ADVANCES IN FRACTURE MECHANICS ANALYSES OF PRIMARY SYSTEM PERFORMANCE UNDER OPERATING AND ACCIDENT CONDITIONS

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SUMMARY

Safety research sponsored by the Nuclear Regulatory Commission, Division of Reactor Safety Research, has resulted in notable advances in several areas of importance in the safety evaluation of reactor primary systems under normal operations and accident situations. First, the methods of Linear Elastic Fracture Mechanics and of Elastic Fracture Mechanics have been validated for prediction of pressure vessel performance by the Intermediate Vessel Test program results at the Oak Ridge National Laboratory. The test results from both hydraulically and pneumatically loaded pressure vessels are in very good agreement with the pretest predictions of vessel failure for the given conditions of pressure, temperature, material properties, and type and size of flaw. This ability to confidently predict vessel performance under realistic service conditions has permitted development of the computer program OCTAVIA which computes failure curves for a range of flaw sizes in terms of pressure and temperature for specified pressure vessel material at specific neutron fluence levels. The code then considers the probability of occurrence of flaw sizes and magnitude of pressure and determines the probability of failure for both individual flaw sizes and for the full spectrum.

A second advance in fracture mechanics technology has been made in thermal shock analysis. This advance has been verified by the confirmatory results of testing small tick-walled cylinders under thermal shock conditions in the Heavy Section Steel Technology program, and of warm prestressing tests at the U.S. Naval Research Laboratory. The analyses developed in these efforts have shown that operational pressure-temperature transients can be effectively modeled and applied to pressure vessel materials having appropriate mechanical properties simulating radiation embrittlement, and that the crack initiation/arrest conditions predicted will occur. Thus, the methodology has been developed for the analysis, and it shows continuing integrity of reactor vessels even in the presence of severe thermal shock transients.

Thirdly, the technology of crack arrest has reached a level wherein standardization of test specimens and testing methods is now possible and, indeed, is underway. The detailed factors which affect arrest of running cracks, and how those factors must be evaluated to permit determination of a minimum toughness descriptive of crack arrest are better understood. Although this area of fracture mechanics technology still requires much validation testing, the currently available results are already being utilized in operating and accident analyses.

1. Introduction

The analytical capabilities of the fracture mechanics methodology have been of immense value for the evaluation and establishment of safe design and operating parameters for the primary system components of light water power reactors. A salient characteristic of the methodology is the precision with which the otherwise bewildering array of stresses, loads, flaw sizes, material properties and temperature effects can be handled to produce the necessary answers to questions of either acceptable or unacceptable design, operating, or accident conditions. For higher assurance of safety, it is necessary to evaluate the performance of a component during a postulated accident to see whether adequate safety margins still exist.

Over the past several decades, much basic and applied fracture mechanics research activity has been underway with the result that the linear elastic fracture mechanics (LEFM) methodology is now accepted as well as established and proven for use. Procedures for stress-flaw size-material property evaluations in the elastic plastic toughness range have been developed and are well on the way to verification. Procedures for fully plastic analysis are less well developed and established. Most of the fracture mechanics analysis procedures are completely verified and are being used extensively and effectively on a wide variety of problems of interest to the U.S. Nuclear Regulatory Commission (NRC). Some specific problems and the advances which result in improved primary system safety analysis are discussed in this paper.

2. Fracture Mechanics Verification for Pressure Vessels

As stated above, the fracture mechanics methodologies have been verified for a number of structures and applications. For example, LEFM has been well verified for aircraft structures and for brittle materials like glass. Because of the importance of safety for nuclear pressure vessels, it became necessary to establish a more quantitative analytical method for pressure vessel evaluation to replace some of the relatively qualitative methods in use when the nuclear industry began to place nuclear reactors in service. To establish fracture mechanics methodologies for nuclear reactor applications, the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory (ORNL) conducted a series of failure tests using thick-walled pressure vessels of intermediate size with the test conditions being set by fracture mechanics methods, and the post-test analysis also being performed by these methods. The results are that LEFM was shown to be completely applicable for the temperature and material property range of its application, and several methods for elastic plastic fracture mechanics (EPFM) analysis were shown to be sufficiently accurate for immediate use in safety analyses.

The results of the nine separate intermediate-sized test vessel (ITV) tests, which verify the analytical fracture mechanics procedures, have been published or reported (see references [1 through 6]) and are summarized in reference [7] and Table 1. The tests were conducted at temperatures which resulted in fracture in the elastic range at or near the fracture toughness transition temperature (test 2), fracture in several portions of the transition temperature regime giving increasing degrees of plastic behavior (tests 4 and 9), and fracture totally in the plastic range at ductile shelf temperatures (tests 1, 3, 6, 5, 7, and 7A). The pre- and post-test failure calculations show the acceptably high degree of accuracy for LEFM and the elastic-plastic methodologies for such calculations. It is noteworthy that the ITV's were of sufficient thickness [152 mm (6 in.)], that the deliberately placed flaws were subjected to the full constraint to be expected in full-size reactor pressure vessels currently in service. All flaws had sharp fronts typical or more severe than those to be expected in naturally occurring flaws in service. The flaws ranged in depth from about quarter thickness to almost 90% of the wall thickness. These are much greater than would be expected to occur in service. Flaws were located in the cylindrical shell sections in both the inner and outer vessel surfaces and also at nozzle corners. All but one test was conducted using hydraulic loading. The one other test was conducted with pneumatic (sustained) loading, which could be the loading experienced in an accident situation wherein primary system integrity were breached so that the hot primary water could continuously flash into steam thus maintaining the pressure load on the crack front for a relatively lonq period of time.

The ITV test employing sustained (pneumatic) loading was especially noteworthy because it was a repeat test under conditions identical to one previously conducted under hydraulic loading. In both cases, a flaw almost 90% of the wall depth was placed in the outer surface, and pressurization was effected until fracture. The same ITV was used for both tests with the fracture region being repair welded following the initial hydraulic test. In each test, the failure occurred by slow stable crack growth of the crack front through the remaining ligament. For the case of hydraulic loading, a leak occurred and the crack front resealed itself at a slightly lower pressure, maintaining that pressure until relieved by manual system unloading. For the case of sustained (pneumatic) loading, the remaining ligament separated and the pressure was sustained by a thin stainless steel membrane welded to the inside of the vessel covering the crack region. The pressure was maintained at the failure pressure for a significant time (approximately 30 minutes) but the crack front did not advance even in a slow stable manner.

As previously stated, the analytical results based on LEFM and those from methods of EPFM, based on the results of these tests, closely predicted the experimental results. These test results and the analyses of reference [7] made it possible to develop the computer code OCTAVIA to calculate the failure pressure required under specified conditions of irradiation, initial properties, and assumed initial flaw size. The analysis is restricted to postulated overpressurization transients that occur isothermally and flaws that might exist on the inner surface of a reactor pressure vessel (RPV) in the beltline region. This computer code was then extended to consider the probabilities of flaw size distribution and occurrence of overpressurization transients, and was finally related to the probability of an overpressurization transient causing a reactor pressure vessel failure.

Sensitivity studies using the OCTAVIA Code have provided some preliminary results that are of particular interest to nondestructive examination, both applied and research. Figure 1 illustrates a typical result indicating the relative importance of various flaw depths to overall probability of failure. On the basis of assumptions made, it is quite clear that medium-sized flaws, between 1/2 inch and 1-1/2 inch (13 mm and 38 mm), are by far the most important in determining the probability of failure of an RPV due to an overpressurization transient. Smaller and larger flaws do not contribute significantly to the failure probability. This will have a direct bearing on the direction of research on nondestructive examination for the results indicate that more emphasis must be placed on the ability to detect flaws that are or could become within the critical range, with less emphasis on the detection of both very small flaws and very large flaws. Other similar results are expected from this computer code when it has been fully developed and when more use has been made of it.

3. Thermal Shock Analysis

One of the most severe accidents that has been postulated for an RPV is an acute thermal shock. For licensing purposes, it is assumed that a loss of coolant accident (LOCA) occurs as an instantaneous, double-ended rupture of a main coolant line, followed by operation of the emergency core cooling system (ECCS). Since this accident is postulated to occur at operating conditions, the vessel wall would be at 288°C (550°F); the blowdown would occur very rapidly, and cold emergency core cooling water would be injected into the RPV contacting and, thereby, shocking the inner surface. The stresses induced by this thermal shock are considered to be acting in the presence of postulated flaws in subsequent analyses which are performed to determine vessel integrity.

It is well known that linear elastic fracture mechanics (LEFM) will properly characterize the response of flaws to mechanically-induced stresses, and there is every reason to believe the same for thermally-induced stresses. Thus, methodology was developed using LEFM to consider not only the initiation of rapid crack propagation, but also arrests and reinitiations under the thermally-induced stresses of thermal shock transients. In addition, the beneficial effects of warm prestressing to crack initiation have been incorporated into the methodology. Verification that this methodology does properly characterize the response of cracks subjected to thermally-induced stresses has now been obtained in an analytical-experimental program conducted under the auspices of the NRC-sponsored HSST program at ORNL (references [8 through 11]).

Normal, unirradiated, quenched and tempered RPV material is much too tough to attain crack instability for any depth of initial crack, assuming an initial temperature of 288°C (550°F) and a coolant of cold water. Therefore, RPV steel was metallurgically degraded to achieve crack instability under these test conditions, by quenching the steel from 871°C (1600°F) with no tempering. Approximately 25 mm (1 inch) was removed from the cylinder surfaces to eliminate both residual stresses and metallurgically-different material to approach material homogeneity.

Table 2 shows the results of the four thermal shock tests conducted on flawed cylinders made from the degraded material. Analyses of these results, given in references [8 through 11], show that LEFM does characterize them. The general subject of thermal shock analysis, including a summary of these results, is presented in reference [12]. In summary, for the first thermal shock experiment, TSE-1, no crack propagation was predicted and none occurred (although this was a severe thermal shock with degraded material). In TSE-2, the crack was predicted to grow longer, but not deeper, and this did occur as predicted. In TSE-3, the crack was predicted to grow deeper and arrest, and this occurred; the arrest position occurred as predicted within the estimated crack arrest toughness limits. In TSE-4, the predictions and results were essentially the same as for TSE-3.

Considering the variations that exist in temperature profiles, K profiles, and K_{Ic} vs temperature, this series of tests has verified that LEFM does indeed characterize the mechanics of crack initiation and arrest in thermally-shocked cylinders.

An important addition to the LEFM analysis methodology occurs due to warm prestressing (references [13 and 14]). These recent tests at the U.S. Naval Research Laboratory have verified that given a K-value at a tough condition (high temperature), if the K-value and the temperature were then simultaneously decreased (as occurs during ECCS operation following a LOCA), crack initiation will not occur. This is true regardless of the ratio of the imposed stress intensity factor to the fracture toughness. Additionally, the results indicate that after partial unloading, if the loading is increased isothermally to failure, the K-value at failure is always equal to or greater than the warm prestressed K-value, and much greater than the K_{Ic}-value without warm prestressing. Therefore, crack initiation is not predicted when K equals or exceeds K_{Ic} if the K-value has been higher earlier in the transient. Studies are currently underway to provide verification of warm prestressing effects on cylinders.

In summary, these results indicate that no initial crack in the wall of an RPV could be driven through the wall to cause a leak as a result of ECCS operation following a LOCA. This summary is valid for pressure vessel steel irradiated to end-of-life fluence. Analyses of the expected static fracture toughness of the irradiated material, considering the toughness gradient through the wall coupled with the thermally-imposed loads acting on postulated flaws of virtually any depth show that initiation may occur for a certain range of initial flaw depths, but reinitiations effecting deep penetration of the wall do not occur due to warm prestressing.

4. Crack Arrest

Crack arrest is a very important aspect in the safety analysis of a nuclear power plant because there must be assurance that adequate toughness exists in the material to arrest a running crack. Establishment of crack arrest theory and analysis methodology has been a high priority task over the past several years. During this time, crack arrest methodology has been advanced significantly in large measure as the result of a cooperative program sponsored by NRC and the Electric Power Research Institute. An interim status on developments in fast fracture and crack arrest will be reported in reference [15]. The contractors in this cooperative program are Battelle Columbus Laboratories (BCL), the University of Maryland, and Materials Research Laboratory (MRL).

The laboratories (BCL and MRL) have developed test methods and related analyses; methods of crack velocity measurement; methods of applying the initial loading, including duplex specimen design and manufacture; loading system effects; and many effects characterizations, such as side groove geometry and depth and test specimen thickness. BCL has developed one-dimensional and two-dimensional dynamic computer codes.

The University of Maryland has used dynamic photoelasticity to further characterize the dynamic behavior of propagating and arresting cracks. The K-value versus crack velocity relationship has been determined for three birefringent plastics--one brittle, one moderately tough, and one tough. This characterization was found to be virtually independent of specimen geometry and to have the shape of an inverted "L."

The current efforts in crack arrest analytical development are basically dynamic generalizations of LEFM. There is general agreement that the minimum in the K-value versus crack velocity relationship represents a reasonable and conservative measure of the crack arrest toughness. If test specimen dynamics are minimized, it may be possible to measure this minimum toughness using small crack jumps, but the crack velocity must attain sufficient magnitude to go through the velocity at which the minimum toughness resides. Further characterization of this type is required for RPV steels. Sophisticated dynamic analyses are needed to assure that a "static" application of this minimum toughness concept provides a conservative analytical tool for RPV analyses. The dynamic analyses appear to properly characterize specimen dynamics, and the same is expected when structures are evaluated. Crack arrest prediction methods have been refined over the past several years by results of many tests on both steel and photoelastic plastics using a number of different specimen geometries. Sufficient validation of the analytical procedures has been obtained for specimen geometries that standard crack arrest test specimens and analysis methods have now been proposed for American Society for Testing and Materials acceptance. Furthermore, task group E-24.03.04 on dynamic and arrest fracture toughness of the committee on fracture testing is in accord with this proposal and is proceeding to develop a cooperative testing program. The objective of this program is to test the applicability of the proposed methods for measurement of the crack arrest toughness in the toughness range of practical interest. In addition, it is expected that this program will result in such clarifications and refinements as are necessary. The applicability of the crack arrest analysis methodology to structures will now be validated by studies at ORNL, starting with crack run-arrest tests on small model vessels, and ending with a crack run-arrest test on an intermediate test vessel. Fine tuning of the methodology should then be possible for direct application to the American Society of Mechanical Engineers (ASME) Code and to NRC licensing.

The test data to date provide added verification for the reference fracture toughness (K_{IR}) curve as presented in the ASME Code, Section III, Appendix G, up to the highest toughness levels that have been measured to date. This curve has been considered as representative of the lower bound of all fracture toughnesses. These data thus enhance the present confidence in the use of that curve for code and licensing applications.

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SUMMARY OF TEST CONDITIONS AND RESULTS OF HSST INTERMEDIATE TEST VESSELS

Table 1

Vessel No.	(°C/°F) Test Temp.	Mode of Failure	Remarks	(mm/in.) Flaw Size a	(mm/in.) Flaw Size 2b	(MPa/ksi) Failure Pressure and Strain (%)	Calculated Results: Pressure, MPa/ksi Strain, % LEFM	Calculated Results: MPa/ksi Plastic Instability	Failure Pressure / Design Pressure	Vessel No.
2	0/32	Flat	Transition Range	64.3/2.53	210.8/8.30	192.3/27.9 0.194	188.9/27.4 0.206	--	2.87	2
4	24/75	Mixed	Transition Range	76.2/3.00	209.6/8.25	182.7/26.5 0.168	180.6/26.2 0.163	--	2.73	4
9	24/75	Mixed	Transition Range	30.5/1.20	Nozzle	185.4/26.9 1.05	93.1-126.8/13.5-18.4 0.070-0.096	--	2.77	9
1	54/130	Mixed	Static Upper Shelf	65.0/2.56	209.6/8.25	198.5/28.8 0.92	189.6/27.5 0.345	206.1/29.9	2.96	1
3	54/130	Mixed	Static Upper Shelf	53.6/2.11	215.9/8.50	213.7/31.0 1.47	189.6/27.5 0.398	217.1/31.5	3.19	3
6	88/190	Shear	Static and Dynamic Upper Shelf	47.5/1.87	133.4/5.25	219.9/31.9 2.0	189.6/27.5 0.479	232.3/33.7	3.28	6
5	88/190	Leak	Static and Dynamic Upper Shelf	30.5/1.20	Nozzle	183.4/26.6 0.25	131.7/19.1 0.100	--	2.74	5
7	91/196	Leak	Static and Dynamic Upper Shelf	134.6/5.30	472.4/18.6	147.5/21.4 0.12	143.4/20.8 0.109	131.7/19.1	2.20	7
7A	91/196	Leak (Sealed)	Static and Dynamic Upper Shelf	134.6/5.30	472.4/18.6	144.2/20.9	143.4/20.8 0.109	131.7/19.1	2.15	7A

SUMMARY OF TEST CONDITIONS AND RESULTS OF HSST THERMAL SHOCK EXPERIMENTS

Table 2

O.D. = 533 mm (21 in.), I.D. = 241 mm (9.5 in.), Length = 914 mm (36 in.)
SA508 CL 2, Quenched from 871°C (1600°F), Test Duration = 30 minutes

Thermal Shock Experiment	TSE-1	TSE-2	TSE-3	TSE-4
Thermal Shock Vessel	TSV-1	TSV-2	TSV-1	TSV-2
Flaw Geometry	long axial	semi-circular	long axial	long axial
a, mm (in.)	11 (0.42)	19 (0.75)	11 (0.42)	11 (0.42)
T initial, °C (°F)	290 (550)	298 (552)	291 (555)	291 (555)
T sink initial, °C (°F)	4.4 (40)	-23.1 (-9.5)	-23 (-10)	-25 (-13)
Coolant	water	40% methyl alcohol 60% water	40% methyl alcohol 60% water	40% methyl alcohol 60% water
K/K_{Ic} maximum	0.74	1.33*	1.13	1.29
Propagation	no propagation	none in depth 6½ inch final length	grew to 36.5 mm (1.44 in) in depth	(0.42 in) grew 11 mm in depth

*At 75° angle from maximum depth position

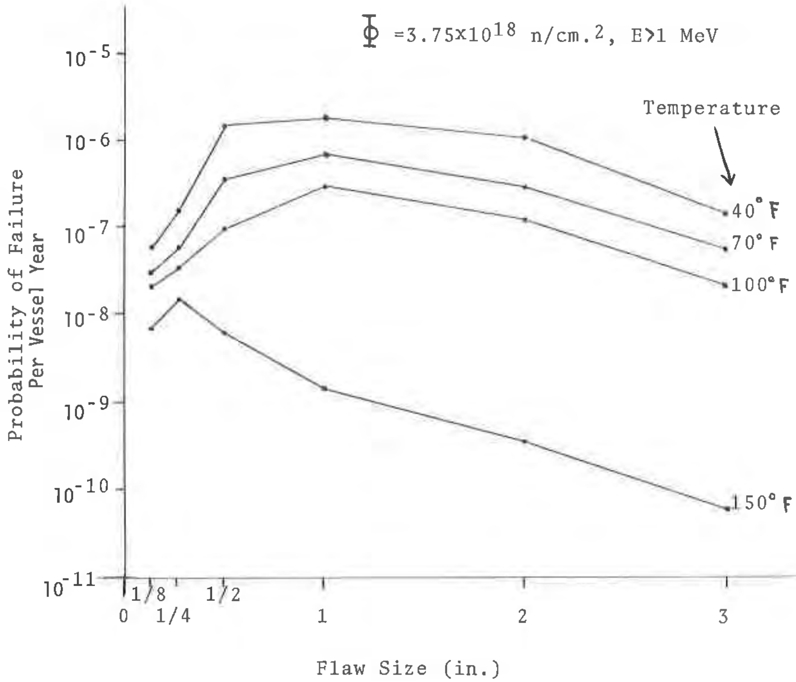

FIGURE 1

Computer output from OCTAVIA code showing influence of flaw size on failure probability. Flaws between 1/2 and about 1 1/2-inches are seen to have much more significance to failure probability than very small flaws or flaws larger than about 1 1/2 inches.