Conference reports

The reliability of reactor pressure components

by K. Kussmaul*

Twenty-seven countries and five international organizations accepted the invitation of the IAEA to participate in an International Symposium on the Reliability of Reactor Pressure Components held in Stuttgart, in the Federal Republic of Germany, from 21 to 25 March 1983. The symposium was organized in co-operation with the Staatliche Materialprüfungs-anstalt (MPA) at the University of Stuttgart. About 45 contributions were presented in seven sessions.

Statistics gathered by the World Energy Conference and the IAEA show that there is no significant difference in the availability of fossil-fuelled and nuclear power plants. Nuclear power plants have attained more than 2800 reactor years of service with, in some cases, availabilities of 80% or more. It could thus be shown that the safe and economic operation of such plants is possible on the basis of existing regulations and design principles. However, the optimization of nuclear power plant design and materials technology, including quality assurance and controlled operation with repeated inspection, is an important prerequisite if safety and reliability are to be guaranteed in the future.

The potential still available to improve design has been shown by improvements in the primary circuit of German Pressurized-Water Reactors (PWRs). As a consequence, expenditure on non-destructive testing and risks in production could be decreased drastically: for example, the total length of welds in the reactor pressure vessel was shortened from 122 to 61 m, and the number of welds in the reactor coolant piping cut from 250 to 60. Together with reliable theoretical and experimental stress analysis and the use of optimized materials, these measures serve as the basis for a high degree of safety and availability. Bursting tests under the operational conditions of a PWR have been conducted on full-scale pipes with longitudinal and circumferential flaws. Taking into account the "leak before large break" criterion, it has been possible to define an allowable flaw size for operational conditions. For the main coolant piping it has now been proved that in keeping with the Basis Safety Concept a spontaneous catastrophic failure is not to be expected.

The Basis Safety Concept puts a special emphasis on materials. The development of reactor pressure vessel steels and welding techniques for reactor components has been characterized by the fact that the maximum ingot size for forged rings has been increased from 220 to 570 tonnes in the past 20 years. This increase has been accompanied by continuous optimization of quality with regard to homogeneity and isotropy, fracture toughness and weldability. By choosing suitable material, it has been possible to govern the phenomenon of stress relief cracking in the heat-affected zone of weldments.

The problem of irradiation embrittlement is still of great interest. Studies of the shifting of transition temperature in notch impact bending tests have shown that its influence on the upper shelf of the A_v-T curve was in the magnitude of data scattering. Generally, it is recommended that a high degree of purity be obtained for reactor pressure vessel (RPV) steels. The definition of "best estimate" properties as against "lower-bound" values has been discussed, in order to allow a more critical and realistic evaluation of the benefits achieved by improved steelmaking and fabrication practices. On the basis of available data, it can be concluded that for modern RPV steels, up to end-of-life condition, only a little degradation in toughness due to embrittlement caused by thermal ageing, strain ageing, and neutron irradiation is to be expected.

Several investigations into the fracture safety of PWR RPVs – for example, the results obtained by the Marshall Study Group in the UK – have shown that great importance should be attached to the analysis of the core shell and inside nozzle corners, mainly for the case of a large steam line break. The evaluation of accidents that have occurred has made a considerable contribution to the optimization of calculation, design, material selection, manufacture, and operation. The assumption that transients may occur in PWRs which lead to pressurized thermal shock, even with repressurization in the RPV, has been confirmed in service.

* Professor K. Kussmaul, Staatliche Materialprüfungsanstalt, University of Stuttgart, Fed. Rep. of Germany.

Cracks in PWRs have developed mainly in the inlet area of the feedwater piping in the steam generator. The cracks have been caused on the one hand by dynamic load (water hammer), and on the other by corrosion fatigue due to temperature fluctuations. Further, the numerous cracks developing in the feedwater piping attached to nozzles of BWR* RPVs could be clarified. Crack growth occurring in areas containing geometric or corrosive notches has been aggravated by oxygen attack on the crack tip. If local embrittlement is taken into account, it becomes obvious that only high component toughness and a wall thickness higher than ordinarily used would provide a sufficient safety margin against catastrophic failure.

Because of the special significance attached to non-destructive testing techniques, this subject was the centre of interest. Besides the RPV, the testing of steam generators has become of the greatest importance. The automation of test and analytical systems for the testing of steam generator pipes, and the application of multi-frequency Eddy current techniques, have brought about an effective reduction in inspection times, and in radiation exposures received by personnel. Advanced non-destructive testing techniques such as ALOK (Amplitude-time of flight-locus curves) will provide improved flaw detection and sizing in future; and holography, SAFT (Synthetic-Aperture Focussing Technique) and Phased Arrays represent the next generation of equipment in analytical systems.

The systematic collection, codification, and evaluation of operational experience is a considerable tool for enhancing the availability and safety of nuclear facilities. Data have therefore been collected in the past, mainly on a national basis, in systems such as the US "Significant Event Reports" and the "Nuclear Plant Reliability Data System" (NPRDS), the French "Système de Recueil de Données de Fiabilité" (SRDF) or the "Safety Information System" (SIS) used for the WWER-400 type of PWR in the CSSR.

The Power Reactor Information System (PRIS) of the IAEA has an international character, covering about 80% of all reactors in the world since 1970 and providing information, *inter alia,* on more than 10 000 shutdowns. PRIS is not, because of the limited amount of detail in the data it contains, supplying data on the availability of individual components but is offering information on parameters which are of importance mainly for planning purposes.

Apart from such data banks, an effective system has been introduced for continuous in-service verification of design principles. Taking all transients into account, this compares real plant behaviour on-line with design conditions and, if design conditions are exceeded, it provides adequate operational measures.

The Staatliche Materialprüfungsanstalt (MPA), Stuttgart, organized in parallel with the symposium, a Reliability Engineering Exhibition in which 15 international exhibitors demonstrated, for example, types and efficiency of non-destructive testing techniques and devices. An original pressure vessel for a 900 MWe BWR with various manipulator systems for automated ultrasonic testing was the main attraction. It will serve at the MPA to validate automated non-destructive examination equipment, and data processing.

* Boiling light-water-cooled and moderated reactor.