of mechanical stress or thermal stress or both. Under certain combination of steady state and cyclic loading there is a possibility of large distortions as a result of ratchet action, that is, the deformation increases by nearly equal amount for each cycle. Therefore, plastic cyclic ratcheting has to be prevented. The limits are

$$\Delta\sigma_{th} \leq (S_y)^2/P_m \qquad \text{for } 0 \leq P_m/S_y < 0.5 \qquad \ldots (4a)$$

$$\Delta\sigma_{th} \leq 4\,(1-P_m/S_y) \qquad \text{for } 0.5 \leq P_m/S_y < 1.0 \qquad \ldots (4b)$$

where, $\Delta\sigma_{th}$ is maximum allowable thermal stress range in presence of P_m.

(iii) *Margin in fatigue life design*: The fatigue life in terms of crack initiation is kept sufficiently large to preclude any chances of formation and propagation of cracks. ASME design fatigue curve (*S-N*), which is used for predicting crack initiation, incorporate a factor of 2 on stress amplitude and 20 on number of cycles, whichever is conservative at a point. The fatigue damage will start at stress concentration sites or where the stress triaxiality is high. There are different types of load cycles anticipated during the design life of components. Table 1 shows types and number of load cycles anticipated in design life of a typical Indian nuclear power plant.

Table 1: Types of Load Cycles in Typical 500 MWe Indian PHWR Plants (this is a partial list given to illustrate the type cyclic loads)

Sl.	*Type of Cycle*	*No. of Cycles*	*Remarks*
1.	Operation at 100% power	0	continuous operation
2.	Heat-up from cold shutdown (CSD) to hot stand-by (HSB)	1000	planned reactor start-up
3.	Start-up from HSB to 100% power	4000	planned reactor start-up
4.	Shut-down from 100% power to HSB	750	planned for remedy
5.	Cool-down from HSB to CSD	750	planned for remedy
6.	Power manoeuvring at 100 % power	15000	manual/automatic
7.	Reactor trip from 100 % power	1000	protective action
8.	Pump trips at 100 % power	250	due to power failure

Let there be m types of cycles. Each type of load cycle is analysed in order to evaluate the range of stress ($\Delta\sigma$), S_{alt} ($=1/2\ \Delta\sigma$), σ_{mean} *etc*. Let n_1 be number of first type of cycles, S_{alt1} be stress amplitude for first type of cycles, and so on to m number of cycles. Then, for n_1 cycles

$$S_{alt1} = K_{ei} \cdot K_t (P_m + P_b + Q)/2 \qquad \ldots (5)$$

where P_m+P_b+Q = total nominal stress at a point

K_{ei} = elastic-plastic correction factor

$= 1.0$ if $P_m+P_b+Q \leq 3S_m$

> 1.0 if $P_m+P_b+Q > 3S_m$ but $P_m+P_b+(Q)_{av} < 3S_m$

$(Q)_{av}$ = average secondary stress

K_t = theoretical stress concentration factor for the given notch geometry.

For each S_{alt1}, S_{alt2}, ... S_{altm}, the corresponding allowable number of cycles N_1, N_2, ... N_m, are determined from the design fatigue curve. The damage can be determined using linear cumulative damage rule as a cumulative damage factor (CDF) given as

$$CDF = n_1/N_1+n_2/N_2+ \ldots n_m/N_m \leq 1 \qquad \ldots (6)$$

The CDF is also called *usage factor* and must be less than or equal to 1.0.

It must be noted that if, at any location, $P_m+P_b+(Q)_{av} > 3S_m$ then design modification is required. In case the number of cycles are considerably large, may be greater than 1×10^6, then the fatigue life is governed by endurance limit (S_e). In that case the following condition has to be satisfied:

$$S_{alt} < S_e/2 \qquad \ldots (7)$$

Flaw Assessment Procedures

In order to ensure the operational integrity of nuclear power plants, the utility owners regularly carry out the in-service inspection (ISI). ISI is conducted as per ASME code specified intervals. The aim of ISI is to detect defects/cracks developed during operation. Although the resistance against crack initiation is ensured at the design stage itself, cracks can still form due to fatigue or stress corrosion cracking.

If cracks are not detected then components are put back into service till next scheduled of ISI. If cracks are detected then provisions of ASME B&PV Code Section XI are to be followed. The crack growth is sub-critical in nature and is characterised by fatigue crack growth rate (FCGR) models such as the Paris law, eq.8, or its modified form

$$da/dN = C\,(\Delta K)^m \qquad \ldots (8)$$

where, da/dN is crack growth rate, ΔK is range of stress intensity factor and C and m are material constants. In the presence of cracks following steps are to be followed:

(i) *Flaw characterisation*: This procedure translates the indications of ultrasonic examination, during ISI, into simple shapes amenable to analysis. The flaw is assumed to be of semi-elliptical in shape with a as crack depth and l as crack length. The characterisation is done such that $a/l < 0.5$.

As an example, characterisation for some of the detected crack sizes is shown in Fig.1. The characterised flaw depth and length is treated as initial crack depth and length if certain conditions are satisfied. These conditions are given below:

a) Crack depth corresponding to threshold value of stress intensity factor (ΔK_{th}). If the stress intensity factor is less than ΔK_{th} then fatigue crack behaves as non-propagating crack.

b) Maximum crack depth, which cannot be detected by NDE.

c) Critical crack depth corresponding to hydrostatic testing conditions.

If characterised crack depth does not satisfy the above conditions then the initial crack depth is set equal to the maxima of these conditions and simultaneously satisfying the principles illustrated in Fig.1.

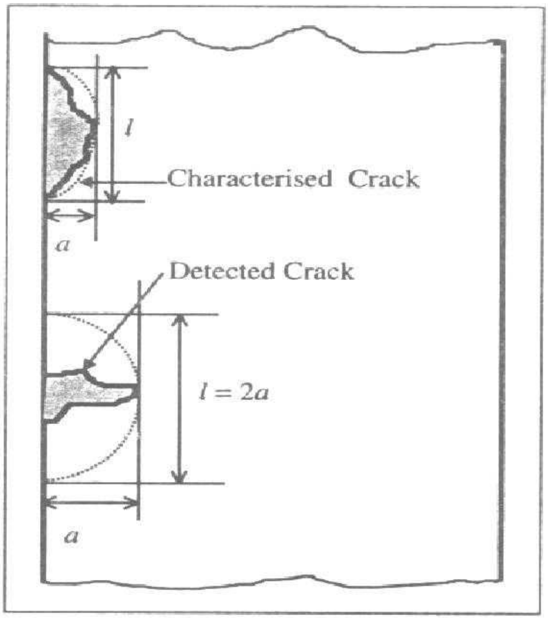

Fig.1: Crack Characterisation Procedure (as per ASME Code)

(ii) *Determination of stresses*: Determine the stresses at the location of observed crack (assuming that crack is not present) for different conditions such as normal conditions, accident conditions *etc*. The stresses can be determined analytically or using finite element method.

(iii) *Fatigue crack growth evaluation*: Fatigue crack growth evaluation is performed using suitable FCGR model, like Paris law. The Paris law is given by eq.8, which is integrated numerically and final crack growth a_f (till next scheduled ISI) is calculated. Next, the minimum critical crack size under normal conditions, a_c, and under accident conditions, a_i, are evaluated. The flaw growth after each cycle is assumed to be concentric, that is, a/l ratio during the growth is constant. Sometimes this assumption leads to very conservative assessment.

In order to evaluate fatigue crack growth using eq.8, the values of ΔK ($= K_{max} - K_{min}$) are needed, which in turn is function of crack depth a. For semi-elliptical surface cracks the SIF (K_I) can be evaluated using ASME code equation

$$K_I = \sigma_m M_m \sqrt{\pi a/Q} + \sigma_b M_b \sqrt{\pi a/Q} \qquad \dots (9)$$

where, σ_m, σ_b = membrane and bending stress
M_m, M_b = geometry factors for membrane and bending stress
Q = flaw shape parameter
$\cong 1 + 1.464\,(2a/l)^{1.65} - 0.212\,[(\sigma_m+\sigma_b)/S_y]^2$

The stresses at the flaw location are resolved into membrane and bending stresses across the wall thickness and all forms of loading are considered in the analysis.

(iv) *Flaw acceptance criterion for pipes*: From basic material properties and stress analysis following two ratios are determined:

$$K'_r = K_I/K_i \quad \dots (10)$$

$$S'_r = \text{Applied Load/Limit Load} \quad \dots (11)$$

where, K_i is defined as $\sqrt{J_{1c}.E}$, in which J_{1c} is the representative toughness of the given material. ASME also allows the usage of an alternative called J_{1mm}. J_{1mm} is estimated as $J_{1mm} \approx 10$ CVN, where CVN is Charpy energy, in ft-lb and J_{1mm} is in lb/in. ASME has provided fracture toughness of some of the commonly used materials. However, users can also determine fracture toughness by conducting tests on compact tension (CT) or three point bend (TPB) specimens. The flaw acceptance criterion is as follows:

(a) acceptance criteria based on crack size:

The crack is acceptable if

$a_f < 0.1\, a_c$ for normal operating conditions ... (12a)

$a_f < 0.5 a_i$ for accident conditions ... (12b)

(b) acceptance criteria based on stresses:

From eq.10 and eq.11 determine the parameter

$$S_c = K'_r \,/\, S'_r \quad \dots (13)$$

(I) If $S_c > 1.8$ then likely failure mode will be non-ductile fracture. In that case crack is acceptable if:

$K_I < K_i \,/\, \sqrt{10}$ for normal operating conditions ... (14a)

$K_I < K_i \,/\, \sqrt{2}$ for accident conditions ... (14b)

(II) If $S_c < 0.2$ then likely failure mode will be plastic collapse. In that case crack is acceptable if:

$\dfrac{\text{collapse load}}{\text{applied load}} \geq 3$ for normal operating conditions ... (15a)

$\dfrac{\text{collapse load}}{\text{applied load}} \geq 1.5$ for accident conditions ... (15b)

The collapse load, of cracked component, can be determined analytically or using finite element method.

(III) If $0.2 < S_c < 1.8$ then likely failure mode will be ductile tearing fracture. In that case crack is acceptable if:

$$\text{applied load} < \frac{\text{unstable tearing load}}{\sqrt{10}} \quad \text{for normal operating conditions} \qquad \ldots (16a)$$

$$\text{applied load} < \frac{\text{unstable tearing load}}{\sqrt{2}} \quad \text{for accident conditions} \qquad \ldots (16b)$$

The tearing load can be determined from rigorous J-tearing modulus (J-T) approach. However for straight pipes, ASME code recommends the Z-factor approach. It is a simplified method and tearing load can be simply evaluated by modifying collapse load using Z-factors. The equations for calculating Z-factors are provided in the code. This factor is a function of pipe size, crack type and material of the pipe.

The rigorous J-T approach involves detailed elastic-plastic finite element analysis to determine J-Integral under applied loads (J_{app}). The results are plotted in the form of J_{app} v/s Load curves, from which the tearing modulus under applied load (T_{app}) can be calculated. Using a pre-cracked CT specimen, the material resistance curve can be plotted. This curve is plotted in terms of J versus crack growth (Δa) and is called the J_R curve. A typical J_R curve for ductile materials is shown in Fig.2. The J_R curve also helps in determining the material J_i. The material J_i is the value of J_R at which the crack initiates after blunting. Crack extension will be stable up to a certain crack length for ductile materials. Unstable failure will only occur if applied tearing modulus exceeds the material tearing modulus. The criteria can be stated as: $J_{app} > J_i$ and $T_{app} > T_{mat}$.

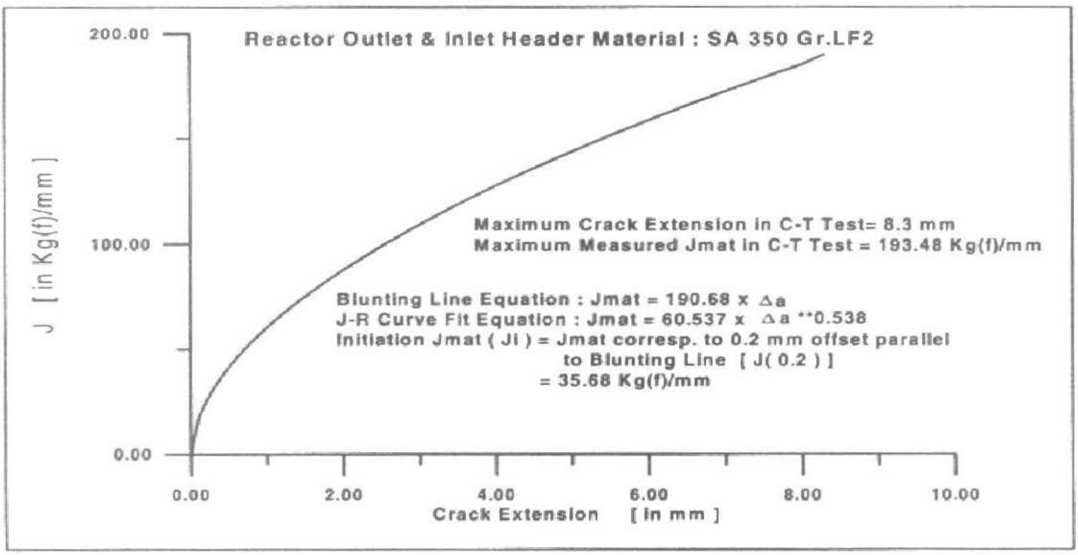

Fig.2: Material J-resistance curve for header material based on test at 250 °C

From the J_R curve, material tearing modulus (T_{mat}) can be calculated for the total range of J. The J versus T_{mat} and T_{app} data are plotted on the same plot, as shown in Fig.3 (right hand figure). The point of their intersection is projected on J_{app} versus load curve (left hand figure) and corresponding unstable tearing load can be evaluated.

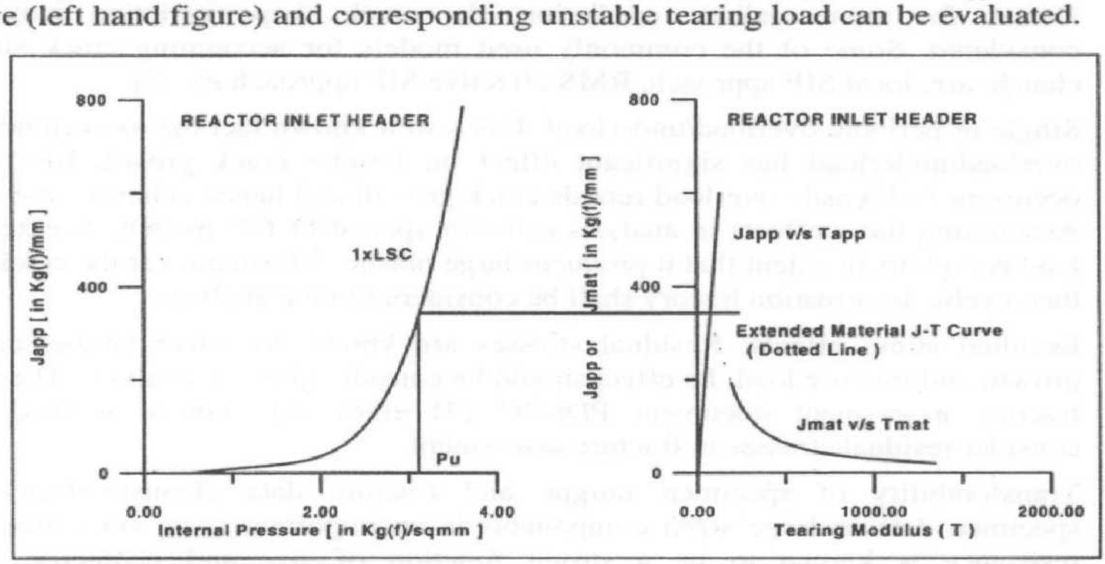

Fig.3: Tearing analysis for evaluation of unstable pressure (P_u) for 1xLSC in reactor inlet header. On the left side is J_{app} v/s internal pressure curve for crack size equal to 1xLSC. On the right side are J_{app} vs T_{app} and J_{mat} v/s T_{mat} curves.

(v) <u>*Flaw acceptance criterion for pressure vessels*</u>: For pressure vessels the ASME flaw acceptance criteria is as follows:

a) acceptance criteria based on crack size: same as eqs.12a and 12b.

b) acceptance criteria based on stress: here the aim is to check for only non-ductile fracture, therefore the flaw acceptance criteria are

$$K_I < K_{Ia} / \sqrt{10} \qquad \text{for normal operating conditions} \quad \ldots (17a)$$

$$K_I < K_{IC} / \sqrt{2} \qquad \text{for accident conditions} \quad \ldots (17b)$$

where, K_{Ia} is the fracture toughness for crack arrest condition for the corresponding crack tip temperature as opposed to K_{IC}, which represents fracture toughness for crack initiation condition at corresponding crack tip temperature.

(vi) <u>*Repair/maintenance or replacement decisions*</u>: If the above stated criterion cannot be met then component is taken up for repair or maintenance. However, if that is also not possible then component might have to be replaced.

Special Issues in Fatigue Growth and Fracture Assessment

Many issues are associated with fatigue crack growth and fracture assessment. Therefore, it is necessary to consider these carefully for safe and economical operation of nuclear power plants. Some of these important issues are:

a) Variation in crack shape: The ASME recommends that, during fatigue crack growth analysis, crack aspect ratio may be assumed to remain constant throughout the growth. Tests have shown that this assumption is far from true. Hence, for more realistic prediction, the crack shape variation may be considered. Some of the commonly used models for accounting crack shape change are, local SIF approach, RMS effective SIF approach *etc* [2].

b) Single or periodic overload/underload: It is a well known fact that occurrence of overload/underload has significant effect on fatigue crack growth life. The occurrence of tensile overload retards crack growth and hence enhances the life. Accounting these effects in analysis calls for special FCGR models. If external load is high to an extent that it produces large plastic deformation at the crack tip then cyclic deformation history shall be considered in the analysis.

c) Residual stress effects: Residual stresses are known to affect fatigue crack growth and fracture load. Its effect should be considered in the analysis. The BSI fracture assessment document PD6493 [3] gives approximate methods to consider residual stresses in fracture assessment.

d) Transferability of specimen fatigue and fracture data: Transferability of specimen data to large sized components is an important issue since fracture resistance is known to be a strong function of size and geometry. The transferability of linear and elastic-plastic fracture mechanics concepts was investigated at MPA [4], with the help of tests on large scale specimens. It has been found by Roos and his co-workers [4] that specimen *J-R* curve can be transformed to the component if multiaxiality coefficient is approximately the same.

e) Effect of cyclic and dynamic loads on fracture resistance: These conditions may degrade fracture resistance. The extent of degradation depends upon the type of steel used.

LEAK-BEFORE-BREAK DESIGN

The rules and criteria adopted for prevention of fracture generally ensure that there is adequate margin against failure. The primary pressure boundary pipe and piping components, such as elbows and tees, are designed and constructed as per the ASME B and PV code. The materials are highly ductile and fabrication processes are very stringent.

However, over and above this the safety regulations require postulation of instantaneous double-ended guillotine breaks (DEGB) in the high energy large sized pipes. The main aim of postulating DEGB is for:

a) design of emergency core cooling system (ECCS).

b) environmental qualification requirements for electrical and mechanical equipment.

c) evaluation of asymmetric blowdown loads on the reactor internals.

d) design of piping and vessel supports under rupture loads.

e) design loads for containment.

f) Evaluation of requirements for pipe-whip restraints and jet impingement barriers.

The postulation of DEGB leads to a large economic penalty on design because the resulting forces on component supports are very high. In addition, one has to provide pipe-whip restraints, jet protection shields *etc.* at all the potential rupture locations. Till mid 1980s DEGB was postulated in all the high energy piping of nuclear power plants to meet the above requirements. At the same time, DEGB was not generally believed to be a realistic event for large diameter pipes / pipe components of nuclear power plants because of following reasons:

a) Usage of ductile materials and existence of adequate margin even under postulated accident events.

b) If a crack develops due to cyclic loads then it will be detected during ISI and corrective action can be taken.

c) Even if a through thickness crack develops during service, then it is likely to produce detectable leaks that would permit the plant operators to safely shut down the plant rather than result in the catastrophic rupture of the pipes / pipe components. In other words, components would *leak before they break.*

The above three facts about nuclear components constitute the three levels of leak-before-break (LBB) assessment. Level 1 is satisfied by stringent specifications in material, design, fabrication, non-destructive examination and pre-service inspection and testing. It is shown at design stage itself that chances of crack initiation are very very remote. The second level is related to the minimisation of crack growth, and third level is related to prevention of catastrophic failure.

In view of the aforementioned facts about inherent resistance of piping components to DEGB, United States Nuclear Regulatory Commission (USNRC), issued a new set of rules in which it was accepted that "DEGB of primary loop piping, of pressurised water reactor (PWR) is unlikely to occur, provided it could be demonstrated by fracture mechanics analyses that postulated small through-wall flaws in pipes/pipe components would be detected by plant's leakage monitoring systems long before the flaws could grow to unstable sizes" [5].

These rules led to the formal development of acceptance criteria of LBB. The details regarding limitations and acceptance criteria for LBB analyses were later published by USNRC [6,7]. International Atomic Energy Agency (IAEA) has also published guidelines for application of LBB [8,9]. This criteria and the resulting steps are listed below:

1. Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water-hammer or low and high cycle fatigue.

2. Identify the material and the associated plant specific material properties.
3. Calculate the applied loads. Identify the location at which the highest stress occurs.
4. Postulate an inside surface crack at the governing location. Determine fatigue crack growth. Show that a through-wall crack will not result.
5. Postulate a through-wall crack at the governing location. The size of crack should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the piping is subjected to normal operating loads. Demonstrate a margin of 10 between the calculated leak rate and the leak detection capability. This crack size is called LSC.
6. Under accident load (maximum load), demonstrate that there is a margin of at least 2 between the LSC and the critical crack length, and for LSC the margin between faulted load and instability load is at least 1.4 (or 1.0 for absolute combination of loads).

LBB Assessment and Verification Tests

LBB concept is now universally used to design the primary heat transport (PHT) piping of nuclear power plants. The same is being followed to design the next generation Indian 500 MWe pressurised heavy water reactors. However, this requires extensive analysis and experimental verification. Recently BARC has undertaken different projects in the area of fracture assessment and LBB verification programme to resolve some of the issues related to fatigue crack growth and unstable fracture of PHT piping of pressurised heavy water reactors. The main aim is to understand some of the unresolved issues. Some of the important observations is discussed briefly in the following paragraphs.

Fatigue Crack Growth Tests on Straight Pipes and Elbows

In support of the second level of LBB evaluation, a number of fatigue tests were planned on straight pipes, elbows, and also on specimens. With regard to second level of LBB, the most obvious approach is to conduct fatigue crack growth rate tests on standard specimens. However, from the past experience it is known that there may be significant difference between the data obtained from standard specimens tests and component test. This difference is due to stress distribution at the crack tip, which in turn depends on the location of the notch, geometry of the component and type of loading. Therefore fatigue crack growth tests are being conducted on straight pipes and elbows. The tests on 8 inch NB size (219 mm OD and 15.1 mm thickness) pipes and elbows, made of carbon steel grade SA333Gr6, have been performed. The tested components were provided with notches of various depths and lengths and were subjected to cyclic bending moments.

The role of non-destructive examination becomes very important during tests. It is important to characterise the crack shape and size during the entire period of test. Techniques such as UT and ACPD are being employed for this purpose.

Part-through-thickness notches of constant depth and root radius, were machined in the circumferential direction. The studies have been carried out for the notches having notch depth to thickness (a/t) ratio 0.125, 0.25 and 0.375 and notch length to notch depth ($2c/a$) ratio 56, 28 and 18. Fatigue tests have been carried out for load ratio (R) of 0.1 and 0.5. Pipes were subjected to bending moment by applying four point bending loads. Parameters recorded during the fatigue tests are:

i. Number of cycles for crack initiation from machined notch.

ii. Crack depth measurement at several locations along the notch length.

iii. Number of cycles required for crack to penetrate through wall.

iv. Length of the crack at the inside surface, after the crack penetrates through wall.

It has been observed that initiation of crack from machined notch is highly dependent on R and initial notch depth. Effect of notch depth for a given R shows that number of cycles required for crack initiation decreases with increase in initial notch depth. Effect of R for given initial notch depth shows that number of cycles required for crack initiation increases with increase in load ratio. Crack growth has not been observed in length direction for notch dimensions under study. Analytical studies has also been carried out for evaluation of number of cycles required for crack initiation for a given notch dimensions and loading conditions. Fatigue crack growth has been obtained analytically by evaluating SIF at the crack tip and the Paris constants obtained from tests on specimens. For one of the tests on pipe, the comparison between analytical and experimental results is shown in Fig.4. Although there is a reasonable match between the two, there is scope for further improvement, specifically with regard to plasticity correction factor.

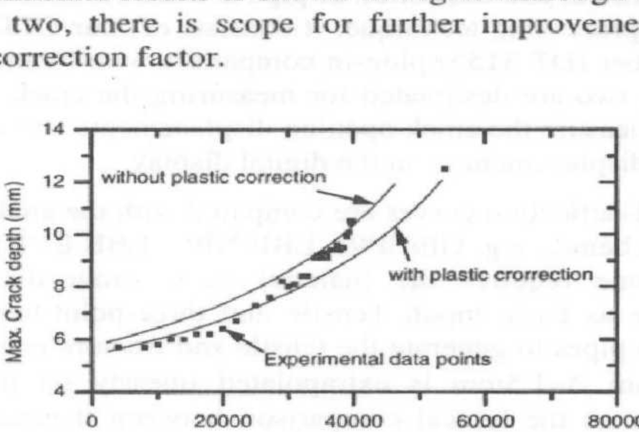

Fig.4: Comparison of experimental and analytical results for 219mm OD pipe of Carbon steel

Fracture Tests on Pipes, Elbows and Specimens

Over the years, material J_R curve has been evaluated by conducting tests on CT or TPB specimens as per ASTM E-813. J_R curve was earlier believed to be a material property, provided certain conditions were met. Thus the J_R curve obtained through the CT or TPB specimens of the same material could as well be applied to determine the instability load of the prototype structure. However, it is increasingly seen that real life structures exhibit fracture resistance which is different from that of specimens. This difference is attributed to the difference in crack-tip constraints of the specimens and components. The transferability of specimen J_R curve to component level for a material is thus an important issue in fracture mechanics. There is another issue related to extrapolation of specimen J_R curve which also requires investigation. Specimen J_R curves generated through the testing of CT or TPB specimens are often limited to very small amount of crack growth (Δa = 2-8 mm). A real life component, on the other hand, often undergoes substantial amount of stable crack growth (Δa = 50-200 mm). The extrapolation of specimen J_R for large amounts of crack growth is also an important issue in the field of fracture mechanics. Keeping these objectives in view, 45 tests are planned on pre-cracked straight pipes and elbows at Structural Engineering Research Centre (SERC), Chennai, India, in order to determine the component J_R curves. The pipes and elbows are of sizes ranging from 8 inch to 16 inch (200 mm to 400 mm) in diameter, and thickness varied from 0.75 inch to 1.5 inch (19 mm to 38 mm). The material of the pipes and elbows is SA333Gr6 carbon steel. TPB, CT and CCP specimens machined from the above pipes and elbows are also tested to generate specimen J_R curves. Test specimens consist of straight pipes with through-wall circumferential cracks at the middle of its length. These pipe specimens are subjected to four point bending load applied under displacement control. The rate of displacement was fixed as 0.055 mm/sec. During the fracture experiments, instrumentation are mounted to measure the total applied load, load line displacement, ACPD measurement at the crack tip, crack opening displacement at various locations of the notch, crack growth and deflection of pipe at different locations. Crack growth is measured by image processing technique. It consists of four CCD cameras connected to a PCI frame grabber (DT 3155) plug-in compatible with Pentium-II computer. Out of the four cameras, two are designated for measuring the crack growth at two crack tips, one is used to measure the crack opening displacements and one for recording the load and load-point displacement from the digital display.

Experimental load-deflection curves are compared with the analytical predictions by various estimation schemes, e.g. GE/EPRI, LBB.NRC, LBB.BCL1 and LBB.BCL2. In these predictions, one requires the material stress-strain diagrams and fracture resistance (J_R) curve as basic input. Tensile and three-point-bend (TPB) specimens were machined from pipes to generate the tensile and fracture properties. Specimen J_R curve with maximum Δ=1.5mm is extrapolated linearly on the J-T space up to Δa=60mm. Fig.5 shows the typical comparison between theoretical predictions and experimental observations. It may be seen that predictions match reasonably well. The difference between predicted and experimental load deflection curve, therefore, indicate the difference in the specimen and component J_R curve. This highlights the importance of transferring the specimen J_R curve to the component level considering the crack tip constraint. Fig. 6 shows the component J_R curves generated for 16 inch (400 mm) diameter pipes with various sizes of throughwall circumferential cracks. The figure also shows the TPB specimen J_R curve which are obtained for crack growth of maximum 2.4 mm.

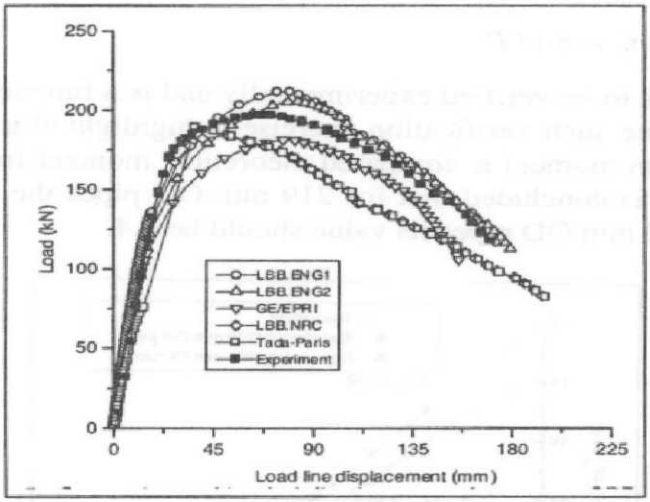

Fig.5: Comparison of load deflection curve for test no.SPBMTV ($2\theta=94^o$) considering component J-R *curve*

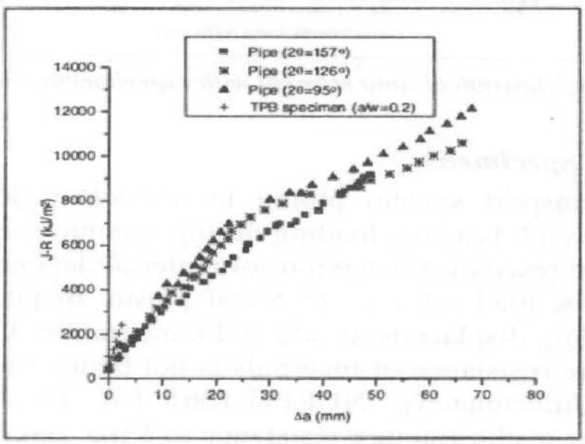

Fig.6: Comparison between component J-R curves of 16 inch diameter straight pipes with various through-wall circumferential cracks and 1T TPB specimen J-R curve

Once the fracture properties are known, then ductile fracture calculations can be done, as explained in step (iv) of flaw assessment procedure (page 151). However, if the fracture toughness is high, as for austenitic stainless steels, then plastic collapse is the governing mode of failure. Hence, the maximum load carrying capacity is lower of fracture load, as evaluated from tearing analysis, or plastic collapse load. The plastic collapse load can be calculated based on flow stress of the material. The flow stress (σ_f) is conventionally defined as average of yield strength (σ_y) and ultimate tensile strength (σ_u). This definition works well only for few types of material. In addition, it also depends upon pipe size and flaw size/shape. In order to generalise this parameter, it may be defined as

$$\sigma_f = (\sigma_y + \sigma_u) / F \qquad \ldots (18)$$

where, the factor F has to be verified experimentally and is a function of material, pipe size and flaw type. One such verification exercise is highlighted in Fig. 7, where the experimental maximum moment is compared theoretical moment for 219 mm OD and 400 mm OD pipes. It is concluded that for 219 mm OD pipes the factor F should be equal to 2, and for 400 mm OD pipes its value should be 2.4.

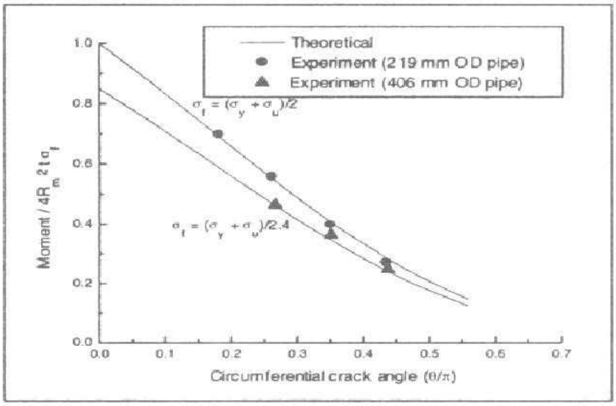

Fig.7: Comparison of theoretical limit moment with experimental maximum moment

Cyclic Tearing Tests on Specimens

The primary heat transport system piping in a nuclear power plant could be subjected to large cyclic and dynamic loading during seismic events. The significance of cyclic load on fracture resistance behaviour of material depends on flaw orientation with respect to pipe axis, load ratio, incremental plastic displacement between two consecutive cycles, loading displacement rate and temperature. Currently the effect of cyclic loading on fracture resistance of materials is not being considered in the piping integrity assessment. Unfortunately, NUREG-1061 [6], IAEA-TECDOC-710 and 744 [8,9] do not consider cyclic fracture resistance in LBB assessment of high energy piping components. The aim of the present study is to develop fracture properties and analytical procedure to characterise the cyclic elastic-plastic fracture resistance, considering the effects of load ratio, incremental plastic displacement, loading displacement rate and temperature.

In this study experiments has been carried out on compact tension specimens having dimensions conforming to the ASTM standard E1737. The tests have been carried out for base as well as weld material. Cyclic elastic-plastic fracture resistance behaviour has been characterised by cyclic-J parameter. It has been observed that

a) there is no significant effect of cyclic loading on fracture resistance when load ratio is greater than zero. But there is significant reduction in fracture resistance properties of the material subjected to cyclic loading with load ratio less than zero

b) for given incremental plastic displacement, fracture resistance of material decreases with decrease in load ratio.

c) effect of loading displacement rate shows that fracture resistance properties decreases with increase in displacement rate.

d) there is almost no blunting of the crack tip, which is unlike the observation in monotonic fracture loading where large blunting precedes the stable crack growth.

It has also been observed that the cyclic fracture resistance.(J_R curve) is significantly lower than the monotonic resistance for a given test condition.

Estimation Methods for Fracture Assessment

In recent years different estimation methods have been developed for quick analysis for fracture assessment. These methods are based on certain approximations but nevertheless yield the results within acceptable accuracy. Some of these methods are, GE-EPRI technique, modified limit load method, MPA moment method, R-6 method, ASME *Z*-factor approach, LBB_BCL1 method, LBB_BCL2 method, LBB-NRC method *etc*. These techniques have been applied on the piping of Indian 500 MWe PHWRs for LBB qualification. The comparison of the results based on different methods show that they yield nearly similar results. As an example, a comparison of results is shown in Fig.8 for steam generator inlet pipe of 500 MWe Tarapur Atomic Power Project, India. The pipe was assumed to have through-wall circumferential crack at the highly stressed location. The results are compared with respect to those based on the finite element method.

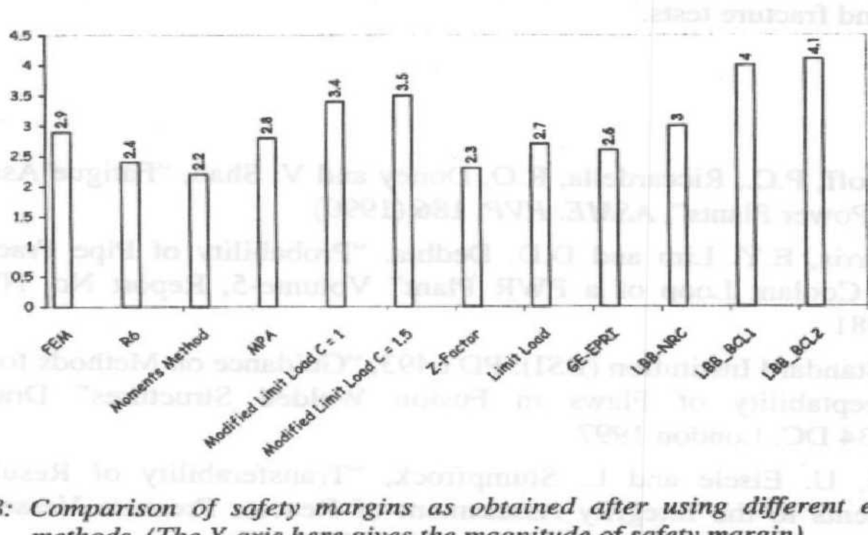

Fig. 8: Comparison of safety margins as obtained after using different estimation methods. (The Y-axis here gives the magnitude of safety margin)

Among all the above methods, the R-6 method is the most popular method. It is based on two criteria approach in which fracture and plastic collapse modes of failure are considered simultaneously and it involves evaluation of two parameters namely K_r and S_r. K_r is the ratio of stress intensity factor to fracture toughness, and S_r is the ratio of load to limit load. High value of K_r signifies nearness to non-ductile fracture, while high S_r signifies nearness to plastic collapse, and intermediate values signify the chances of ductile fracture. Using the R-6 method, one can evaluate the load corresponding to initiation of crack growth and instability.

CONCLUSIONS

The paper gives a brief overview of the requirements for fracture assessment of pressurised nuclear components. The design codes, such as ASME B and PV code, and design practices, such as LBB design, assure that there is adequate margin against sudden failure of the components. However, it requires considerable expertise for the usage of these design rules specifically in the area where there are unresolved issues such as size and geometry effects in fracture and fatigue, effect of residual stresses *etc.* BARC has undertaken a component integrity programme to resolve the above issues. Design also involves large amount of analytical work without which the appropriate failure loads cannot be evaluated.

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REFERENCES

1. S. Leshnoff, P.C., Riccardella, R.O. Doney and V. Shah, "Fatigue Assessment at Nuclear Power Plants", *ASME, PVP*, **186** (1990)
2. D.O. Harris, E.Y. Lim and D.D. Dedhia, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant" Volume-5, Report No. NUREG/CR-2189, 1981
3. British Standard Institution (BSI): PD 6493, "Guidance on Methods for Assessing the Acceptability of Flaws in Fusion Welded Structures" Draft Version 97/714934 DC, London 1997
4. E. Roos, U. Eisele and L. Stumpfrock, "Transferability of Results of PTS Experiments to the Integrity Assessment of Reactor Pressure Vessel", *Nuclear Engineering and Design*, **198** (2000), pp173-183.

5. GDC-4, "Nuclear Regulatory Commission Modification of General Design Criteria: Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures", Federal Register, Volume 52, No. 201, pp. 41288 to 41293, 1987
6. Report No.: NUREG-1061, Vol. 3, "Evaluation of Postulated Pipe Breaks". Report of the U.S. Nuclear Regulatory Commission, Piping Review Committee, 1984
7. SRP-3.6.3, "Leak Before Break Evaluation Procedures (Public Comments Solicited)", Standard Review Plan, Federal Register, Volume 52, No. 167, pp. 32626 to 32633, 1987
8. IAEA-TECDOC 710 "Applicability of the Leak Before Break Concept", 1993
9. IAEA-TECDOC 774 "Guidance for the Application of the Leak Before Break Concept", 1994