PREFACE: SYMPOSIUM ON FUEL ROD FAILURE AND ITS EFFECTS

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The UO_2 fuel of a large, modern light-water reactor (LWR) is contained in upward to 40 000 zirconium alloy tubes. These tubes serve three basic purposes.

1. They maintain a definable, reactive geometry of the fuel.
2. They provide the heat transfer surface that ultimately results in power generation.
3. They provide retention of the radioactive fission products generated during operation of the reactor.

Further protection against release of fission products to the environment is provided by the primary system and the containment building.

However, a loss-of-coolant accident (LOCA), which is a breach in the primary system, can be postulated and is considered in the design of LWRs. During this accident the cladding will undergo a severe thermal transient. Initially, the transient is primarily the result of redistribution of the heat stored in the fuel, but decay heat of the fission products and the exothermic contribution of the zirconium-water reaction become predominant at later times and higher temperatures, respectively. Were this transient allowed to progress unhindered, it would result in gross core damage and permit release of the total inventory of volatile fission products to the containment.

Even though it is highly unlikely that a LOCA could occur, emergency core cooling systems (ECCS) are provided in LWRs to terminate the postulated accident transient. Before such systems become effective, portions of the core may reach temperatures sufficient to cause rupture of the fuel rod cladding.

The aforementioned rupture occurs when the hoop stress resulting from the fission gas pressure exceeds the strength of the cladding. Considerable swelling (plastic deformation) of the rods may occur prior to rupture and reduce the passage for the emergency coolant through portions of the core. Flow blockages can be hypothesized which would result in transient termination being seriously delayed, increasing the probability of embrittlement and subsequent disintegration and/or melting of the cladding and fuel.

Because of these possibilities, a concerted effort has been under way to

1. assess the modes of failure of the fuel rod cladding in terms of accident conditions and reactor operating history
2. determine to what extent deformation and rupture may affect emergency cooling capability
3. determine the margin of safety relative to the above.

The progress made to date in these areas is outlined in the following papers.

Papers by Emery et al., Hobson et al., Waddell, and Lorenz et al. report the results of experiments performed on the deformations of Zircaloy cladding under conditions simulating Zircaloy cladding under conditions simulating LOCAs. Both in-pile and out-of-pile tests have been conducted. The test geometries have included both single tubes and multiple-rod bundles. Test specimens were prepared from irradiated tubing as well as from unirradiated material.

Papers by Bingham and Lowe, Carbiener, Bock et al., and Hench and Liffengren describe analytical models developed to predict the distribution of cladding failures in a large power reactor and the effect of these deformations on ECCS performance. Heat transfer experiments conducted on deformed rod bundles provide the basis for appropriate heat transfer correlations and are discussed in the papers by Davis and by Cermak et al.

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