Chairman's Report on the Panel Discussion on High-Temperature Design

At the Third International, Conference on Pressure Vessel Technology, Tokyo, Japan, April 19–22, 1977[1]

Panel Members:
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Because of the large and growing interest in the design of structural components for fast reactors, this topic provided the main theme of discussion. As a starting point, Dr. Pai described the development of Code Case 1592 and the way in which this provided methods to assess the integrity of components against the various possible modes of failure. The Code Case, published by the American Society of Mechanical Engineers, provides the only freely available set of design rules for a high-temperature nuclear plant, and was originally drawn up with the liquid metal fast breeder reactor in mind. It provides an extension of the nuclear pressure vessel rules of Section III of the ASME Boiler and Pressure Vessel Code to temperatures above 375 C for ferritic materials and 450 C for austenitic steels. At the present time Code Case 1592 cannot be considered definitive. It is continually being improved and updated as the results of research work become available.

The ensuing discussion revealed strong similarities in the approaches adopted in the different countries represented at the meeting, in the way in which they were tackling the development of design methods and design criteria and in the way that they were conducting programs of research to resolve outstanding issues. There was general agreement that a very large amount of work was still needed in the fields of structural analysis and material properties before components could be designed with sufficient confidence to assure their survival over a working life of 25 to 30 years, under the extremely arduous conditions found in fast reactors.

The first topic for detailed consideration was design against creep failure, in circumstances where cyclic plastic strain could be neglected. In some countries the main emphasis was on computational methods to predict stresses and strains throughout the life of a component, these predictions then being compared against allowable values to assess the design. Other countries were developing simplified methods of analysis, which were claimed to be preferable on the grounds that they eliminated the need for expensive computations, and provided the designer with a better appreciation of the factors controlling structural performance. Whichever method was chosen, accurate prediction of component life required an adequate description of material behavior.

Constitutive equations, describing the deformation of the material, were required as input to calculations. Failure criteria were needed to assess whether the computed stresses and strains were acceptable. There was a general belief that constitutive equations for creep, in the absence of plastic cycling, were being developed to a point where they could be used with confidence. There was, however, a question of cast to cast scatter in properties and how this should be dealt with by the designer.

On the subject of failure data, opinions were more divided, with doubts being expressed about design methods based on specified acceptable strain levels. One concern was that strain to failure would be considerably less in a multi-axial tensile stress field than in a uniaxial test. It was also felt that some materials would not maintain a high creep rupture ductility throughout the whole working life of the reactor, and even where high ductility was maintained in tests in air, the sodium environment and irradiation effects could lead to embrittlement. An instance was quoted where significant cast to cast scatter in terminal ductility had been noted in an austenitic material, despite the fact that all casts had been within the specified composition.

Concern was also expressed about the performance of welds in a creep situation. The view was expressed that too much attention had been paid to parent material properties and too few creep tests had been performed on the welds. It is clear that further creep data are required on fast reactor materials, particularly in respect of long-term behavior. This is reflected in the materials testing programs which were mentioned by various contributors.

In the fast reactor, many of the more important components are subject to thermal shock and thermal cycling of a sufficiently high amplitude to cause plastic yielding. This can lead to unacceptably large distortion in a component, brought about by ratchetting effects. It can also lead to failure by fatigue. Several speakers drew attention to the difficulty in making satisfactory predictions of ratchetting, even in the lower temperature portions of the reactor, because of a lack of understanding of deformation behavior under cyclic loading. They pointed out that, in the higher temperature regions, where time dependent creep takes place, the difficulties are likely to be even greater. For similar reasons, it is dificult to estimate stress levels and strain ranges in a component under these conditions, and such information is an essential step in predicting routine life.

Concern was also expressed about the failure criteria to be used when assessing combined creep and fatigue damage. The essence of the problem is that data can only readily be generated in accelerated tests, and there is no fully validated method of extrapolating the results to provide the design data for a 25 to 30-year life. There was general agreement that much further work was required. In the longer term the hope was expressed that physical metallurgy could provide an understanding of the damage mechanisms involved. In the shorter term, there seemed to be no alternative to undertaking long term combined creep and fatigue tests on the materials, despite the heavy cost involved. Some speakers expressed reservations about the expense of the elastic-plastic-creep cycling computations needed to analyse high temperature components subject to plastic cycling. Development of simplified methods was seen as a desirable objective.

In addition to having to withstand normal operating conditions, most fast reactor structural components have to be designed to survive accidental overloads. The view put forward by some speakers was that the stress limits of Section III and Code Case 1592 were too simplistic. The real point at issue was: Could the component withstand the overload imposed on it, taking into account the need to demonstrate integrity in the event of an overload occurring towards the end of reactor life. Considerations of ductility and fracture toughness were likely to be more important than strength. This would imply much further experimental work to establish the likely end of life properties of fast reactor materials, taking into account such degradation as might occur by thermal aging, by combined creep and plastic damage, by the effect of a sodium environment, and in some cases by the effect of irradiation.

During the discussions, several references were made to the

[1] Three volumes of Conference Proceedings are available through the ASME Order Department, 345 East 47th St., New York, N. Y., 10017.

possible improvement in design codes for a more conventional plant, as a result of the Research and Development work carried out in the fast reactor area. The general feeling was one of caution. Examples were quoted where application of the nuclear design philosophy could lead to thicker sections than currently considered acceptable, and known to give reliable performance. Questions of economics apart, thicker sections were technically undesirable because they increased the risk of thermal fatigue damage. Further study was needed before existing design codes were altered.

Finally, returning to fast breeder reactors, two areas were identified where international collaboration would be desirable.

The first concerned the verification of computational methods. There is need to carry out benchmark calculations to demonstrate that the computer program used in different countires give similar results when applied to identical problems, and also to confirm that the computations agree with experiments on structural components. In parallel, there is need to investigate the validity of the simplified methods which are being developed, and comparison with computed results for the same problems would go a long way to meeting this objective.

Secondly, and equally importantly, there is a need to bring together data on materials used in fast reactors and other high temperature plants. A large amount of data has already been published, scattered throughout a wide variety of technical journals. In addition, much additional information is available in unclassified company reports. The task to be undertaken is largely that of seeking out such data and collating it.