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LIGHT WATER REACTOR FUEL RESPONSE DURING REACTIVITY INITIATED ACCIDENT EXPERIMENTS

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ABSTRACT

Experimental results from six recent Power Burst Facility (PBF) reactivity initiated accident (RIA) tests are compared with data from previous SPERT, TREAT and NSRR programs. The RIA fuel behavior experimental program recently started in the PBF is being conducted with coolant conditions typical of hot-startup conditions in a commercial boiling water reactor. The SPERT, TREAT and NSRR test programs investigated the behavior of single or small clusters of light water reactor (LWR) type fuel rods under approximate room temperature and atmospheric pressure conditions in capsules containing stagnant water. As observed in the previous tests, energy deposition, and consequent enthalpy increase, in the PBF test fuel appears to be the single most important variable. However, the consequences of failure at boiling water hot-startup system conditions appears to be more severe than previously observed in either the stagnant SPERT or NSRR tests. Metallographic examination of both previously unirradiated and irradiated PBF fuel rod cross sections revealed extensive variation in cladding wall thickness involving considerable plastic flow and fuel shattering along grain boundaries in both restructured and unrestructured fuel regions. In addition, swelling of the gaseous and potentially volatile fission products in previously irradiated fuel resulted in volume increases of up to 180% and blockage of the flow shrouds surrounding the fuel rods.

INTRODUCTION AND REVIEW OF PREVIOUS WORK

The rapid, inadvertent insertion of reactivity into a light water reactor (LWR) core leading to high cladding temperature has long been recognized as a potential mechanism for fuel rod failure. Complex analysis techniques are used to estimate the effects of postulated reactivity initiated accidents (RIAs) in LWRs. These techniques generally couple the transient neutronics behavior, fuel rod thermal and mechanical response, and the coolant hydrodynamic response. Verification of these analytical models is incomplete, however, due to limitations of existing fuel behavior data. Much of the applicable RIA experimental data were obtained several years ago in the SPERT (Capsule Driver Core) and TREAT test programs, which investigated the behavior of single or small clusters of fuel rods under near room temperature and atmospheric (or

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near atmospheric) pressure conditions, no forced coolant flow, and zero initial power. Similar tests have been performed in the Japanese Nuclear Safety Research Reactor (NSRR)[1].

In each of these facilities, a driver core with encapsulated test fuel in a central flux trap was operated to produce a power excursion. The magnitude and time duration of these excursions were comparable to those of severe, hypothesized RIAs in LWRs. The experiments were performed with single fuel rods (or a small cluster of rods) placed at the center of test capsules containing stagnant water. Energy deposition, and consequent enthalpy increase, in the test fuel was the single most important independent variable. The threshold for failure of unirradiated fuel was generally about 240 to 265 cal/g UO_2[a], and was relatively insensitive to cladding material, cladding heat treatment, fuel form, fuel material, and gap width. Correlation of the cladding temperature and failure behavior data for several test rod designs indicated that the incipient failure threshold had a stronger dependence on the energy deposition near the fuel surface than on the radial average energy deposition. Reduction of the water/fuel ratio through the use of shroud enclosures or small clusters caused a slight reduction in the failure threshold. Prepressurization of NSRR fuel rods caused a reduction in the failure threshold for internal pressures ≥ 1.2 MPa. Rods prepressurized to 2.9 MPa failed in the range of 150 to 160 cal/g UO_2[1].

In tests with fuel rods previously irradiated to burnups of up to 32 GWd/t, rod failures occurred at lower energy depositions in some cases than similar unirradiated fuel rods, with little sensitivity attributable to the degree of burnup. The lower failure threshold was not statistically established because only a few previously irradiated rods were tested.

The consequences of unirradiated fuel rod failures were insignificant below about 300 cal/g UO_2. In the 300 to 500 cal/g UO_2 range, fuel rods were broken up and fragmented, but the resulting coolant pressures did not exceed a few MPa and nuclear-to-mechanical energy conversions did not exceed 1% of the total nuclear energy deposited.

Metal-water reaction was first detectable at about 200 cal/g UO_2 and increased to about 50% of the cladding wall thickness at 500 cal/g UO_2. Tests at over 600 cal/g UO_2 had more severe consequences, with resultant coolant pressure increases to 12 MPa, energy conversions to nearly 3%, and metal-water reaction to nearly 100% of the cladding. Pressure and mechanical energy generation were detected at lower energy depositions (≥ 200 cal/g UO_2) for preirradiated rods; however, the observed magnitudes were relatively insignificant[1].

TEST DESIGN AND CONDUCT

A new RIA fuel behavior experimental program was recently started in the Power Burst Facility (PBF) with coolant conditions typical of hot-startup conditions in a commercial boiling water reactor (BWR)[b]. To date, six tests have been completed: four single rod scoping tests (Tests RIA ST-1, RIA ST-2, RIA ST-3, and RIA ST-4) with radial average energy depositions ranging from about 220 to 560 cal/g UO_2 and two four-rod tests (RIA 1-1 and RIA 1-2) with energy depositions of approximately 320 and 200 cal/g UO_2, respectively.

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[a] All energy values presented in this paper are radial average values at the axial peak power location of the test rod at the end of the power pulse (not to be confused with energy deposition at failure). These values include an enthalpy of 15 cal/g UO_2 associated with hot-startup conditions.

[b] Coolant temperature, pressure and shroud flow rate of 538 K, 6.45 MPa nad 85 cm^3/s, respectively.

The four RIA scoping test rods were fabricated with unirradiated cladding, fresh fuel pellets and dimensions typical of pressurized water reactor (PWR) fuel. Two fuel rods previously irradiated in the Saxton[c] reactor to approximately 4.6 GWd/t burnup and two unirradiated Saxton rods were used in the RIA 1-1 test and four previously irradiated Saxton fuel rods were used in the RIA 1-2 test. One of the RIA 1-1 previously irradiated fuel rods was backfilled with a mixture of helium and argon to 0.1 MPa. The other RIA 1-1 previously irradiated fuel rod was unopened prior to testing. Based on measurements of similar rods, the internal gas pressure of helium and fission gases was about 0.1 MPa. The two unirradiated RIA 1-1 fuel rods were backfilled with helium to 0.1 MPA. Two of the RIA 1-2 previously irradiated fuel rods were backfilled with a mixture of helium and argon to 2.4 MPa, a third rod was backfilled with the same mixture of gases to 0.1 MPa, and the fourth rod was unopened prior to testing. Each 0.91 m long test rod was surrounded by a separate flow shroud with inlet orifice and inside diameter sized so that coolant behavior during the transient would be similar to that expected within a boiling water reactor (BWR) assembly during an RIA at zero power initial conditions. The annular flow area thus defined for each rod was about the same as for a rod in an 8 x 8 BWR-6 fuel bundle[d] at hot-startup conditions.

EXPERIMENTAL RESULTS

As observed in the previous SPERT, TREAT and NSRR tests, energy deposition, (thus consequent enthalpy increase) in the test fuel appears to be the single most important variable. The threshold for failure of unirradiated fuel subjected to an RIA under boiling water reactor hot-startup system conditions is about the same as was observed in SPERT and NSRR. In PBF, unirradiated rod failures occurred at about 270 cal/g UO_2 and above and did not occur at 230 cal/g UO_2.

However, the consequences of previously unirradiated fuel rod failure at boiling water reactor hot-startup system conditions appears to be more severe than observed in either SPERT or NSRR. Extensive cracking and crumbling of embrittled cladding and fuel occurred during the PBF tests, presumably during cooldown. Approximately 12 cm of the RIA ST-2 rod (270 cal/g UO_2) disintegrated into large fragments of fuel and cladding. (An 8 cm-long region of disintegrated fuel is shown in Figure 1). Another 14 cm of that rod contained the large crack shown in Figure 2 which penetrated the entire rod. Approximately 33 cm of both of the unirradiated RIA 1-1 test rods (320 cal/g UO_2) crumbled into either fine fuel powder or larger chunks of fuel and cladding. In the SPERT and NSRR tests, cladding cracks were generally evident in rods subjected to energy depositions between 240 and 300 cal/g UO_2 and the zircaloy cladding was oxidized and embrittled; however, the rods remained relatively intact. The SPERT and NSRR test rods broke into large pieces when subjected to energy depositions between 300 and 350 cal/g UO_2, but the fuel did not crumble into fine powder or somewhat larger chunks as did the RIA 1-1 rods.

Metallographic examination of the previously unirradiated PBF fuel rod cross sections revealed not only radial deformation of the cladding but extensive variation in wall thickness involving considerable plastic flow. A cross section of Rod RIA ST-1 (280 cal/g UO_2) near the peak power elevation is shown in Figure 3. This cross section reveals rod deformation, cladding fracture and fuel loss. The rod is no longer circular and has regions of cladding thickening and thinning amounting to approximately 170 and 60% of the original wall thickness, respectively. The cladding was heavily and uniformly oxidized around the circumference from both the UO_2-zircaloy reaction on the cladding inside surface and the zircaloy-water reaction on the outside surface.

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[c] A small prototype pressurized water reactor built by the Westinghouse Electric Company and located in Saxton, Pennsylvania.

[d] The coolant conditions are closely representative of the average conditions within the lead (high power) bundle of a BWR-6 at hot-startup conditions.

Cladding thickening and thinning was not reported by the SPERT or NSRR programs [1] but was observed. The uniform oxidation around the deformed cross section indicates that the cladding oxidation on the outside surface occurred after the deformation. The presence on the cladding inner surface of duplex zirconium rich and uranium-rich uranium-zirconium reaction layers and the absence of ZrO_2 indicates that the cladding oxidation on the inside surface took place with the rod intact and the UO_2 fuel in contact with the cladding[2]. In regions where the cladding has thinned, the oxidation has consumed all of the prior beta material, leaving only brittle ZrO_2 and oxygen stabilized alpha. The rod apparently fractured upon being quenched. These cladding structures (along with cladding temperature and coolant pressure measurements) suggest the following scenario: during the power burst the fuel expands out against the cladding; film boiling heat transfer is initiated and the cladding temperatures reach values near the melting point; rapid heat transfer to the coolant results in vaporization and a modest pressure pulse ($\leq$ 2 MPa) which acts on the ductile cladding to deform it into thin and thick regions; the fuel rod cladding remains in film boiling for about 20 seconds during which the cladding reacts with both the fuel and coolant (steam) and becomes heavily oxidized; and, finally, rod fracture occurs upon quench.

The fuel structure at the 0.35-m axial elevation of the previously unirradiated RIA ST-1 rod is shown in Figure 4. Grain growth and extensive grain boundary separation are shown. There is no evidence of fuel melting. The grain boundary separation (fuel powdering) probably occurred due to a combination of severe loss of grain boundary strength at elevated temperature and significant thermal stresses upon quenching from a film boiling condition[3]. As shown in Figure 3, a considerable amount of powdered fuel is missing and presumably was subsequently washed out of the rod.

The two previously irradiated Saxton fuel rods tested during the RIA 1-1 experiment exhibited even more unusual behavior. Complete flow blockage of the shroud coolant occurred within 4s after the power burst. Approximately 48 and 64 cm of those rods (subjected to an energy deposition of about 320 cal/g UO_2), disintegrated into either fine fuel powder or larger chunks of fuel and cladding. In addition, portions of the rod swelled and blocked about 80% of the flow channel. The remaining approximately 20% of the flow channel was blocked by debris (powdered fuel). A cross section of the previously irradiated Rod 801-1 and its flow shroud near the peak power elevation is shown in Figure 5. Complete channel blockage is observed. As shown in Figure 6 this blockage was formed by the swelling molten or nearly molten fuel around a fragment of pellet and cladding (Figure 6 is a view of the ground and polished portion of Figure 5 that includes the shroud and blockage). The prior molten fuel in the blockage region exhibits considerable porosity. The pellet fragment also exhibits considerable porosity as shown in the magnification of Figure 7. This porosity is attributable to the release, coalescence and expansion of gaseous and potentially volatile fission products from the fuel matrix. The porosity in Figure 7 appears to have been somewhat influenced by a temperature gradient and swept toward the outer surfaces of the pellet. The outer cladding diameters of the RIA 1-1 test rods were about 10 mm and the inner diameters of the flow shrouds were 16.3 mm. Therefore, an 80% coolant channel blockage required a fuel volume increase of about 180%. The disintegration of a significant portion of the unirradiated RIA ST-2 and RIA 1-1 test rods did not completely block their coolant channels.

The four previously irradiated Saxton fuel rods tested during the RIA 1-2 test were subjected to a peak energy deposition of approximately 200 cal/g UO_2. One of the two unpressurized RIA 1-2 rods (a rod which was not opened) remained relatively intact but failed. This rod was found to have twenty-two longitudinal cracks starting at about 18 cm and extending to about 72 cm from the bottom of the 91-cm fuel stack. The cracks had the appearance of stress-corrosion type cracks as shown in Figure 8. The total pellet energy at the

18- and 72-cm locations was about 150 cal/g UO_2. Thus it is tentatively concluded that the failure threshold during an RIA event starting from hot-startup for fuel rods with burnups of about 5000 MWd/t is approximately 150 cal/g UO_2 conditions.

The other RIA 1-2 unpressurized rod (which had been opened to insert a plenum pressure transducer) did not fail. The two prepressurized rods (2.4 MPa) also did not fail but did experience cladding ballooning with a volume increase of approximately 20%. These results suggest that previously irradiated zircaloy cladding (which has experienced fast neutron damage) is susceptible to something similar to stress corrosion cracking due to pellet-cladding interaction (PCI) when the fission product chemistry remains undisturbed. Prepressurization of the previously irradiated Saxton rods probably provided a cushion between the fuel and cladding and lowered the strain rate and opening of the rods probably also changed the fission product chemistry and thereby decreased the propensity for PCI failure during an RIA.

CONCLUSIONS

The failure threshold of previously irradiated LWR type fuel rods subjected to an RIA with coolant conditions typical of hot-startup conditions in a commercial boiling water reactor is about 150 cal/g. The failure mechanism appears to be stress-corrosion cracking due to pellet cladding interaction. The failure threshold of unirradiated rods is between 230 and 270 cal/g. Unirradiated rods subjected to 270 to 280 cal/g experienced extensive cracking and crumbling apparently due to cladding wall thickness variations and complete oxidation of the thin sections. Previously irradiated test rods which were subjected to an energy insertion of 320 cal/g UO_2 swelled and blocked the coolant flow channel. Unirradiated test rod damage at an energy insertion of 320 cal/g was extensive but did not result in coolant flow blockage.

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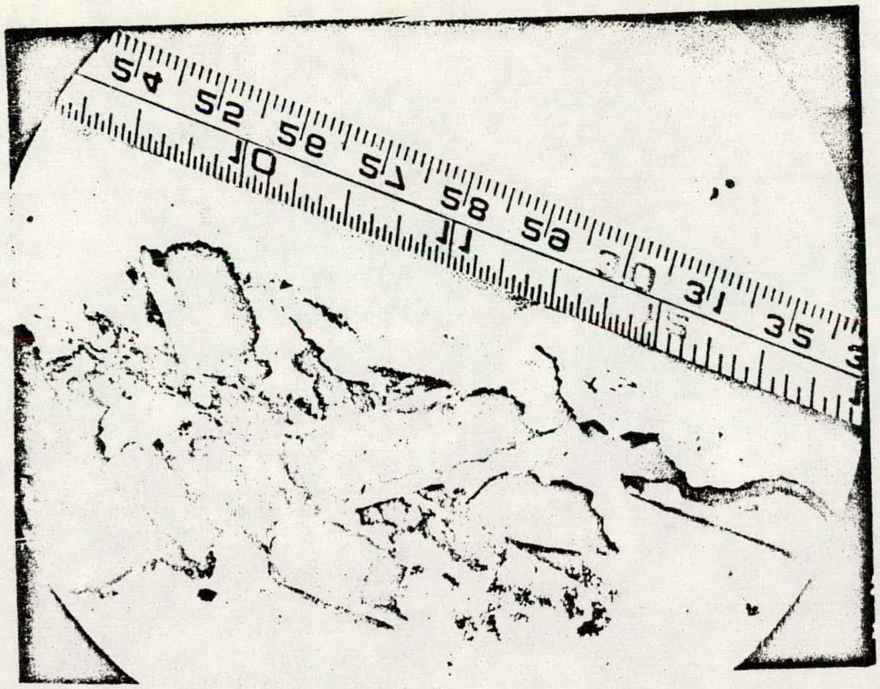

Fig. 1 Peak flux region of RIA-ST-2 rod

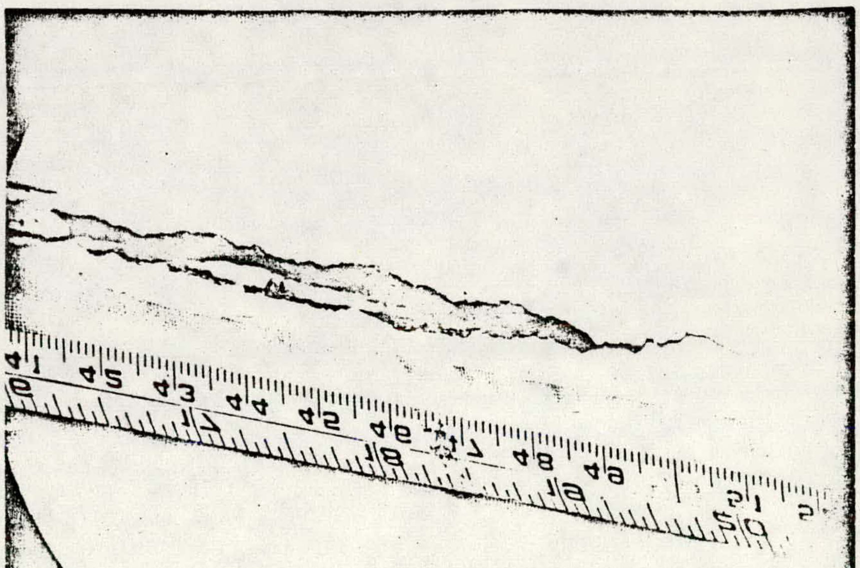

Fig. 2 RIA-ST-2 fuel rod above the flux peak

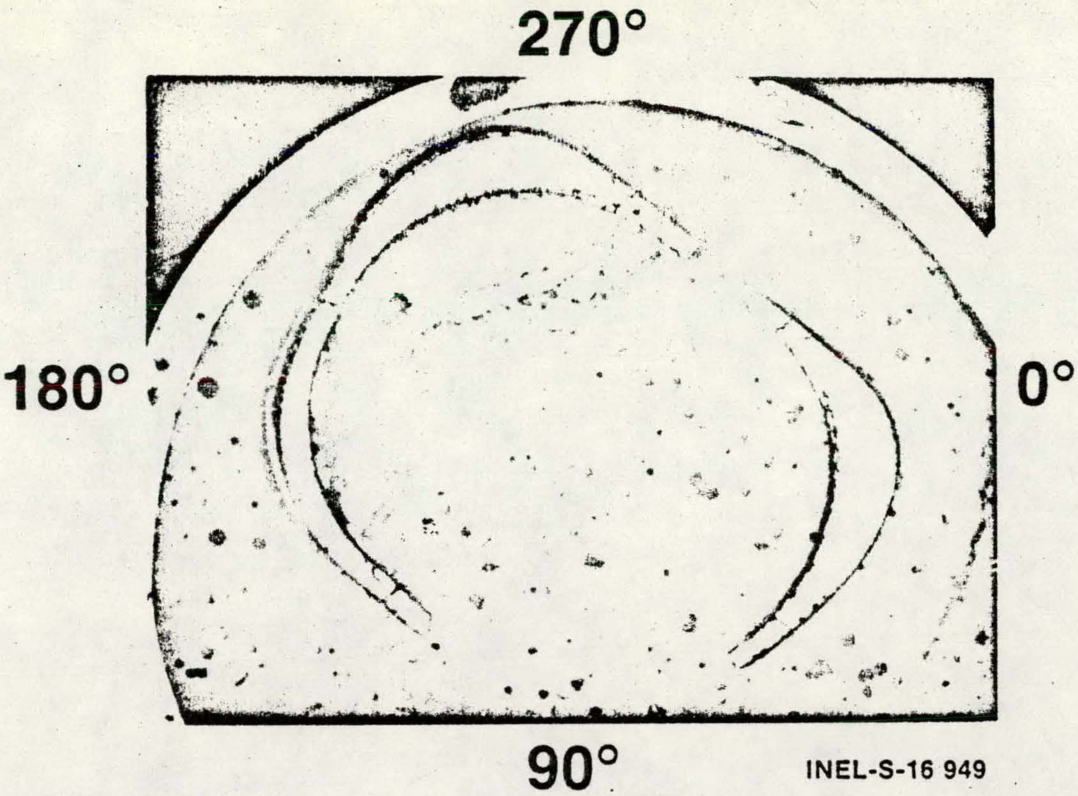

Fig. 3 RIA Scoping Test 1 - Cladding near the peak flux location (0.35-m elevation)

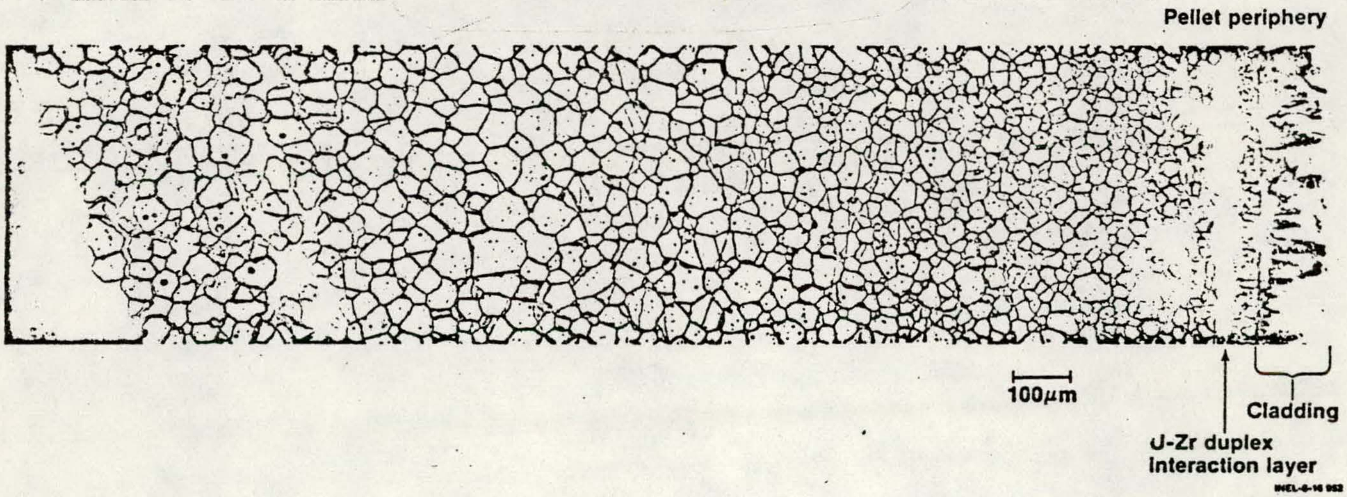

Fig. 4 RIA Scoping Test 1 - Fuel shattering at the 0.35-m elevation, 240° orientation

Fig. 5 Cross section of the previously irradiated Rod 801-1 fuel, cladding, and shroud near the peak power elevation (Test RIA 1-1)

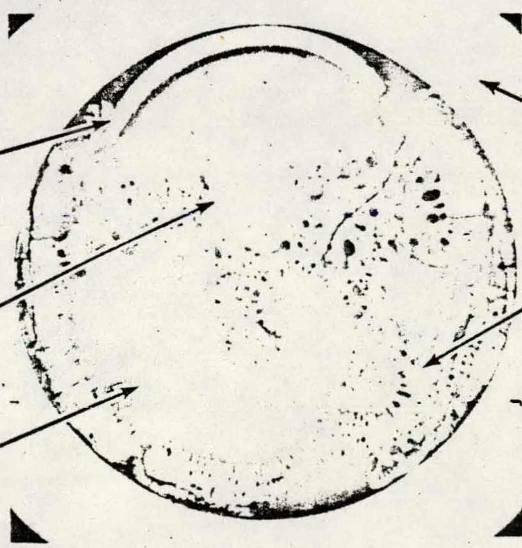

Fig. 6 Cross section of the previously irradiated Rod 801-1 fuel, cladding, and shroud near the peak power elevation (Test RIA 1-1)

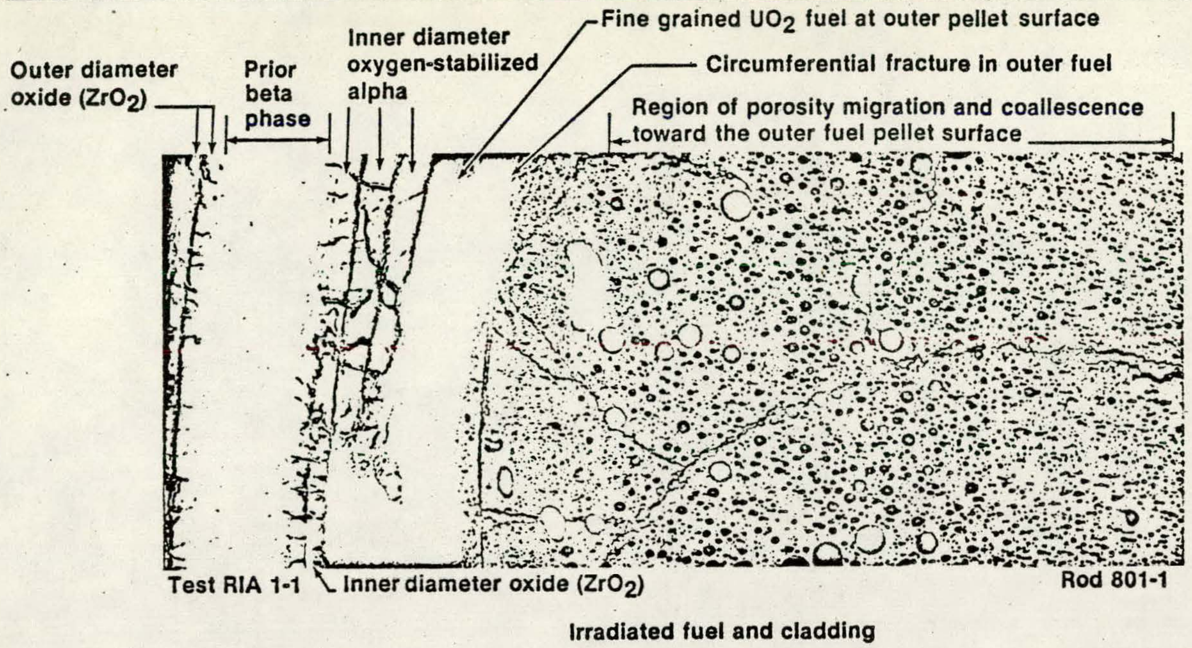

Fig. 7 100x view of the previously irradiated Rod 801-1 cladding and pellet fragment in the blockage region near the peak power elevation (Test RIA 1-1)

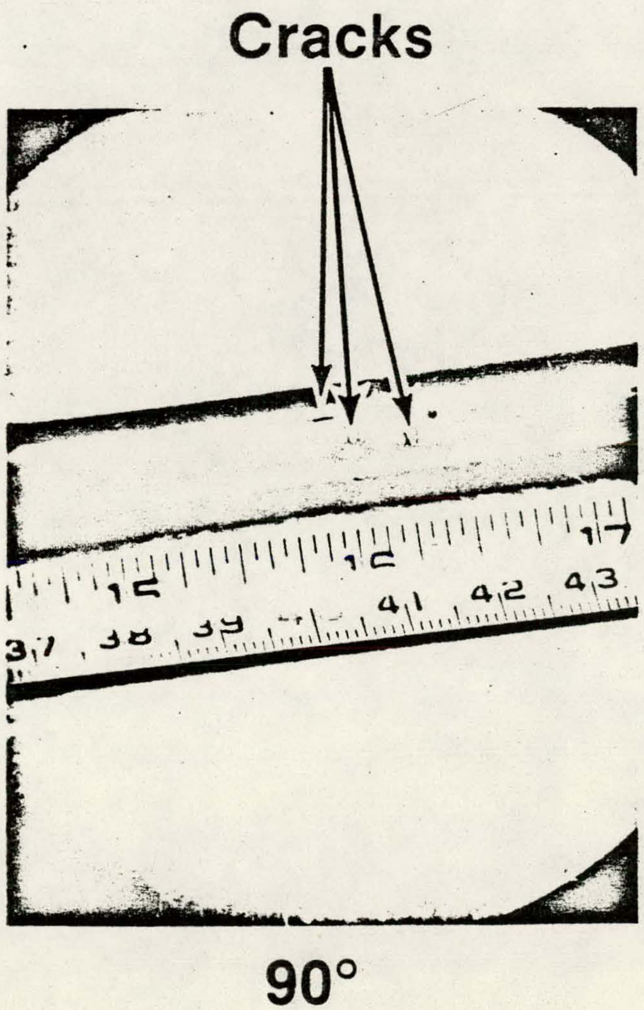

Fig. 8 Test RIA 1-2, Rod 2-3, cladding cracks