Integrity Assessment of Operating Gas-Cooled Reactor Components

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ABSTRACT

This paper presents various studies undertaken by Electricité de France to assess the integrity of operating gas-cooled reactor components. Exemples include those of Bugey, Chinon and Saint-Laurent units which needed such studies to handle unusual situations. Fracture mechanics methodology used was quite sophisticated as compared to original design analysis which generally prove to be more conservative than today's approach.

1. INTRODUCTION

Since the 60's, Electricité de France has been involved in the design, construction and operation of several gas-cooled nuclear reactor. Five of them are still connected to the grid : Chinon unit 2 and 3, Saint-Laurent unit 1 and 2 and Bugey unit 1. In the design analysis, the risk of fast fracture was addressed using what was, at that time, the current practice in fracture mechanics, i.e. the Pellini fracture analysis diagram. For various reasons, more detailed and sophisticated analysis were required. Three examples are given in the present paper.

2. THE CHINON A2 CASE

2.1. Design analysis and problem description.

The Chinon A2 unit is a 180 MW gas-cooled reactor which started operation in 1965. The core is contained within an almost spherical shell made of low alloy ferritic steel equivalent to ASTM 302 gr.A. Diameter is about 18.3 m and thickness 95 mm. The normal operating pressure is now 21 bars and the temperatures are around 200°C in the lower part and 400°C in the upper part. There are four inlet nozzles in the equatorial plane (Ø 1600 mm), four outlet nozzles in the upper part (Ø 1600 mm) and a large number of smaller perforations. At the design stage, the risk of fast fracture was assessed using Pellini Fracture Analysis Diagram. The results were turned into a "pressure-temperature" limit curve as shown on figure 1.

In may 1983, an unusual situation brought locally the temperature down to 62.5°C when the pressure was maintained at 13.6 bars. The corresponding point was outside the allowed domain (see fig. 1). Then EDF was required to study the significance of this incident with respect to the risk of fast fracture.

2.2. Fracture Mechanics Analysis

The idea was to estimate what should have been the size of a flaw for the incident to induce failure of the structure. Linear elastic fracture mechanics methods were applied.

First, the case of a current part of the shell was examined. The pressure stress was analytically evaluated and so was the thermal stress corresponding to the temperature drop during the incident (about 50°C in 1 hour). As far as material characteristics are concerned, the K_{IC}, K_{IA} and K_{IR} curves of ASME code were shown to be acceptable, the RT_{NDT} being equal to 55°C as indicated by the construction and safety reports. Testing and surveillance program allowed to neglect any irradiation embrittlement effect. Assuming an infinite (or more exactly, axisymetric) inside surface defect [1], the critical size was found to be larger than 60 mm, even when taking in account Irwin type plastic zone correction. The corresponding J_I value ($\simeq$ 25 kN/m)is much smaller than any possible J_{IC} limit so that small defects did not have the possibility to undergo ductile extension during the incident. At last, rapid but conservative analysis showed that, for that critical size, there is no risk of plastic instability ($\sigma << \sigma_{flow}$).

Then, the case of specific locations in the shell was addressed. In the nozzle corner, concentration factors taken from the safety report were applied on a quarter-circular corner crack [1]. Critical size remained above 50 mm. An other severe location is the part of the shell between CO_2 inlet and outlet where there may be a quite important temperature gradient on a short distance : 200°C during normal operation and about 50°C during the incident. Corresponding stresses were estimated using analytical model [2]. The critical size was found to be above 40 mm.

As a conclusion, the possibility of having a defect leading to component failure or degradation was ruled out. Then, such an incident was considered to have no consequence on structure integrity.

2.3. Pressure-temperature limit

The idea was to examine what would become the original pressure-temperature limit curve when applying today's practice with regard to protection against non-ductile failure. The methodology was imitated from ASME -section III- Appendix G or RCCM code - Annexe ZG (1st method).

A surface defect of 25 mm was postulated. Stress intensity factors associated with pressure loading (K_{IM}) and thermal gradient during heat-up and cool-down (K_{IT}) were calculated . The values of toughness K_{IR} were taken from the above mentionned codes.

The pressure-temperature limit curve was given by the equation :

$$2\,K_{IM} + K_{IT} = K_{IR}$$

This analysis led to the limit curve shown on figure 1. It can be noticed that this curve envelops the original one as also the point corresponding to the incident.

3. THE SAINT-LAURENT CASE

The Saint-Laurent A1 unit is a 400 MW gas-cooled reactor which started operation in 1969. The residual heat removal system of this reactor includes heat-exchangers with carbon steel tubes (O.D. 38 mm and thickness 3.66 mm). The intent of the plant operators was to allow this heat exchanger to be used as a safety "shut down" system which implies to put it quickly in operation with cold water. The question was to prove that the tubes were able to sustain the corresponding thermal shock of about 250°C.

Finite element computations were performed using different values of heat exchange coefficient. Then, stress intensity factor was computed for various crack sizes and configurations with influence function and plastic zone correction. Results are shown on figure 2.

Bibliographic investigations led to the conclusion that, in the given conditions, the toughness of the steel was above 100 MPa $\sqrt{m}$. No irradiation embrittlement had to be accounted for. It was then considered that under such a thermal shock, the fast fracture of the tube was very unlikely.

4. THE BUGEY 1 CASE

4.1. Problem description

The Bugey A1 unit is a 540 MW gas-cooled reactor which started operation in 1972. In this reactor, the graphite piling is supported by a very complex welded structure made of low alloy ferritic steel. This structure was designed so that stresses during operation do not exceed allowable values. During shut-down period, some fuel and component handling operations were allowed only if the temperature of the core support structure was above 70°C. Indeed, it was shown to be necessary for the structure to be protected against brittle fracture in case of fuel or component drop. The demonstration relied on Pellini Fracture Analysis Diagram and FTE point consideration and the anticipated irradiation embrittlement was accounted for. This temperature limitation being very restraining for plant operator, effort was attempted to get a better estimate of the fast fracture risk and to see whether the 70°C limit could be reduced to about 25°C or not.

4.2. Structural analysis

The first part of the study was devoted to a re-estimate of the effort associated with various fuel and component handling accidents. Drop tests were performed on a mock-up of the core support structure by the Commissariat à l'Energie Atomique and results were correlated with analysis. An impact force upper bound value of 800 kN was found and the corresponding stress field was computed using finite element technique.

With regard to the fast fracture prevention, presence of defects had to be postulated. In fact partial penetration welds between plates and girders were supposed to be the most sensitive points for fast fracture, the unpenetrated part of the joint (between roots of each side fillet welds) being considered as the largest "crack" which could be present in the structure. Applying computed stresses to such defects in a simplified local model allowed to estimate stress intensity factors. When mixed modes were present, an equivalent mode I stress intensity factor, K_I^*, was defined as [3] :

$$K_I^* = K_I + 1.22\,K_{II} + 0.74\,K_{III}$$

Then, all informations concerning material characteristics were gathered and reviewed, particularly the results obtained through the surveillance program which allowed to account for irradiation embrittlement. Using correlation with Charpy-V toughness, it was possible to determine lower bound values of dynamic toughness K_{Id} as a function of temperature : 70 MPa $\sqrt{m}$ at 25°C and 140 MPa $\sqrt{m}$ at 70°C.

As far as the criteria are concerned, it was felt necessary to take in account a possible interaction between crack extension and plastic instability. Then, a "double criteria" similar to the C.E.G.B. - R6 procedure [4] was found to be most appropriate. With this criteria, the risk of failure is assessed by positionning the point (K_R, S_R) in the Failure Assessment Diagram.

- K_R, which characterizes the risk of brittle fracture, is defined as K_I^* / K_{Id}
- S_R, which characterizes the risk of plastic instability is defined as $\sigma/\bar{\sigma}$

  . σ is the net primary stress
  . $\bar{\sigma} = \frac{1}{2} (\sigma_y + \sigma_u)$ is the flow stress

The K_R and S_R parameters were estimated for several partial penetration welds and several accidental conditions, at 25°C and 70°C. The corresponding points were plotted in the Failure Assessment Diagram as shown on figure 3. It can be seen that, if the margin with respect to brittle fracture is slightly smaller at 25°C than at 70°C, the overall situation is still acceptable.

All the computations were performed by the Neyrpic Cie nuclear division.

4.3. Consequences

Using state-of-the-art structural and fracture mechanics (finite element method, R-6 type Failure Assessment Diagram...), it was possible to show that the 70°C temperature limitation was too severe and that 25°C was acceptable with, still, a good safety margin. As a consequence, modification of specifications was authorized by Safety Authorities, giving plant operator the possibility to save significant time on outage periods.

5. CONCLUSIONS

It has been shown how sophisticated structural analysis may greatly improve knowledge and understanding of component behaviour and lead to more realistic failure assessment. In several cases, it allowed to demonstrate that design specifications could be safely relaxed in order to extend plant operation possibilities.

ACKNOWLEDGEMENTS

Help and support of E.D.F., C.E.A. and Neyrpic staff are gratefully acknow ledged.

REFERENCES

1 - ROOKE, CARTWRIGHT, "Compendium of Stress Intensity Factors", London, 1976

2 - TIMOSHENKO, "Strength of materials", Van Nostrand Ed., 1941

3 - POOK, "The significance of mode I branch cracks for combined mode failure", Proceedings 3rd European Conference on Fracture, 1980

4 - HARRISON, LOOSEMORE, MILNE, "Assessment of the integrity of structures containing defects" CEGB - Report R-H-R6, U.K., 1976

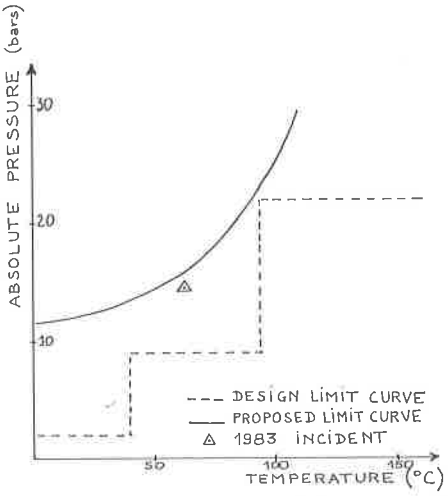

Figure 1 - Pressure - Temperature limit curves (Chinon A2)

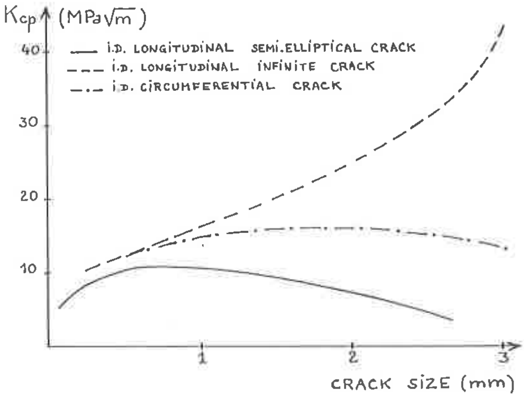

Figure 2 - Stress intensity factor for various crack sizes and configurations (Saint Laurent A)

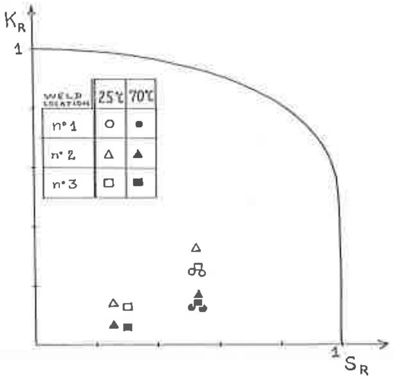

Figure 3 - Failure Assessment Diagram for various welds and accidental conditions.