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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

SUBJECT: NEW DATA HAVING A POTENTIAL EFFECT ON THE REGULATORY PROCESS: HYDROGEN EMBRITTLEMENT OF ZIRCALOY FUEL CLADDING RUPTURED IN A SIMULATED LOSS-OF-COOLANT ACCIDENT STUDY

SUMMARY

Data recently obtained at the Japan Atomic Energy Research Institute (JAERI) on hydrogen absorption in, and embrittlement of Zircaloy fuel element cladding ruptured and oxidized in steam is causing concern in some licensing organizations that the present USNRC ECCS embrittlement criteria, as stated in 10 CFR 50, Appendix K may not be sufficiently conservative. The new data are reviewed below, with the inclusion of other pertinent data on embrittlement of Zircaloy cladding. We conclude that the new data do not decrease the conservatism of the ECCS embrittlement criteria, 10 CFR 50, Appendix K, but should be considered in any revision of these criteria.

INTRODUCTION

During the Sixth Water Reactor Safety Research Information Meeting at the National Bureau of Standards, Gaithersburg, Maryland, November 6-9, 1978, Dr. S. Kawasaki, JAERI, presented his most recent observations on hydrogen embrittlement of internally oxidized Zircaloy fuel cladding that had been ruptured during a heating transient in a steam environment (Enclosure 1). While the data indicate a degree of embrittlement by hydrogen in a ring compression test of portions of the ruptured and oxidized specimens, the "embrittled" sections supported considerable load in the test and allowed some radial deflection of the wall of the specimen. Other data, presented at the same meeting by Dr. T. F. Kassner, Argonne National Laboratory (ANL) showed that cladding specimens prepared similarly absorbed significant amounts of energy in impact loading before fracture occurred. Accordingly, we feel that there is as yet, insufficient evidence available to indicate that the conservatism of the ECCS embrittlement criteria of 10 CFR 50, Appendix K has been reduced.

DISCUSSION

Dr. Kawasaki has conducted three types of experiments related to the hydrogen embrittlement of Zircaloy fuel cladding during oxidation in steam. In one type of experiment, 22cm long tubular specimens filled with alumina pellets to simulate fuel were pressurized to burst at a temperature near 850°C, plunged into a muffle furnace at the desired oxidation temperature (ranging from 950 to 1200°C nominal) in flowing steam, held (after bursting) at the oxidation temperature for times up to 240 seconds, and then placed in a cold section of the furnace. The specimens were then sectioned into 15mm long pieces, and compressed diametrically (ring compression tests) in a tensile/compression testing machine at test temperatures of 100, 200 and 300°C. fFractured specimens were then analyzed for hydrogen content. The data show that the diametral deflection to fracture can be reduced to approximately 0.3mm (the cladding was 10.75mm in diameter) for those specimens located at the start of the bulged section of the ballooned and burst cladding, where the gap between the pellet and the cladding is small and the access of the steam entering the rupture is limited. Though this specimen is stated to be "embrittled," the specimen supported a load of more than 50kg before failure. Other data (Enclosure 2) presented by Dr. Kawasaki to members of my staff in private discussions on November 3, 1978, indicate that this "embrittlement" may be caused by oxygen, not hydrogen, even at hydrogen contents as high as 1000ppm. In the second type of experiment, Dr. Kawasaki found that ring compression specimens oxidized in stagnant steam had very little deflection before fracture while those oxidized in flowing steam deflected significantly even after 20 minutes of oxidation at any of the oxidation temperatures. In the third type of experiment, he found that ring compression specimens isothermally oxidized at temperatures from 950 to 1150°C in various volumetric ratios of hydrogen to steam had significantly less deflection to fracture if the ratio of hydrogen to steam was greater than 0.3 - 0.4 and the oxidation temperature was 1000°C or higher, independent of the time of oxidation (the shortest test time was about 3 minutes). Dr. Kawasaki concludes that, in the ruptured and oxidized cladding, the hydrogen is absorbed in a region of the specimen where the access of steam is too limited to maintain an oxide film on the inner surface and the faster diffusion of hydrogen in the helium gas in the gap allows a relative enrichment. We agree with both the data and this conclusion, but do not agree with any conclusion that the cladding is "embrittled" by the hydrogen absorbed to the extent that the present USNRC ECCS embrittlement criteria are not conservative.

Immediately after the presentation by Dr. Kawasaki to the WRSR Information Meeting on November 6, 1978, Dr. T. F. Kassner, Argonne National Laboratory, presented the results of his research on embrittlement of Zircaloy fuel cladding that has been supported by RES. He showed, among other data (see Enclosure 3), that specimens prepared similarly to those of Dr. Kawasaki could withstand and absorb relatively large amounts of energy in impact loading before fracture occurred. Specimens oxidized for four minutes and more at temperatures of 1300°C and below not only survived "reflood quenching" with water, but also survived loading to an impact energy of 0.3J, at room temperature, produced by a weighted pendulum striking the specimen near the area of hydrogen absorption and "embrittlement." These data are shown in Figure 4 of Enclosure 3. This energy level is equivalent to 1.5×10^4 J/m^2 and 0.96 ft-lbs energy absorption by a standard full-size Charpy impact specimen (a standard method of specifying toughness in pressure vessel steels, but the energy levels defining the ductile-brittle transition cannot be transferred from one metal system to another). The energy required to produce failure in the specimens surviving 0.3J impact could not be determined in the experimental equipment used. It is now being determined in new experiments in other impact testing equipment. Additional work is now underway at ANL to examine the problem further, and to provide a more quantitative tie between the ANL and the JAERI data. Completion is expected shortly.

Since the fuel pellets present in the pertinent zone of burst and oxidized fuel elements in a commercial power reactor would resist diametral deflection of the hydrided cladding to high crushing loads, and the impact energy absorbed before fracture of such material is relatively high, we can envisage, at this time, no accident scenario in which significant damage to the fuel assembly would be produced, including that of the dropping of a fuel bundle during disassembly of the core. Our analysis thus indicates that a coolable core geometry would be maintained, and that there has been no significant reduction in the conservatism of the present ECCS embrittlement criteria.

If you have any question, please call me (427-4341) or M. L. Picklesimer (427-4266), the program manager on my staff responsible for the cladding research programs of the Fuel Behavior Research Branch, Division of Reactor Safety Research.

Sincerely,

Saul Levine, Director
Office of Nuclear Regulatory Research

Enclosures: As stated

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D. B. Vassallo, ADLWR

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