RECENT RESULTS FROM LOCA STUDY AT JAERI

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SUMMARY

With a view to obtaining basic data to evaluate high burnup fuel rod behavior under loss-of-coolant accident (LOCA) conditions, a research program is being conducted at the Japan Atomic Energy Research Institute (JAERI). The program consists of integral thermal shock tests and other separate tests for oxidation rate and mechanical property of fuel claddings. In the integral thermal shock tests, short test rods were heated up, burst, oxidized in steam and quenched by flooding water to evaluate failure-bearing capability of fuel claddings under simulated LOCA conditions.

Degradation of cladding ductility due to reduction of cladding wall thickness by waterside corrosion and hydrogen absorption can have the greatest influence on failure-bearing capability of the high burnup fuel cladding. Prior to the tests of irradiated cladding tubes, the integral thermal shock tests were performed with non-irradiated cladding tubes, which were mechanically thinned and pre-hydrided to evaluate the separate effects. The test rod was axially restrained at the end of the isothermal oxidation to simulate the possible loading during cooling and quenching in the bundle geometry. However, the fully restraint condition is too conservative. Then, the tensile load was controlled not to exceed pre-determined load levels to realize intermediate restraint conditions, in addition to the fully restrained condition. The influence of pre-hydriding was obviously seen on the failure threshold value under axially restrained conditions. The threshold value of pre-hydrided cladding tubes was as low as 10% ECR under the fully restrained condition, which is the most conservative loading condition. The failure threshold generally increased as the pre-set maximum restraint load was decreased, and it was estimated to be higher than 20% ECR when the maximum restraint load was limited to below 535N (37MPa for initial cross section of cladding tubes).

The tests with irradiated claddings were started since January 2003 after test apparatus for the integral thermal shock tests were successfully equipped in the Reactor Fuel Test Facility at JAERI. Two PWR fuel rods, irradiated to 39 and 44GWd/t (rod average) at Takahama unit-3 reactor, are currently subjected to the tests. The cladding material is low-Sn Zircaloy-4 and initial oxide layer thickness ranged 18 to 25μm. Ten tests are planned for these fuel rods and three tests have been conducted. Fuel pellets were removed from 190mm-long segments, and alumina dummy pellets were loaded in the defueled claddings. Zircaloy end-plugs were welded at the both ends of the claddings and the fabricated test rods were pressurized to 5MPa with Ar gas. In the tests, the rods were heated up at a rate of 10K/s in the steam flow and isothermally oxidized at temperatures ranging from 1157 to 1192ºC. Cladding rupture took

place at temperatures between 780 and 810ºC during the heat up. After the isothermal oxidation for pre-determined periods, the rods were cooled in the steam flow to about 700ºC and finally quenched with water flooding from the bottom. The test rods were axially restrained to the maximum load of 540N (30 to 35MPa for initial metal cross section) in those tests. The maximum restraint load of 540N was selected to be slightly higher than the highest restraint load reported in references.

One cladding oxidized at 1180ºC to about 30%ECR* during the isothermal oxidation failed during quench. Since the failure boundary of non-irradiated claddings containing similar hydrogen concentrations lies at about 28%ECR, the failure of the irradiated cladding agrees with the failure criteria for non-irradiated claddings. Two claddings oxidized at 1192 and 1157ºC to about 16 and 18%ECR, respectively, survived the quench. This indicates that failure boundary is not reduced so significantly by irradiation to the examined burnup level.

More tests are planned with the 39 and 44GWd/t PWR fuel cladding and higher burnup PWR and BWR claddings (MDA**, NDA***, ZIRLO^{TM}, and Zircaloy-2) to investigate the influence of irradiation in detail.

Acknowledgment
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* ECR is estimated by the Baker-Just equation, taking account of double sided oxidation and wall thinning by ballooning. The 'initial' cladding thickness used in the estimation is metallic thickness after corrosion during the reactor operation.

** Mitsubishi Developed Alloy (Zr-0.8Sn-0.2Fe-0.1Cr-0.5Nb) was developed by Mitsubishi Heavy Industries, Ltd.

*** New Developed corrosion resistance Alloy (Zr-1.0Sn-0.27Fe-0.16Cr-0.1Nb-0.01Ni) was developed by Nuclear Fuel Industries, Ltd. and Sumitomo Metal Industries, Ltd.