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TREAT SODIUM LOOP EXPERIMENTS ON PERFORMANCE OF UNBONDED, UNIRRADIATED EBR-II MARK I FUEL ELEMENTS

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Description of sodium loop experiments performed in the Transient Reactor Test (TREAT) Facility specifically to provide guidance for analyses on the behavior of an element which either has no sodium thermal bond between fuel and cladding or which, due to some unspecified defect, loses the bond upon reactor startup.

Transient heat transfer calculations of the experiments were performed utilizing an idealized model of meltdown in which fuel slumping against the cladding was assumed to occur instantaneously along the length of the pin when the fuel surface at the axial midplane reached 1050°C.

The result of TREAT experiments and of the calculations imply that although fuel melting may accompany a loss-of-bond type failure in an EBR-II driver fuel element, the short term consequences are acceptable from the viewpoint of reactor safety. Comparison of results from calculations and experiment indicate that the relatively simple modelling used represents a conservative description of the temperature rise in the cladding following fuel-cladding contact resulting from the fuel melting.

1. Introduction

Although the conservatism of fuel design and reactor operation limits – and the reliability of fuel fabrication and inspection – has been demonstrated by the fact that more than 40 000 Experimental Breeder Reactor II (EBR-II) metallic driver fuel elements were irradiated in the EBR-II core before the first indication of a defect, this experience does not relieve safety evaluations from the need to consider the possible consequences of a defective driver fuel element being loaded into the core. Accordingly, sodium loop experiments are being performed in the Transient Reactor Test (TREAT) Facility specifically to provide guidance for analyses on the behavior of an element which either has no sodium thermal bond between fuel and cladding or which, due to some unspecified defect, loses the bond upon reactor startup. Two such experiments have been performed, ID-RP-1 and ID-RP-2. (Preliminary accounts have been made [1–3].)

Previous TREAT sodium loop experiments and their analyses had provided a model describing potential failure mechanisms and their thresholds for fuel elements [4]. Earlier experiments had supplied basic rate data for attack of cladding by molten fuel [5], and TREAT in-pile tests confirmed the application of these rate data to both bonded and unbonded fuel elements under conditions of transient nuclear heating [6].

Calculations performed prior to ID-RP-1 had indicated that exposure of an unbonded EBR-II element, inside a TREAT loop, under EBR-II-like conditions would cause some degree of fuel melting but would not produce a cladding failure [3]. Because of the lack of a detailed model describing the microscopic kinetics of the incident, uncertainties over the role played by transient thermal stresses, etc., experimental verification was sought.

2. Experimental

Each experimental sample consisted of an unirradiated 6% enriched EBR-II fuel element surrounded

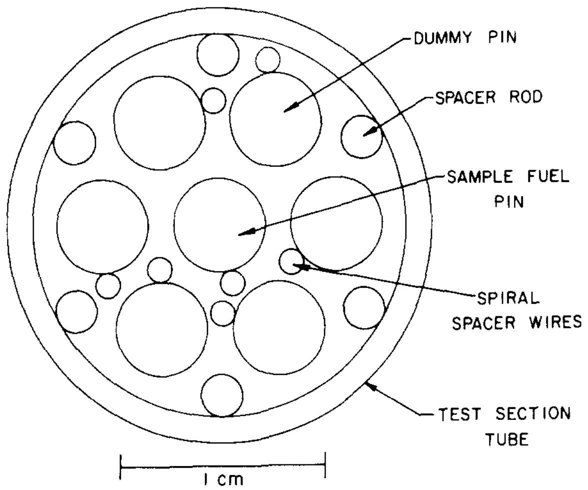

Fig. 1. Transverse cross section through sample, ID-RP-1 and ID-RP-2 (112-5637).

by a ring of six dummy elements on an EBR-II lattice pitch as shown by transverse cross section in fig. 1. Each fuel element contained one atm of argon at room temperature, with no sodium bond, thus providing a limiting case simulation of complete expulsion of bond sodium. The fuel pin was a cylinder of standard EBR-II alloy of uranium-5 wt o/o fissium [7], 0.366 cm in diameter by 35.6 cm long, clad by a 0.022 cm thick type 304 stainless steel jacket of 0.442 cm outer diameter. Each dummy element consisted of a cladding tube, with end fittings, filled with one atm of argon at room temperature. The EBR-II lattice spacing was maintained by standard 0.10 cm diameter EBR-II spiral spacer wires which were wound around each cladding tube. A stainless steel sheathed chromel-alumel thermocouple, 0.10 cm in diameter, was positioned in the coolant channel adjacent to the fuel at both top and bottom of the sample. The samples were contained in a standard Mark I integral sodium loop [8,9]. Because the effective flow area per gram of fuel is approximately twice that in a true cluster, the flow rate was set at 4 meters/sec through the test section.

ID-RP-1 was planned as a survey test corresponding, with a modest degree of conservatism, to normal EBR-II operation. ID-RP-2 was deliberately run to appreciably more severe levels of coolant inlet temperature and fuel alloy power level. Experimental coolant temperature rises were not severe but the coolant temperature levels were comparable to, or in excess of, reactor temperatures. A summary of experimental conditions is given in table 1. For comparison, the table includes ID-RP-1 conditions and EBR-II operating parameters for the central (i.e., highest power position) pin under run 27A conditions. Each sample was subjected to a "flattop" TREAT transient [10] in which reactor power is sustained, after reaching the temperature limit corresponding to the initial reactivity insertion, by a preprogrammed withdrawal of a control rod. "Flattop duration" given in the table is the interval between the time of the temperature-limited peak and the reactor scram. In the case of TREAT, "reactor power"

Table 1
Experimental conditions.

Parameter	ID-RP-1	ID-RP-2	Central pin EBR-II, Run 27A
Flattop duration	6.9 sec	6.2 sec	–
Power	57 MW	90 MW	45 MW
Reactor average	(TREAT)	(TREAT)	(EBR-II)
Pin	3100 W/cm^3	4900 W/cm^3	2680 W/cm^3
Coolant			
Inlet initial	390°C	435°C	374°C
Inlet final	430°C	511°C	–
Temperature rise	100°C *	116°C *	123°C

* Maximum

is the time-averaged power over this duration. The pin average power is the axial average, whether in TREAT or in EBR-II, given per cm^3 of fuel using 17.5 grams/cm^3 room temperature density. Two inlet coolant temperatures are given for each experiment, since the loops are package loops with no external heat exchange capability. As indicated in table 1, ID-RP-1 sample power can be related to power of the central pin in EBR-II for EBR-II operation at 52 MW; ID-RP-2 sample power can be related to 85 MW operation of EBR-II.

Sample power in both TREAT experiments was shaped axially to a chopped cosine distribution by inclusion of sheet tantalum neutron absorbers wrapped around the loop test section. The axial max-to-average power for this arrangement is 1.1. (For EBR-II, the ratio is 1.15.) The ratio of sample power to TREAT power and the axial sample power shape were determined from radiochemical calibration runs performed during the earlier TREAT loop experimentation [4].

3. Results

No anomalies were found in the transient instrument record from either ID-RP-1 or -2. Instrumentation included test section inlet flow, inlet pressure, inlet coolant temperature, and outlet coolant temperature. A summary of results follows for the two samples:

ID-RP-1. No cladding damage was apparent upon removal of the element from the loop. No distortion of the circular cladding cross section was found, and the dimensions of the cladding were still within EBR-II specifications. Appearance of the pin after cladding removal is shown in fig. 2. A portion of a meterstick is included in the figure for size comparison. Fuel had clearly slumped and flowed before being cast inside the cladding. The figure shows evidence of shiny fuel-clad eutectic alloy where molten fuel issued through fissures in an oxide coating, or "bag", around the fuel alloy and reacted with the cladding. But, the corresponding attack of the 0.022 cm thick cladding was less than about 0.001 cm. The fuel, originally a cylinder of 0.366 cm diameter, had slumped out to the cladding over much of the pin length (see fig. 2), with a nominal post-experiment diameter of 0.390 cm. Metallographic examination disclosed the fuel to have a relatively uniform fine-grained (about 25 microns) matrix, characteristic of molten alloy cooled rapidly through its solidification range. This is different from production pins which have a grain size varying from approximately 25 microns at the top to about 100 microns or larger at the bottom. Shrinkage voids and irregular surfaces gave additional evidence of fuel melting and rapid solidification.

ID-RP-2. The modified fast neutron hodoscope [11] was used to take pre- and post-transient scans of prompt fission neutron emission from the loop test section region. These measurements, performed *in situ* before the loop was moved, showed no horizontal fuel movements had occurred as a result of the meltdown transient, within an estimated uncertainty of ± 0.025 cm. However, the signals from the top two hodoscope channels (which covered a section region extending downward 4 cm from the pre-transient location of the top of the fuel) were about half the pretransient levels, indicating some fuel slumping. More detail was obtained from neutron radiographs taken at the TREAT Neutron Radiographic Facility after the loop was removed from the core. Fig. 3 shows two neutron radiographs of the test section; one extending to the inlet and one extending to the outlet. The top of the fuel has slumped approximately 2 cm, and there are clear indications of voids in the fuel column. Greater detail is shown in the X-ray photograph, fig. 4, which was taken in the Argonne, Illinois site hot laboratories after the sample was removed from the loop. The X-ray shows extensive regions where the fuel has flowed or slumped against

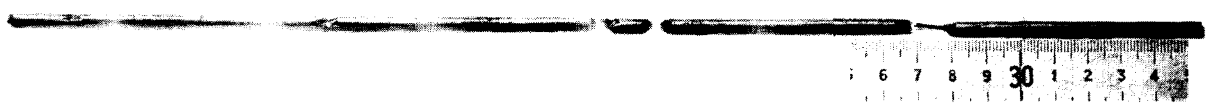

Fig. 2. Post-experiment condition of ID-RP-1 sample after removal of cladding (112-8307).

a Stepped tantalum rings for axial shaping of sample neutron flux.
b Top end fuel motion restrainer.
c Post-transient location of top of fuel.
d Post-transient location of fuel void.
e Loop wall.
f Fuel sample.
g Dummy pins in ring surrounding sample.
h Flange for inlet pressure transducer.
i Wire pin holding bottom end plug spade extension of the pins.
j "Spade" extension of bottom end plug of pins.
k Typical test section electrical pre-heater wire.

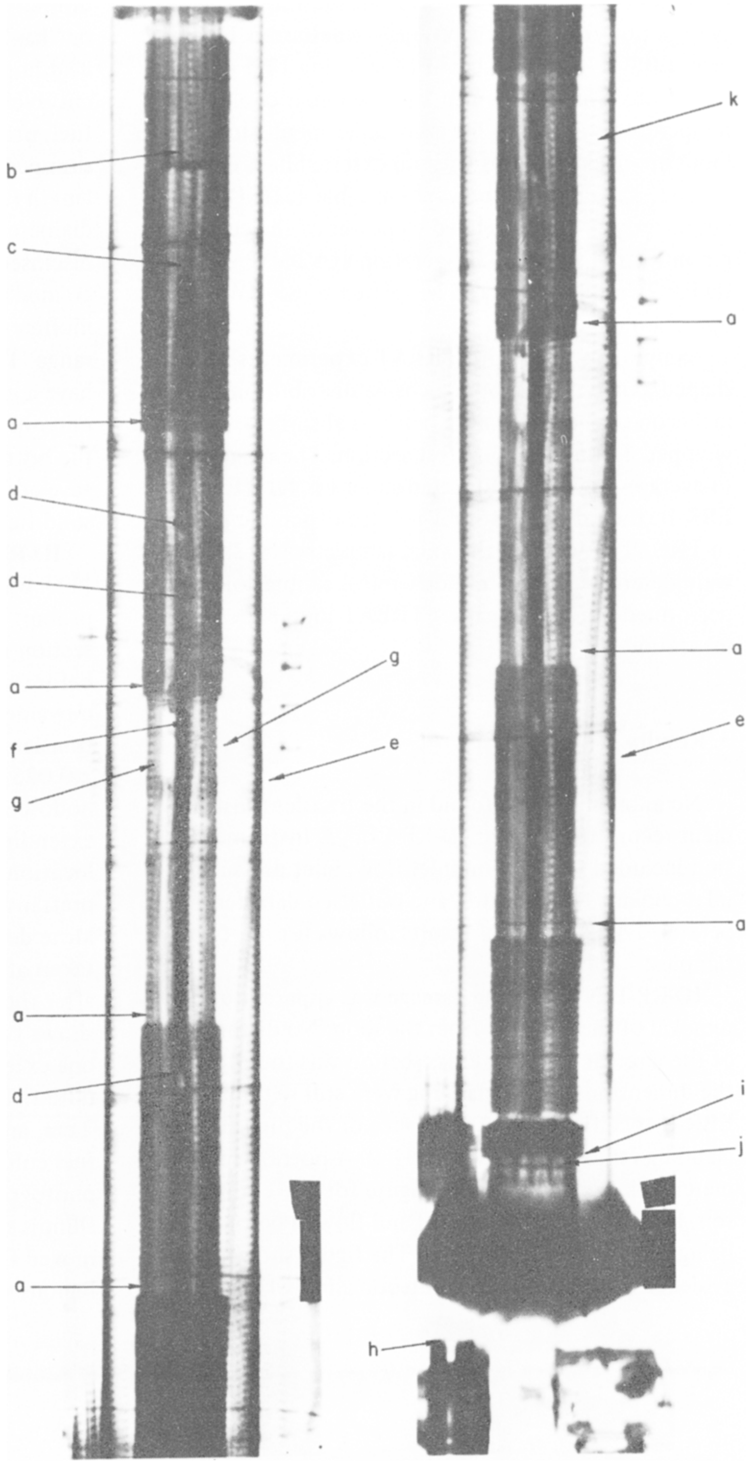

Fig. 3. Post-experiment neutron radiograph of the ID-RP-2 sample in the loop, taken at TREAT (113-598).

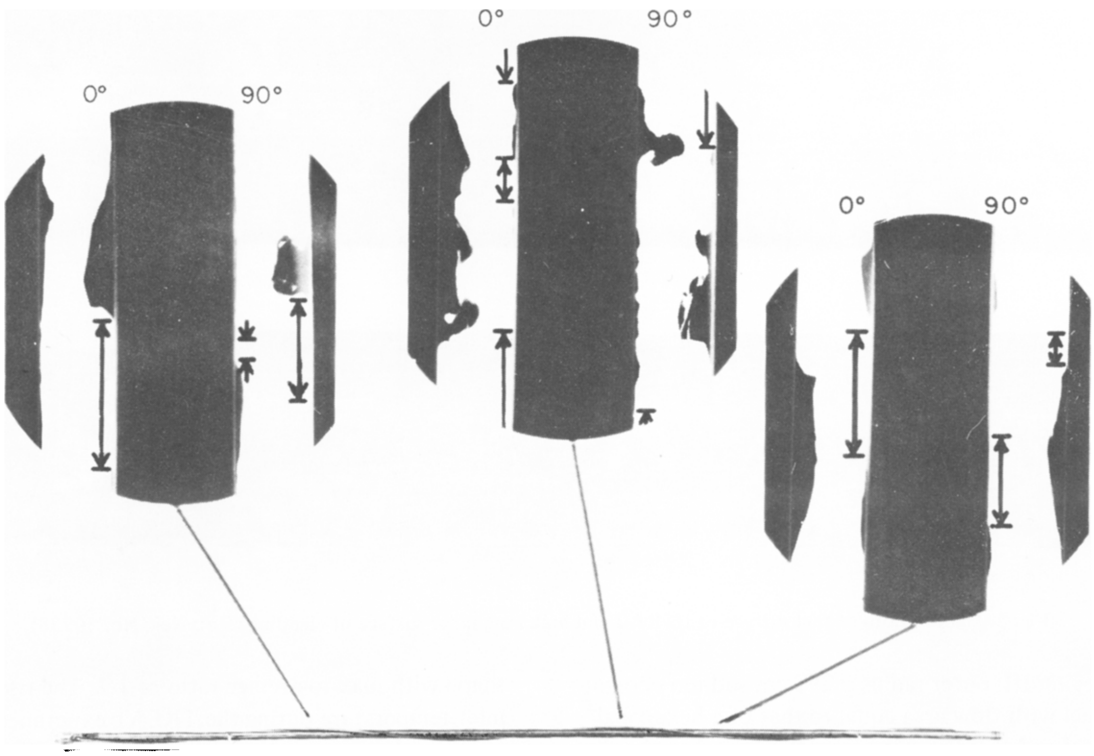

Fig. 4. Post-experiment X-ray photograph of the ID-RP-2 sample, showing fuel voids and slumping against the cladding (113-1217-T-1 - Rev. 1).

the cladding, along with prominent voids. The post-transient cladding dimensions were again within specifications.

The appearance of ID-RP-2 was similar to ID-RP-1, however, there was a greater prevalence of the shiny fuel-cladding eutectic alloy where molten fuel directly contacted the cladding through fissures in the oxide coating of the fuel alloy. This coating has been referred to as an "oxide bag" and may have protected against fuel-cladding interaction, except at places where the oxide film or "bag" had ruptured. Fig. 5 shows typical matching fuel and cladding surfaces [12]. The cladding penetration for ID-RP-2 was a maximum of about 11% of the tubing wall (i.e., 0.0025 cm) and a typical interaction zone is shown in fig. 6.

4.1. *Calculations*

Transient heat transfer calculations of the experiments were performed utilizing an idealized model of meltdown in which fuel slumping against the cladding was assumed to occur abruptly and instantaneously along the length of the pin when the fuel surface at the axial midplane reached 1050°C. The ARGUS transient heat transfer code, a multiregion code written for cylindrical geometry [13] was used, with standard physical properties for the materials in the test section [4].

To convert from the cluster-like geometry of the experiments to the cylindrical geometry of ARGUS, two models were used:

(1) Model I consists of a mockup of an idealized single element cell in a cluster, and has three regions as follows:

Region I; outer radius 0.183 cm, fuel alloy.

Region II; outer radius 0.299 cm, homogenized annular region of fuel-cladding gap, and the stainless steel cladding and spacer wire.

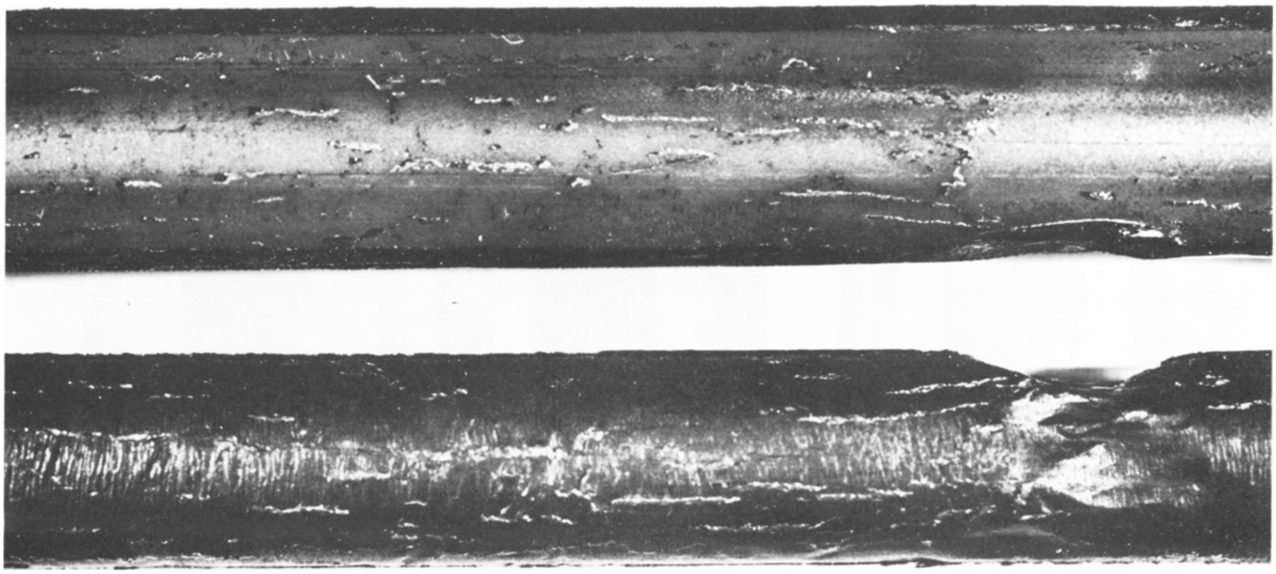

Fig. 5. Macro-photo of fuel surface of ID-RP-2 and matching inner surface of cladding (Met. Neg. No. 50928).

Region III; outer radius 0.297 cm sodium coolant channel with flow area equal to that of a hexagonal channel and the distance across flats equal to the pin diameter plus spacer wire diameter.

Model I calculations were used for two purposes: first, to provide a pessimistic model that would yield conservative calculations of the transient heating of fuel and cladding; and second (because the sodium flow area is about half the true flow area around the fuel pin), to give an indication of the potential severity of a multiple accident in which loss of bond was associated with a local flow reduction to ~ half the design flow rate.

(2) Model II has Regions I and II identical with those of model I. The differences are:

Region III; outer radius 0.519 cm coolant channel with area enlarged to include some of the sodium flowing around the ring of peripheral dummy pins surrounding the fuel sample.

Region IV; outer radius 0.554 cm steel ring mocking up the stainless steel cladding in contact with the sodium of Region III.

Model II calculations were used as the "actual" calculational mockup of the experiment.

In both models, the sample power was shaped axially to correspond to the actual "chopped cosine" shape with max-to-average ratio of 1.1. The rise in inlet temperature during the TREAT experiment due to recirculation of sodium was also included.

An idealized model of meltdown inside the clad was used. The initial fuel-cladding thermal conductance, u, was taken to be 0.28 w/cm^2 - °C. This value was obtained by analyzing the cladding surface transient temperature records from earlier TREAT experiments in which both argon- and sodium-bonded elements were run in pairs, one of each kind in each capsule [6]. This conductance was used until the fuel surface at its axial midplane reached 1050°C, at which point in the calculations the interface conductance was switched to 56 W/cm^2 - °C as an approximation to thermal contact between molten fuel and cladding. Under these conditions, the artificially low heat absorption outside the cladding given in model I would provide an estimate of time-at-temperature for the cladding interface that is more severe than the actual experiment. The actual measurements of transient outlet temperature would be expected to approach those of model II.

4.2. *Calculation results and comparison with experiments*

Model I results indicated the fuel surface cooled

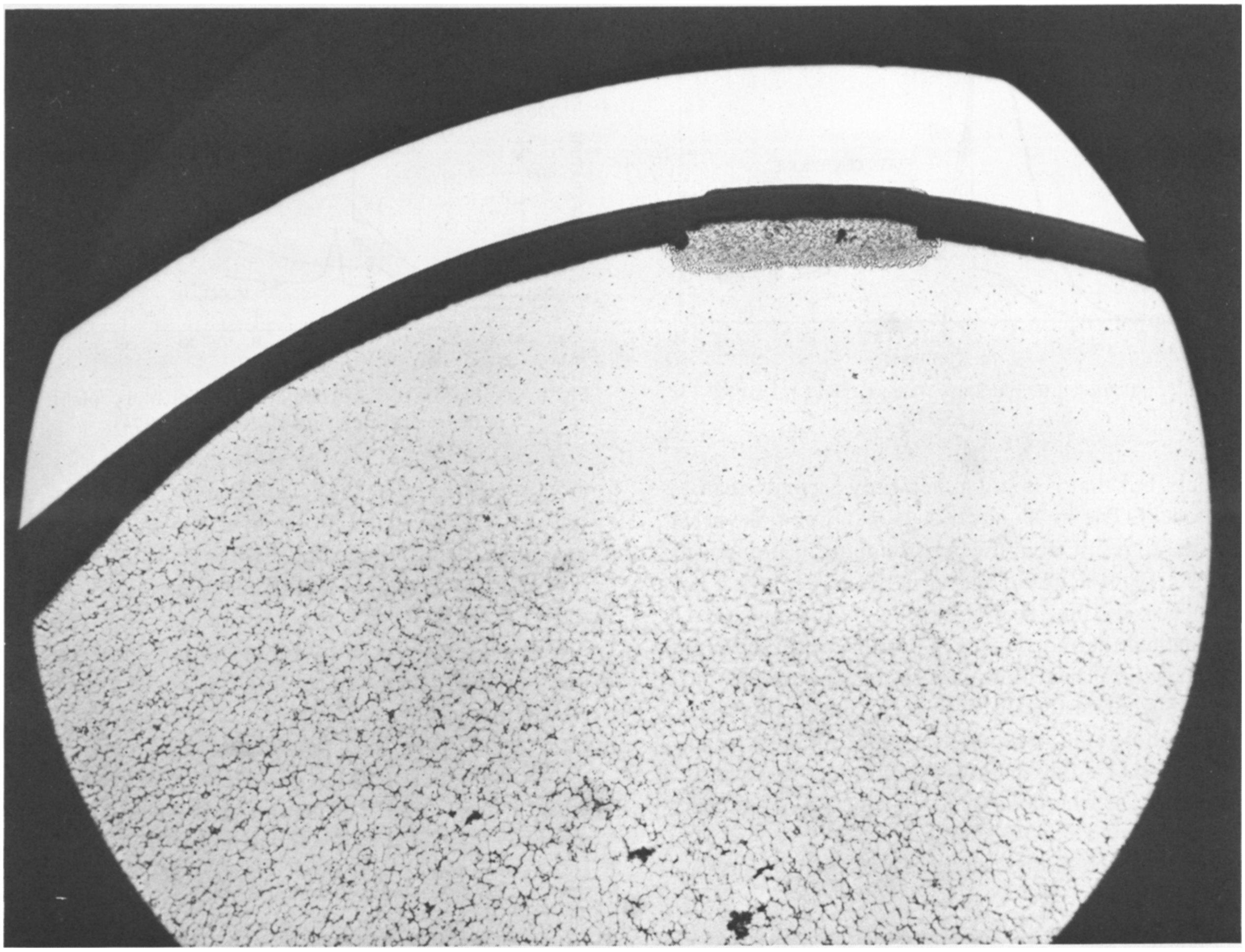

Fig. 6. Micro-photo of cross section through fuel and cladding of ID-RP-2, showing interaction (Met. Neg. No. 50948).

too rapidly in ID-RP-1 after contact * to permit significant penetration of the cladding by eutectic formation, even for penetration rates ~ 0.025 cm/sec typical of attack by molten fuel [5, 6]. Fig. 7 shows calculated temperature traces for fuel centerline and fuel surface at the axial midplane, and for the coolant outlet, for ID-RP-1. The sharp drop of fuel surface temperature calculated once the fuel contacts the cladding is clearly shown. In addition, the high (870°C) calculated outlet coolant spike indicates that incipient coolant boiling could be approached if a flow reduction to ~ 50% of design preceded the loss of bond.

Experimental and calculated temperatures are compared in fig. 8. There is a basic discrepancy of approximately 30°C in the comparison, since the initial calculation temperature was set at 370°C to approximate the actual EBR-II inlet temperature of 371°C, while the actual starting temperature for ID-RP-1 was 400°C. In the early stage of the experiment, prior to slumping, the experimental outlet temperatures are closer to model I than model II; however, the coolant heating associated with the slumping is closer to model II. This result suggests that less thermal mixing occurred than model II pre-

* The maximum cladding inner surface temperature calculated was 745°C.

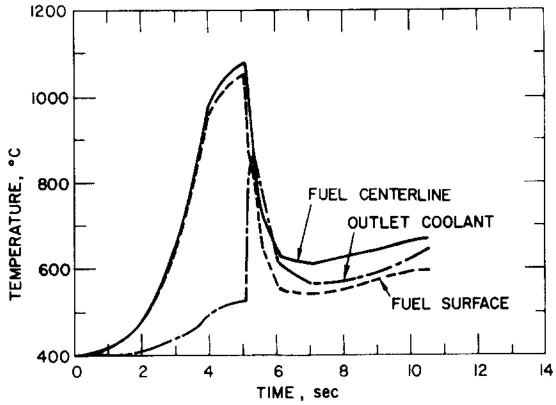

Fig. 7. Calculated temperatures from model I for ID-RP-1 (113-971).

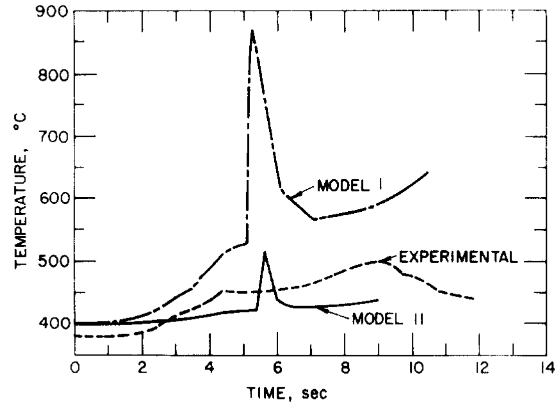

Fig. 8. Comparison of calculated and experimental coolant outlet temperatures for ID-RP-1 (113-974).

dicts, but that it is still more nearly accurate than model I. The small model II temperature spike calculated to result from the post-slumping heat dump to the coolant does not appear in the experimental trace. Because of finite heat absorption by the miniature sheathed thermocouple and possible uncertainties in detailed transient flow patterns, it is not clear from fig. 8 whether the absence of the temperature spike is real (indicating less severe thermal contact between fuel and cladding than that used in the ARGUS calculations), or whether a small spike occurred and was not detected experimentally.

The much more severe transient ID-RP-2 gave a more sensitive test of the calculation model. In contrast with the small (~ 70°C) outlet coolant temperature spike calculated by model II for ID-RP-1, model II predicts an outlet coolant temperature spike ~ 200°C for ID-RP-2. The coolant temperature spike calculated is of sufficiently short duration, however, that the fuel surface temperature is quenched rapidly after it contacts the cladding. This is indicated in fig. 9, which gives calculated fuel centerline and surface temperatures at the axial midplane, and coolant outlet temperatures. The maximum fuel surface temperature attained in fig. 9 is approximately 1090°C. However, the fuel surface has dropped below the

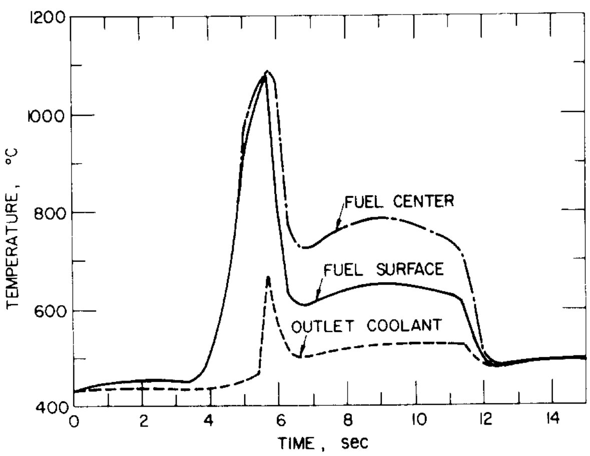

Fig. 9. Calculated temperatures of fuel centerline and fuel surface at the axial midplane, and of outlet coolant from model II, for ID-RP-2 (113-973).

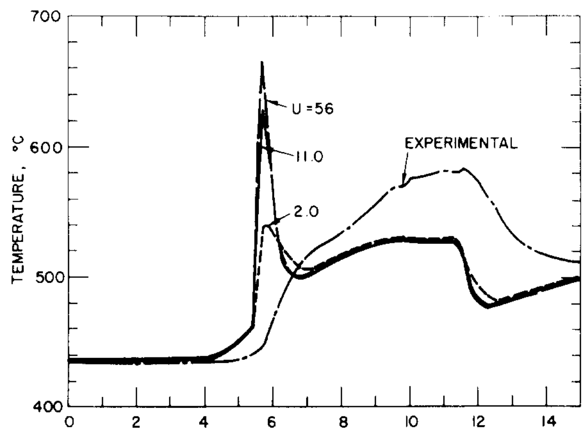

Fig. 10. Comparison of experimental outlet coolant temperatures with temperatures calculated using model II, for three values of post-melting fuel-cladding interface conductance (113-972).

alloy solidus of 1002°C [7] within 0.20 sec after the fuel-cladding contact occurs. The maximum inner cladding surface temperature is 1064°C, and the maximum cladding inner surface temperature duration above 1000°C is 0.25 sec. Under isothermal conditions, the attack rate of stainless steel by molten U-5 wt fission ally just above its liquidus is about 0.025 cm/sec. Hence, the cladding attack for ID-RP-2 can be predicted to be $\gtrsim$ 0.005 cm. It is the time at temperature of the fuel-cladding contact interface after fuel melting that determines the severity of cladding attack. The idealized, abrupt, high-contact conductance model used in the calculations is an upper limit, because it provides for maximum heating of the cladding.

Fig. 10 compares experimental coolant outlet temperatures for ID-RP-2 with outlet temperatures calculated using model II, with the fuel-cladding interface conductance after melting varies as a parameter. Three post-melting conductances were used: 56, 11, and 2 w/cm^2 - °C. From this comparison, it was concluded that the assumption of abrupt, high-contact conductance slumping is much more severe than reality. Comparison of the curve shape of fig. 6 indicates that the assumption of instantaneous fuel-cladding contact along the length of the pin also is too severe. In the absence of direct experimental evidence on the rapidity and degree of coherence of the slumping, it was decided that more detailed modelling of the slumping was not justified for the two loop experiments.

5. Conclusions

The results of calculations, supported by the results of actual TREAT experiments, imply that although fuel melting may accompany a loss-of-bond type failure in an EBR-II driver fuel element, the short term consequences are acceptable from the viewpoint of reactor safety. The fact that fuel remains within the cladding even under higher than normal operating conditions is reassuring. As a consequence the possibility of cladding failure followed by fuel extrusion may be de-emphasized as a credible failure mechanism.

While the results of ID-RP-1 and ID-RP-2 do not define the long term implications of operating with a loss-of-bond type failure it is expected that the re-irradiation of TREAT melted specimens in EBR-II could provide much of the necessary information. Because the specific early stages of the hypothetical loss-of-bond incident are not well defined, they were replaced in both calculations and TREAT experimentation by an assumed (limiting) case of complete argon gas bonding. Comparison of results from calculations and experiment indicate that the relatively simple modelling used represents a conservative description of the temperature rise in the cladding following fuel-cladding contact resulting from the fuel melting.

Examination of the model I calculations and the tests indicate that a loss-of-bond incident associated with flow decreased by $\gtrsim$ 50% could produce coolant temperature approaching incipient boiling.

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