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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 Special Power Excursion Reactor Test (SPERT), and Japanese Nuclear Safety Research Reactor (NSRR) tests. 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 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 SPERT and NSRR 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 appear to be more severe than previously observed in either the stagnant capsule SPERT or NSRR tests. Metallographic examination of both previously unirradiated and irradiated PBF fuel rod cross sections revealed extensive variation in cladding wall thicknesses (involving considerable plastic flow) and fuel shattering along grain boundaries in both restructured and unrestructured fuel regions. Oxidation of the cladding resulted in fracture at the location of cladding thinning and disintegration of the rods during quench. 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 coolant channels within the flow shrouds surrounding the fuel rods.

Réacteurs à eau ordinaire: Comportement du combustible au cours d'expériencés d'Accident d'Insertion de Réactivite

RÉSUMÉ

On compare les résultats de six essais d'Accident d'Insertion de Réactivité (RIA), réalisés dans Power Burst Facility (PBF), aux résultats de tests plus anciens effectués dans Special Power Excursion Reactor Test (SPERT) et Nuclear Safety Research Reactor (NSRR - Japon). Dans PBF, le programme expérimental d'étude du comportement d'un combustible lors d'un RIA est réalisé en partant des conditions de refroidissement représentatives d'un départ à chaud dans un BWR commercial. Les programmes SPERT et NSRR avaient étudié des crayons combustibles ou de petits assemblages (de la filiére à eau ordinaire), enfermés dans une capsule étanche, remplie d'eau à température ordinaire et pression atmosphèrique. Comme on l'avait observé lors des tests SPERT et NSRR, la déposition d'énergie, et par conséquent l'accroissemment d'enthalpie du combustible, est la variable principale. Cependant, les conséquences de la rupture (d'un combustible) aux conditions de démarrage à chaud d'un BWR semblent plus graves que celles observées dans les tests SPERT et NSRR (capsules fermées). L'examen métallographique des sections de combustible, pré-irradié ou non, a montré de grandes variations d'épaisseur de gaine (supposant une déformation plastique importante) et l'éclatement du combustible tout le long des joints de grain, à la fois dans les zones restructurées ou non. L'oxydation de la gaine a provoqué sa rupture dans les zones amincies et la désintegration des crayons pendant le renoyage. De plus, l'expansion des produits de fission gazeux ou volatils a provoqué une augmentation de volume de 180% et l'obturation des canaux de refroidissement, à l'intérieur des chemises qui entourent les crayons combustibles.

VERHALTEN VON LEICHTWASSERREAKTORBRENNSTOFF
BEI REAKTIVSTAETSSTOERFAELLEN

ZUSAMMENFASSUNG

Ergebnisse von sechs Experimenten, die neulich in Power Burst Facility (PBF) unter den Bedingungen eines Reaktivitaetsstoerfalles (RIA) durchgefuehrt wurden, werden mit entsprechenden Ergenbnissen von Versuchen in Special Power Excursion Reactor (SPERT) und in Japanischem Nuclear Safety Research Reactor (NSRR) verglichen. Das RIA Versuchsprogram, das vor kurzem in PBF begonnen wurde, verwendet Bedingungen, die fuer heissen Nulleistungszustand in einem Siedewasserreaktor typisch sind. In den SPERT und NSRR Programmen wurde das Brennstoffverhalten unter Bedingungen untersucht, die durch Einzelstaebe oder kleine Buendelgroesse und stagnierendes Kuehlwasser bei Zimmertemperatur ohne Ueberdruck in abgeschlossenen Kapseln gekennzeichnet sind. Wie schon in den SPERT und NSRR Versuchen festgestellt wurde, erscheint die Energiefreisetzung und folglich die Enthalpieerhoehung im Brennstoff die wichtigste Kenngroesse zu sein. Zusaetzlich deuten die PBF Versuchsergebnisse darauf hin, dass der heisse reaktortypische Nulleistungszustand zu folgeschwereren Brennstoffschaeden fuehrt als bisher in den Kapselversuchen in SPERT und NSRR beobachted wurde.

Metallographische Untersuchungen an sowohl vorbestrahlten als auch nicht vorbestrahlten Brennstabquerschnitten haben erhebliche Aenderungen in der Wandstaerke (bedingt durch plastisches Fliessen) gezeigt und Pulverisieren des Brennstoffs in sowohl umstrukturiertem als auch unbeeinflusstem Brennstoff. Oxidation von Zirklaloy hat das Aufbrechen des Huellrohres an den duennen Stellen bewirkt und Desintegration der Staebe nach der Wiederbenetzung. Zusaetzlich wurde Brennstoffschwellen, bedingt durch gasfoermige Spaltprodukte in vorbestrahltem Brennstoff, beobachtet, das eine Volumszunahme von bis zu 180% bewirkte und eine vollstaendige Blockade des die Brennstaebe umhuellenden Kuehlmittelfuehrungsrohres nach sich zog.

INTRODUCTION AND REVIEW OF PREVIOUS WORK

The rapid inadvertent insertion of reactivity into a light water reactor core, leading to high cladding temperatures, 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 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) test program, which investigated the behavior of single or small clusters of fuel rods under approximate room temperature and atmospheric (or near atmospheric) pressure conditions, no forced coolant flow, and zero initial power. Similar tests have been performed in the Japanese Nuclear Safety Research Reactor[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 fuel-cladding gap width. Correlation of the cladding temperature and failure behavior data for

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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). The PBF values include an enthalpy of 15 cal/g UO_2 associated with the BWR hot-startup conditions of the PBF tests.

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 greater than 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 ($\geq$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 with coolant conditions typical of hot-startup conditions in a commercial boiling water reactor (BWR)[a]. 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[2], and two four-rod tests (Tests RIA 1-1 and RIA 1-2) with energy depositions of approximately 320 and 200 cal/g UO_2, respectively[3]. The reactor periods for these tests ranged from 5.7 to 3.1 ms.

The PBF core is a right-circular annulus 1.3 m in diameter and 0.91 m in length, enclosing a centrally located vertical flux trap 0.21 m in diameter. An in-pile tube fits in this vertical test space and contains the test train assembly. The in-pile tube is a thick-walled, Inconel 718, high strength pressure tube designed to contain the steady state operating pressure and any pressure surges from test fuel rod failures. A flow tube is positioned inside the in-pile tube to direct the coolant flow. Coolant flow enters the top of the in-pile tube above the reactor core and flows down the annulus between the in-pile tube wall and the flow tube. The flow reverses at the bottom, passes up through the test train, and exits above the reactor core at the in-pile tube outlet. A loop coolant system provides cooling water for the in-pile tube at controllable pressures, temperatures, and flow rates. For these tests, this system simulated the hot-startup coolant conditions of a BWR.

The four RIA scoping test rods were fabricated with unirradiated zircaloy-4 cladding, fresh uranium dioxide fuel pellets, and dimensions typical of pressurized water reactor (PWR) fuel. For each of the four scoping tests, a single fuel rod was positioned in a separate

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

flow shroud and then installed in the center of the in-pile tube. Figure 1 presents an illustration of the RIA Scoping Test hardware within the PBF in-pile tube.

Two fuel rods previously irradiated in the Saxton[a] reactor to approximately 5 GWd/t burnup and two unirradiated Saxton rods were used in Test RIA 1-1, and four previously irradiated Saxton rods with about the same burnup were used in Test RIA 1-2. The four separately shrouded fuel rods used in Tests RIA 1-1 and RIA 1-2 were symmetrically positioned within the PBF in-pile tube as shown in Figure 2. 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 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 at hot-startup conditions[b].

The PBF four-rod test train was designed such that the azimuthal power distribution within each test rod was approximately uniform. Cobalt flux wires and self-powered neutron detectors were utilized to measure the integrated and instantaneous neutron fluxes, respectively. The calorimetric measurements used to determine absolute rod power included coolant temperature, coolant temperature rise, coolant pressure, and coolant flow rate. In addition, internal rod pressure, plenum temperature, fuel centerline temperature (unirradiated rods only), cladding surface temperature, and cladding axial elongation were measured for selected test fuel rods.

The two unirradiated Test RIA 1-1 fuel rods and one of the previously irradiated rods were backfilled with helium or a mixture of

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a. A small prototype pressurized water reactor built by the Westinghouse Electric Corporation and located in Saxton, Pennsylvania.

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

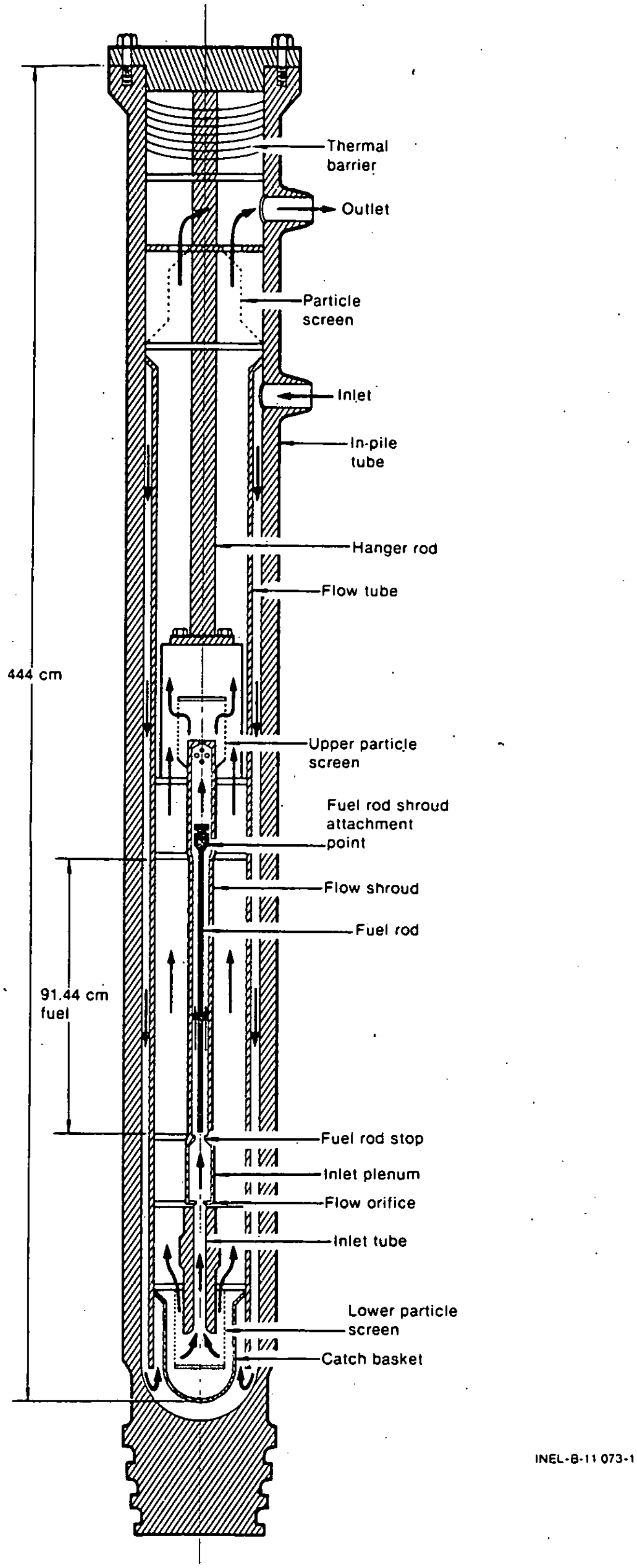

Fig. 1 Illustration of the RIA Scoping Test hardware within the PBF in-pile tube.

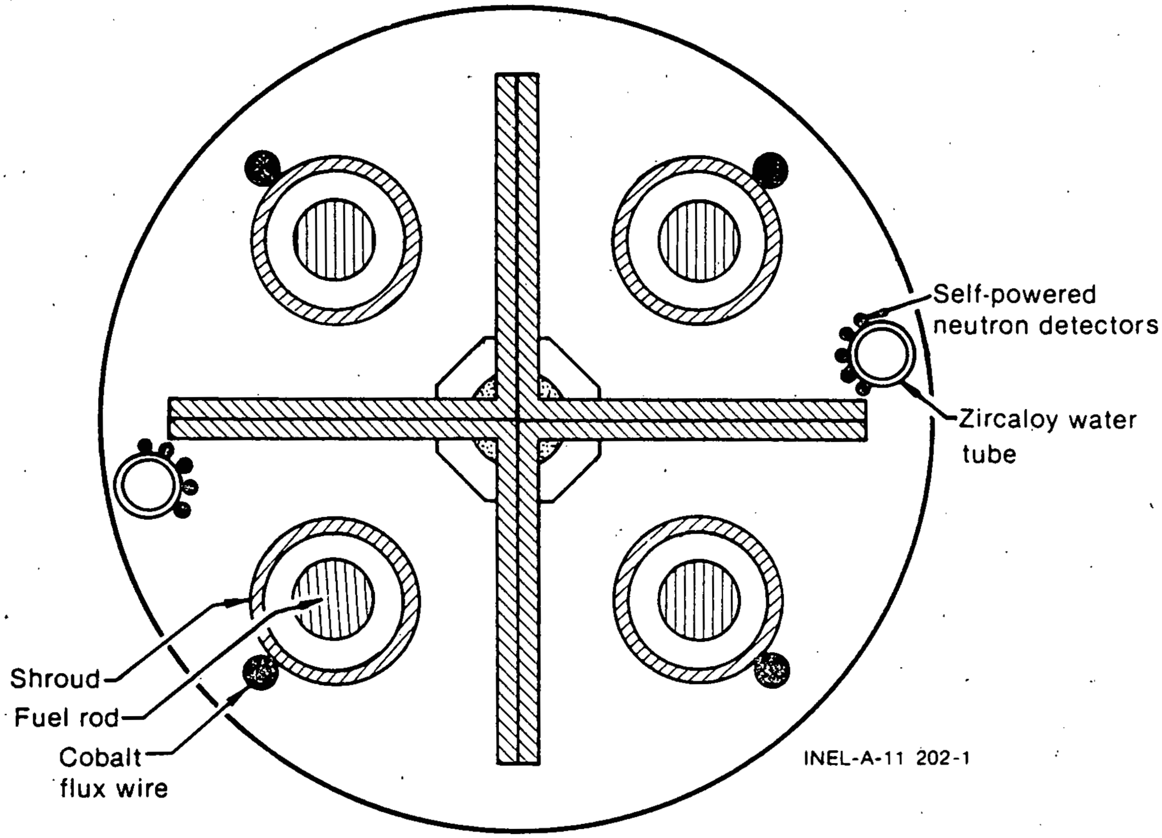

Fig. 2 Tests RIA 1-1 and RIA 1-2 fuel rod configuration.

helium and argon to 0.1 MPa. The other Test RIA 1-1 irradiated fuel rod was unopened prior to testing. On the basis of measurements of similar rods, the internal gas pressure within this rod was also about 0.1 MPa. Two of the Test RIA 1-2 previously irradiated fuel rods were backfilled with a mixture of helium and argon to 2.3 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. The specific burnups and rod internal pressures of the Tests RIA 1-1 and RIA 1-2 rods are summarized in Table I.

TABLE I

TESTS RIA 1-1 AND RIA 1-2 FUEL ROD CHARACTERISTICS

Test	Rod	Burnup (MWd/t)	Internal Pressure (MPa)	Rod Opened
RIA 1-1	801-1	4600	0.1	Yes[a]
RIA 1-1	801-2	4650	0.1	No
RIA 1-1	801-3	0	0.1	
RIA 1-1	801-4	0	0.1	
RIA 1-2	802-1	5200	0.1	Yes[a]
RIA 1-2	802-2	5100	2.4	Yes[a]
RIA 1-2	802-3	4450	0.1	No
RIA 1-2	802-4	4550	2.4	Yes[a]

a. Refilled with a mixture of helium and argon with approximately the same thermal conductivity as the gases originally within the rod.

EXPERIMENTAL RESULTS

As observed in the previous SPERT and NSRR tests, energy deposition (and 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 BWR hot-startup system conditions is about the same as was observed in SPERT and NSRR. In the PBF, unirradiated rod failures occurred at about 270 cal/g UO_2 and above, and did not occur at 230 cal/g UO_2.

The consequences of previously unirradiated fuel rod failure at boiling water reactor hot-startup system conditions in the PBF appear 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

Test 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 3). Another 14 cm of that rod contained a large crack, shown in Figure 4, which penetrated the entire rod. Approximately 33 cm of both of the unirradiated Test RIA 1-1 rods (320 cal/g

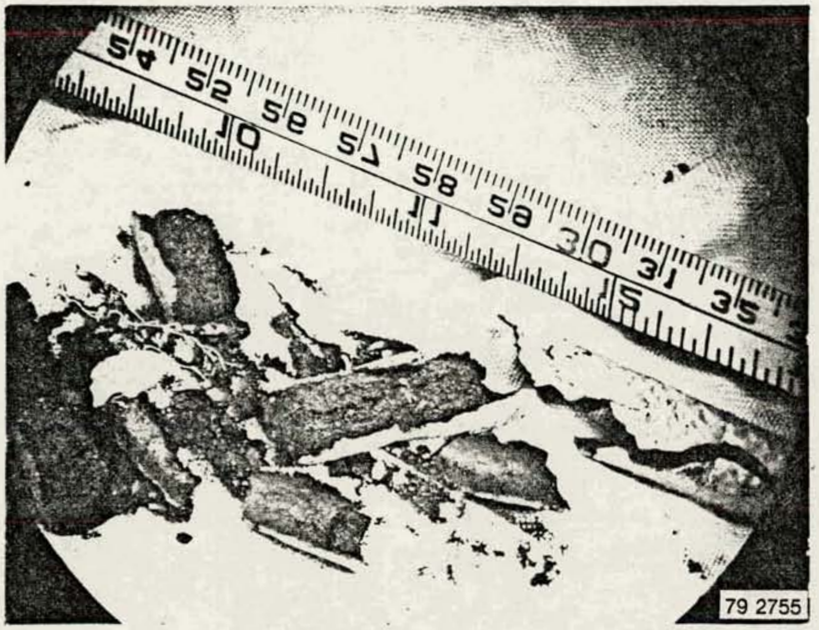

Fig. 3 Peak flux region of Test RIA-ST-2 fuel rod.

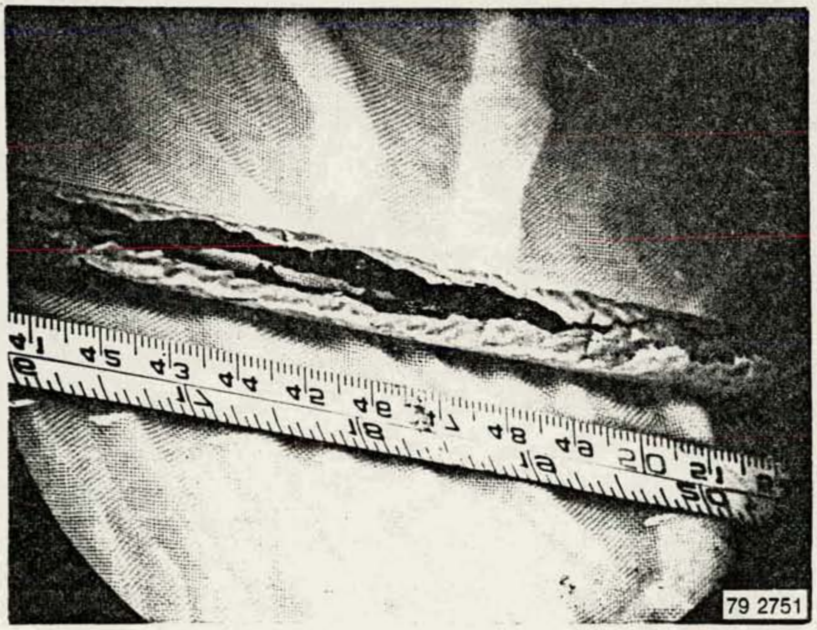

Fig. 4 Test RIA-ST-2 fuel rod above the flux peak.

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 and somewhat larger chunks as did the fuel in the Test 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 the Test RIA-ST-1 rod (280 cal/g UO_2) near the peak power elevation is shown in Figure 5.

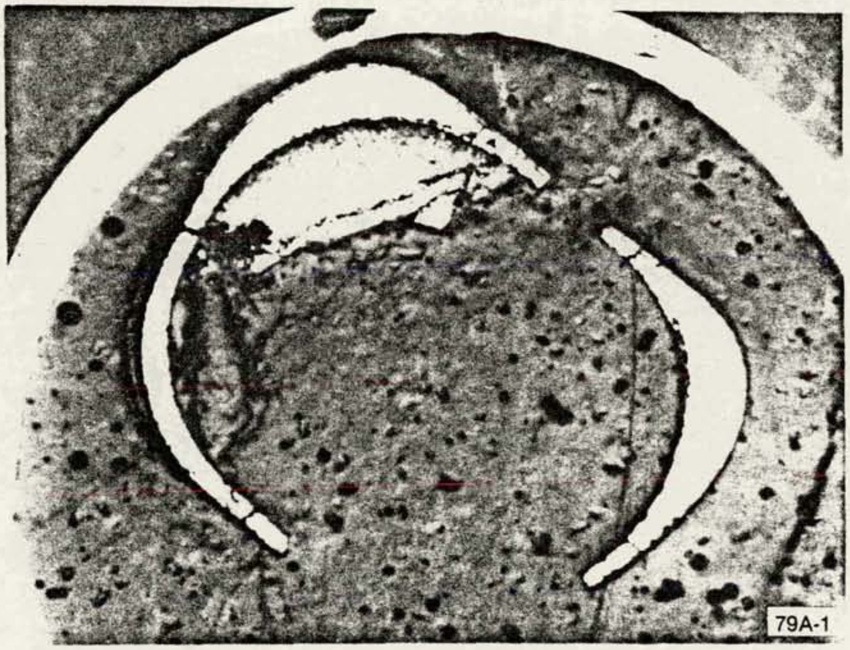

Fig. 5 Test RIA-ST-1 fuel rod cladding near the peak flux location (0.35-m elevation).

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. Cladding thickening and thinning was not reported by the SPERT program[1], but it 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[4]. In regions where the cladding had thinned, the oxidation had 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 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 ($\sim$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 Test RIA-ST-1 rod is shown in Figure 6. 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[5]. As shown in Figure 5, a considerable amount of powdered fuel is missing and presumably was washed out of the rod after the rod was quenched and fractured.

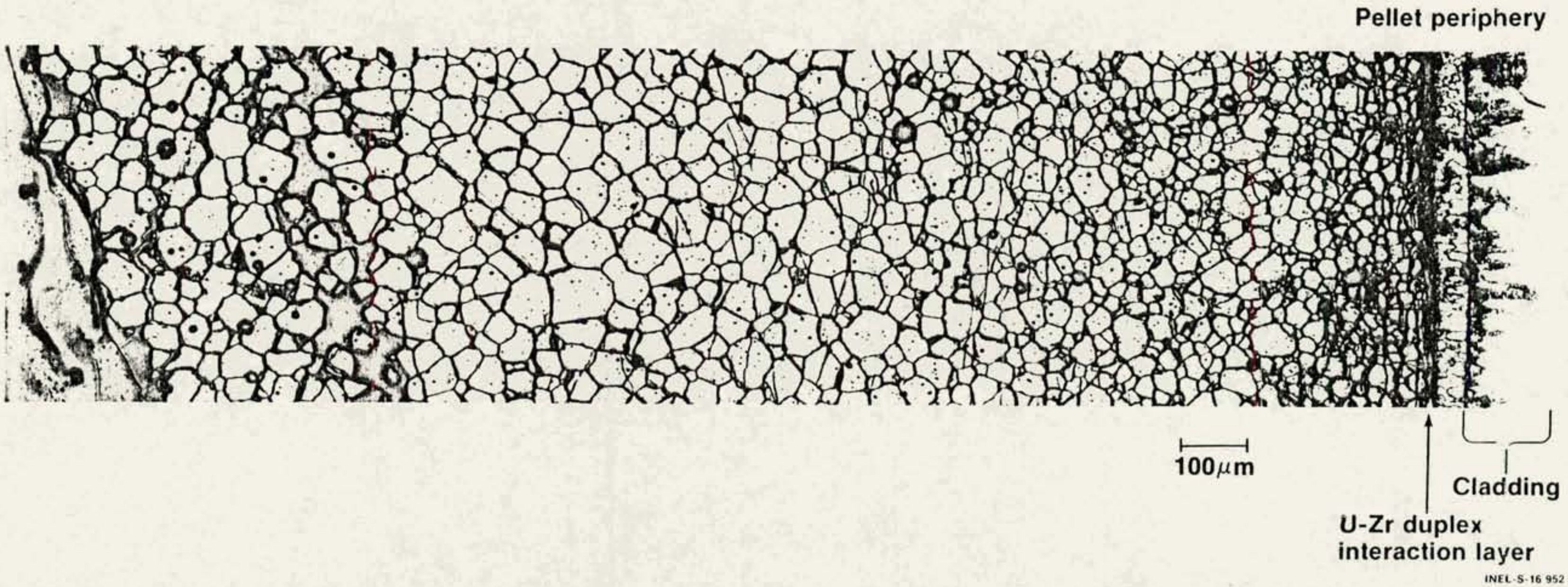

Fig. 6 Test RIA-ST-1 fuel shattering at the 0.35-m elevation, 240-degree orientation.

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 4 seconds after the power burst. Approximately 48 and 64 cm, respectively, 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 rods swelled and blocked about 80% of the flow channel. The remaining 20% of the flow channel was blocked by debris (powdered fuel). A cross section of previously irradiated Rod 801-1 and its flow shroud near the peak power elevation is shown in Figure 7. Complete coolant channel blockage is observed. As shown in Figure 8, this blockage was formed by the swelling molten,

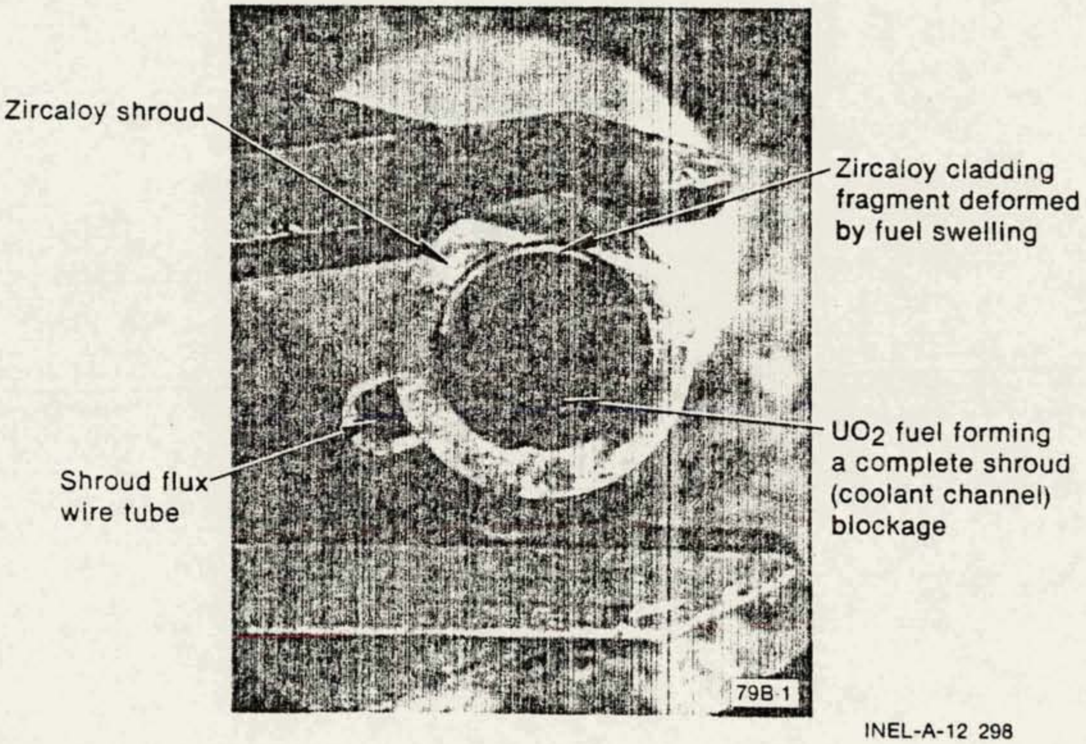

Fig. 7 Cross section of previously irradiated Rod 801-1 fuel, cladding, and shroud near the peak pwer elevation (Test RIA 1-1).

or nearly molten, fuel around a fragment of pellet and cladding (Figure 8 is a view of the ground and polished portion of Figure 7 that includes the shroud and blockage). The prior molten fuel in the blockage region exhibited considerable porosity. The pellet fragment

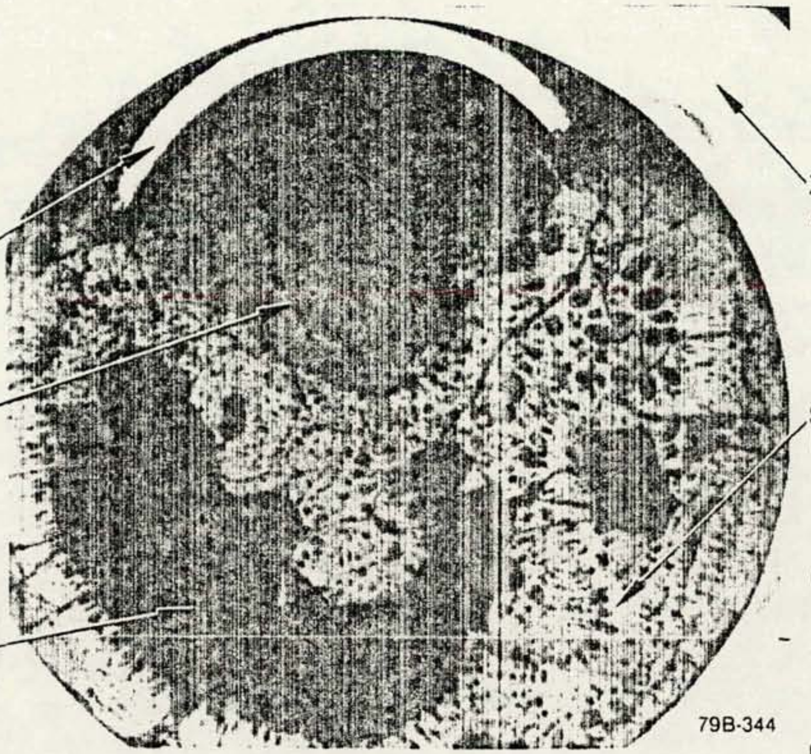

Fig. 8 Ground and polished cross section of Rod 801-1 fuel, cladding, and shroud near the peak power elevation (Test RIA 1-1).

also exhibited considerable porosity, as shown in the magnification of Figure 9. 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 9 appears to have been influenced somewhat by a temperature gradient and swept toward the outer surfaces of the pellet. The outer cladding diameters of the Test RIA 1-1 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 Test RIA-ST-2 and Test RIA 1-1 rods did not result in complete blockage of their coolant channels.

The four previously irradiated Saxton fuel rods tested during the RIA 1-2 experiment were subjected to a peak energy deposition of approximately 200 cal/g UO_2. One of the two unpressurized Test

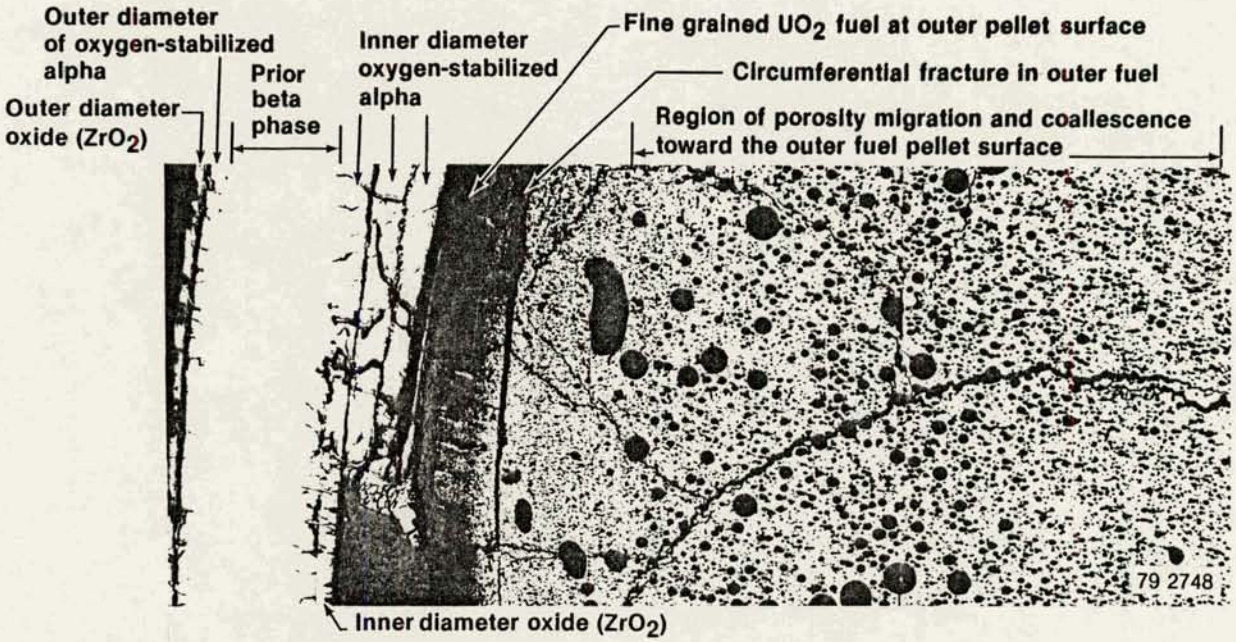

Fig. 9 Previously irradiated Rod 801-1 cladding and pellet fragment in the blockage region near the peak power elevation (Test RIA 1-1).

RIA 1-2 rods (Rod 802-3 which was not opened) remained relatively intact but failed. This rod was found to have 22 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 10. 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 conditions, for fuel rods with burnups of about 5000 MWd/t is approximately 150 cal/g UO_2.

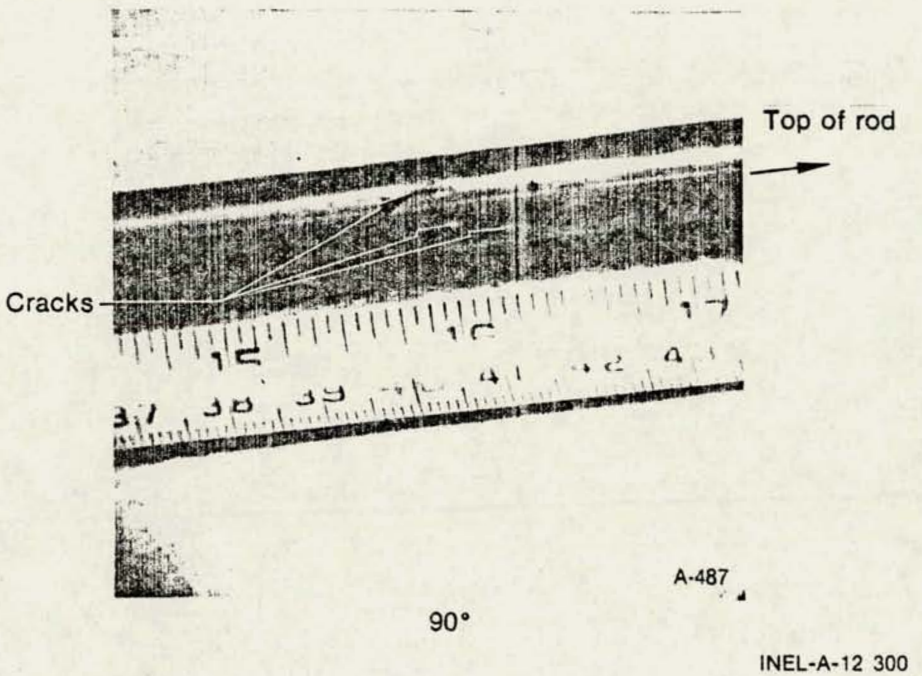

Fig. 10 Cracks in the cladding of Rod 802-3, Test RIA 1-2.

The other Test RIA 1-2 unpressurized rod (which had been opened to insert a plenum pressure transducer) did not fail. The two pre-pressurized 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 UO_2[a]. 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 UO_2. These values are consistent with previous SPERT and NSRR work. Unirradiated rods subjected to 270 to 280 cal/g UO_2 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 UO_2 was extensive, but did not result in coolant flow blockage.

5. REFERENCES

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2. C. L. Zimmermann et al, Experiment Data Report for Test RIA-ST (Reactivity Initiated Accident Test Series), NUREG/CR-0473, TREE-1235 (March 1979).

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a. The energy insertion measurements reported for these tests were obtained using five independent techniques. There is some discrepancy between the values determined by these various techniques for a few of the tests. These discrepancies are not understood at this time and will be further evaluated. This may result in some minor changes in the above energy values when the final reports are issued.

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