Developments in Fracture Mechanics and Non-Destructive Examination

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INTRODUCTION

The purpose of this chapter is to give a brief review of the major developments and trends that have occurred in some of the technical areas covered by Division G of SMIRT since the first conference was held in Berlin in 1971. As with any human activity, time and experience produce organic change so that the technical areas covered by Division G have changed somewhat over the 18 years of its existence. As first conceived by Tom Jaeger, Division G was one of two divisions devoted to Light Water Reactor Components other than fuel and cladding. Division F was conceived with Core Structures and Piping whilst Division G was devoted to the Steel Reactor Pressure Vessels.

The early Conferences devoted much time to Stress Analysis of particular pressure vessel regions, such as nozzles, cylinder/cylinder intersections, vessel heads and flanges. However, from the start there have always been several sessions on fracture mechanics and reactor vessel integrity analysis, a topic which has been one of growth of interest even up to the present Conference.

By 1979 (SMIRT 5), "Failure Assessment" made up all but one of the sessions, and the Division had become a recognized center for discussion of all aspects of fracture mechanics work, attracting papers and discussion from leading experts in the field.

Similar discussions did occur in other divisions, however, and particularly in Division F in relation to Reactor Circuit Primary Piping to which many of the same features applied. This was recognized at SMIRT 6 in 1981 by the first joint session between Divisions G and F under the title Fracture Mechanics of Piping, and by arranging the timetables so that delegates to other Divisions could attend those sessions of Division G that were devoted to the more general aspects of Fracture Mechanics such as methods of toughness measurement and the development of elasto-plastic fracture mechanics treatments.

In 1983 the Chicago SMIRT 7 had no less than 7 joint G and F sessions covering not only the fracture mechanics methods and their applications to both pressure vessels and pipings but also this similar joint interest in stress analysis, thermal loading and fatigue. This situation led to a reconsideration of the terms of reference of the Divisions such that in the later Conferences, Division G has concentrated on failure assessment aspects of all non-fuel steel reactor components, and by the time of the 8th Conference (Brussels, 1985) the title of Division G was changed from "The Structural Analysis of Steel Reactor Pressure Vessels" to "Fracture Mechanics and Inspection", a change maintained in the present Conference. It is to fit in with the present title and terms of reference therefore that this review is directed only to the aspects of Division G coverage, leaving developments in

Stress Analysis and some other topics such as fatigue effects to be dealt with elsewhere.

It must also be noted that Division G has been effective in mounting several pre and post-conference seminars to provide the opportunity for up-to-date and more detailed discussion of selected topics between smaller groups of experts. This also allowed the attendance of experts in related subjects, such as non-destructive testing, who could not be expected to attend the main conference because of the limited content relating to their own expertise. One example of this is the Acoustic Emission Pre-Conference Seminar held in Stuttgart in 1987, whilst the Post-Conference Seminar on Non-Destructive Examination in aid of Structural Integrity has been a feature of each SMIRT meeting from 1979. These seminars have become such an important source of NDE information that it was considered that they must be referred to in this review. A further series of Post-Conference Seminars relating to Structural Integrity of Pressure Boundaries have also a long established feature and have contained much information on materials, stress analysis, fracture analysis and surveillance technique.

It must be remembered that the material presented at these SMIRT Conferences and Seminars reflected the areas where work was in progress because of real technical interest. Sometimes specific national interests led to group papers on one topic like those on Head Design from the UK in 1975 and those on Underclad Cracks from France in 1981. Sometimes they indicated areas of general international interest, such as the progressive growth of papers on Elasto-Plastic Fracture Analysis, on Thermal Shock (PTS) effects and on the Leak Before Break (LBB) concept. This interlinking of the SMIRT papers with the broader trends of development in nuclear technology makes it undesirable to limit this review to referencing individual SMIRT papers. Moreover, such referencing would produce a catalogue which would make uninteresting reading, and it would have considerable duplication due to parallel developments in different countries and progressive statements made at successive Conferences. In this paper then, only the major trends are pointed out by quoting a few examples; these examples are not intended to be comprehensive nor to appear as those of most merit.

The chapter will first deal with some overall aspects of structural integrity of pressure vessels, mentioning the emphasis put on pre and in-service inspection and the development of national codes and attitudes (e.g. ASME XI, RCCM, the German "Basic Safety Approach", the UK evidence for the Sizewell B enquiry). It will emphasize the common features including the right choice of material with good fracture toughness, the use of a sound Fracture Mechanics approach, the use of pre and in-service inspection, the specification of how to deal with flaws found during such inspections.

The next section of this chapter will deal with Materials aspects covering damage effects such as strain ageing, irradiation embrittlement, underclad cracking and stress, the quality and size of available materials; improved welded processes; improvements in the knowledge of lower bound and average fracture toughness levels.

The Fracture Mechanics section highlights the move from Linear Elastic to Elasto-Plastic treatment, refering also to stable crack growth and tearing concepts. The great interest in assessing Thermal Shock effects and the LOCA situation leads on to an interest in Warm Pre-Stressing and Crack Assest. Improvements in theory and experimental methods include discussion of the prediction of large-scale behaviour from small specimens.

The last section of the paper deals with Non-Destructive Examination aspects. The emphasis is put on assessing and improving the effectiveness and reliability in detecting, and sizing flaws of potential significance. This is described partially through the work of the Post-Conference Seminars covering the various reports on the Programme for Inspection of Steel Components (PISC) and on work related to Intergranular Stress Corrosion Cracking (IGSCC). The developments in national codes, ultrasonic tests on austenitic steel, acoustic emission studies and computation will also be described. Finally, the close interaction of NDE development and requirements with those of Fracture Mechanics and Structural Integrity Assessment is emphasized.

OVERALL ASPECTS OF STRUCTURAL INTEGRITY OF PRESSURE COMPONENTS

The period from the first SMIRT Conference to the present date has been one which has seen a progressive development in the procedures, codes and standards related to assuring the Structural Integrity of steel reactor pressure vessels and associated pressure components, this development being associated with the concepts of design by analysis and the use of fracture mechanics (FM) and non-destructive examination (NDE) for flaws. The papers presented at the various SMIRT Conferences and Seminars over this period provide the evidence or describe the results of much progressive development. One of the most important codes in this respect is the American Society of Mechanical Engineers′ Boilers and Pressure Vessel Code, particularly its Section III (Nuclear Power Plant Components) often referred to as ASME III, and the Section XI issued in 1970 on in-service inspection of such components (ASME XI). Although ASME III for Nuclear Vessels was first issued in 1963, it was first used with its present title in 1971, an Edition which was also notable for defining four plant operating conditions (normal, upset, emergency and faulted) which must be examined, and which encouraged design-by-analysis on the rules suggested by Bernard F. Langer in his 1970 Murray Lecture ″Design-stress basis for Pressure Vessels″.

The development of a provision for brittle fracture analysis occurred in the period covered by the first few SMIRT Conferences, Section III being revised in 1974 to require the evaluation of the fracture toughness stress intensity factor K_{1C} as a function of material properties. The appreciation that failure was dependent not only on material toughness but also on the size of any crack-like flaw led to the introduction into Section XI of rules relating to flaw acceptance standards and for fracture mechanics analysis in 1973. Section XI also provides more details on how to carry out such a fracture mechanics analysis and the material properties to be used. One example is the K_{Ir} curves (Figure 1) indicating the maximum fracture toughness values for particular codified materials, the revision of which was the subject of several SMIRT papers in 1979-1981 [1].

Particular aspects of the ASME XI approach which were closely associated with the fracture mechanics research of the period, were the ways specified for dealing with multiple defects depending on their size and closeness and the distinction between surface (or near surface) defects and sub surface defects (Figure 2) [2]. The emphasis on size in the through-thickness direction for such flaws has a major influence on the NDE work reported later in this review. Section XI also provided rules for dealing with fatigue crack growth, the subject of several papers constituting a whole Section at the 6th SMIRT in 1981 and at later Conferences, some of which give attention to environmental effects (Figure 3) [3].

There is no doubt that ASME (Sections III and XI) are of great importance in assuring the structural integrity of reactor pressure vessels and their associated pressure components and that ASME has progressively shown initiative in incorporating Fracture Mechanics and NDE developments. The Code is now widely used internationally and has formed the basis of several national standards. However, in several countries the reappraisal of pressure vessel integrity assessment procedures led to then making some additions and alternations to the ASME requirements. One example of this is the French RCCM document which was based on regulations outlined by Torquat and Roche in 1975 [3]. The fatigue and fracture assessment of this approach were described by Collett, Foult and Faudy at SMIRT 8 (Chicago, 1983). They made use of a computer programme called FISSURE and included cover of the influence of flaw shape. The RCCM approach uses different safety coefficients for normal, emergency and faulted conditions (Table 1) [4].

TABLE 1 - Safety coefficients according to RCCM

Toughness / Operating conditions	K_{Ic}	K_{Ia}	K_{Jc}	J-Δa Resistance curve
Normal	0.4	0.7	0.7	-
Emergency	0.5	0.85	0.85	-
Faulted	0.8	0.8	0.9	0.8

In preparation for the Sizewell B PWR, Marshall in the UK chaired a committee of experts who looked into all aspects of the structural integrity assurance of the PWR reactor pressure vessel. Their first report was summarized at SMIRT 4 (1977) [5], and this estimated failure probability under normal, upset or test transient as being 10^{-6} per vessel.year provided that there was rigorous attention to fabrication techniques, quality control and inspection. It made its assessment of critical crack depths, which in all cases exceeded 0.45 of the thickness on the basis of ASME XI, Appendix A assumptions. In a later paper [6] a reappraisal gave more attention to improved material preparation, to the use of the R6 fracture mechanics approach and to the calculation of the critical defect size under loss-of-coolant accident and steam-line break conditions, again with satisfactory conclusion or integrity but emphasizing the measures needed to provide a reliable and effective inspection for flaws, both in-service and pre-service inspections. Following discussion in SMIRT 4 in 1977, probabilistic fracture mechanics were used in these Marshall Committee reports, this topic was also a feature of the other papers in the G6 Session of 1977, including papers from France [7] and from Germany [8].

Another major re-appraisal of the Structural Integrity requirements was made in the Federal Republic of Germany by their Reactor Safety Commission (RSC) and the associated research led to many papers presented at various SMIRT Conferences. The German RSK Guidelines and their KTA Code was much influenced by the Basic Safety Approach, features of which are shown schematically in Figure 4 [9].

Some of associated development work was described by Kussmaul in his Principal Division Lecture of SMIRT 3 in 1975 and at SMIRT 4 in 1977 [10]. Improvements in material led to reduce occurrence of stress relief cracking and to high upper shelf energy, whilst increased component size allowed the welds to be moved to less critical areas, as shown in Figure 5 [9].

Much more could be said on these developments in structural integrity assurance, but limitations of space prevent this. Suffice it to say that there is general agreement from all the SMIRT work that high integrity can be attained but this needs careful design with low primary, secondary and peak stresses. Such design must include specific failure analysis and must give attention to minimizing the number and volume of welds and to their inspectability; the choice of uniformly clean materials of high fracture toughness with freedom from stress-relief and underclad cracking, tolerant of variations in the fabrication process and with low irradiation and ageing embrittlement, the choice of fabrication methods that give low probabilities of defect production and low degradation effects; the specification and application of highly effective non-destructive examination with acceptance standards defined by fracture mechanics assessment. The control of operational conditions to be within those designed for, with appropriate measurement and recording.

MATERIALS ASPECTS

Materials

The basic function of the pressure boundary is to contain the coolant which is needed to transport the fissile heat from the fuel rods to the plant for power conversion. To provide intensive cooling, optimum pressure and temperature for light water reactors (LWR), liquid metal cooled fast breeder reactors (LMFBR) and gas cooled high temperature reactors (HTR) are set. These parameters and the envisaged thermal output determine the design requirements.

The design of coolant systems comprises style of the components, selection of ferritic and austenitic materials, loading parameters resulting from normal operation, as well as upset, emergency and faulted conditions. A simplified representation of the dependence of loadability on material optimization shows the influence of toughness and the importance of flaws, Figure 6. Advanced design requires the choice and specification of improved materials alloyed and processed with regard to cleanness, homogeneity, isotropy, strength, toughness, corrosion and erosion resistance. This is imperative for plant performance [11,15] constitutes as Basis Safety one of five principles of the German Basis Safety Concept [11].

An important prerequisite for structural materials application is the suitability for welding, Figure 7, which may increase the susceptibility to heat affected zone embrittlement and cracking as well as to sensitization for corrosion attack possibly leading to serious transgranular or intergranular stress corrosion cracking. Components adequately manufactured of optimized high quality materials can be treated by a mechanistic-deterministic safety evaluation through analysis and testing. For poor quality components, a probabilistic evaluation is needed, Figure 6.

As far as possible, plant ageing is being implemented in the design to cope with the potential ageing mechanisms. Corrosion impact on the materials may exhaust dramatically both static and cyclic loading capacity. This becomes even more critical, when flow induced erosion is occurring. Irradiation or creep embrittlement with interaction of the coolant and the stresses may occur in the RPVs, respectively the reactor tanks and the high temperature piping.

The safety margin during the life is dependent on a variety of parameters and may differ through the implementation of different codes and standards. Style of the components, selection of resistant materials and procedures for processing, fabrication and especially welding as well as quality assurance are important for the assurance of the required safety margin.

Uncertainties which are unavoidable can be covered by the assumption of lower bound characteristics. With the knowledge of the worst case condition, the minimum required safety margin can be evaluated and thus the problem of extrapolation to long-range operation eases off.

Ageing Phenomena

Pressure retaining components of a nuclear power plant are subjected to a variety of ageing mechanisms which are - with the exception of neutron irradiation - well known from operation of conventional plants, Figure 8. Considering components essential for safety, it becomes obvious that for LWRs fatigue and corrosion are the most frequently acting ageing parameters, Figure 9. LMFB reactors and HTRs suffer mostly from creep, creep embrittlement and creep-fatigue, in the case of LMFB tanks irradiation assisted.

The remaining life results from the amount of already absorbed loading capacity. Therefore, the more the target life is being approached, the higher are the requirements with regard to validated methodologies and accuracy to assess the safety margin and remaining life, respectively, considering time dependent degradation [16,17]. Upgrading measures such as replacement and mitigation of operating conditions have increased the safety margin to such an extent that lifetime extension even far beyond the design life becomes feasible.